On the History of the Evolution of Light Water Reactor Safety in the United States

by

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(This document represents the work of an individual member of the Advisory Committee on Reactor Safeguards and presents his personal views and interpretations of the various facts and incidents noted in the text.)

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Preface

In casting about trying to develop an approach to an examination of some important aspects of the evolution of light water reactor safety in the United States, I was inevitably drawn to a reliance on that perspective I knew best, that of an ACRS member. Partly because of ACRS self-imposed restrictions on ACRS member interpretation of past ACRS actions, and partly because I decided not to act as an interpreter, and hence a major filter of what information and opinion was to be expressed, I decided to rely heavily on extracts and even complete reproductions of large portions of ACRS minutes, letters and other documents.

This manuscript was deliberately prepared to be fairly lengthy, on the assumption that a shorter version, if desired, could be prepared largely by deletion and compaction of material. Not that the coverage is complete, or even nearly so.

Chapter Two is the central portion of the document. It is intended to provide a historical view of the development of siting policy and the major safety issues which interacted strongly with siting policy, with safety requirements, and with many major changes in the regulatory approach. The other chapters represent a very incomplete selection of the very many important issues and developments in light water reactor safety. And they are presented in much less depth. Coverage of the loss of coolant accident (LOCA-ECCS) has deliberately been abbreviated to cover only a few selected aspects.

It would require a manuscript at least as long as this one to do justice to this topic.

This manuscript, which was completed in the spring of 1978, except for minor editing, does not include any personal assessment of reactor safety nor does it comment on the implications of the Three Mile Island accident. Chapter 1 gives several long excerpts from other publications which relate to the topics of light water reactor safety and societal risk acceptance. These publications were available prior to the spring of 1978 and are intended to provide only a limited sample of opinion.

Acknowledgements

This manuscript was prepared while the author had the benefit of a sabbatical leave from the University of California, Los Angeles and a (second) fellowship from the John Simon Guggenheim Memorial Foundation. During this period (July 1977-June 1978) he was a visitor with the Department de Sureté Nucléaire, Commissariat à l' Energie Atomique, at Saclay and Fontenay-aux-Roses, France; with the Joint Project, International Atomic Energy Agency, Vienna, Austria; and at the Department of Nuclear Engineering, Technion-Israel Institute of Technology, Haifa, Israel, where he held the Isaac Taylor Chair.

Various individuals and groups have contributed in one fashion or another and the author wishes to express his appreciation, hopefully without too many oversights: Donald G. Browne and Thomas E. McKone for editorial assistance; Jolie McNulty for typing; the Staff of the Nuclear Engineering Department at the Technion for typing, xeroxing, and a pleasant environment; the Staff of the Advisory Committee on Reactor Safeguards (ACRS) for finding various documents; Milton Plesset and Stephen Lawroski for strong support of the original idea; Morton Libarkin, Raymond F. Fraley and Dade W. Moeller for struggling through an early, partial draft and supplying detailed suggestions and comments; and the author's family for living for months with boxes of documents at home and for accepting the reductions in leisure time introduced by the writing of this manuscript.

Finally, I would like to express my appreciation for having had the opportunity to work with the members of the ACRS for many years.

Chronology of Some Matters Discussed in Text

Date	Item	Section in Text
1947	Establishment of first Reactor Safeguards Committee	1.2
1950	Issuance of WASH-3, including "Rule of Thumb" siting criterion	2.2
1952	Evolution of concept of containment	2.2
1953	Expression of safety philosophy by Edward Teller	1.2
1953-54	Announcement and approval of Shippingport PWR	2.2
1956	Approval of construction of Indian Point 1	2.2
1956	Exchange of letters between Senator Hickenlooper and AEC Chairman Libby	2.2
1957	Establishment of statutory ACRS.	1.2
1957	First dissent by ACRS member (Abel Wolman) on Plum Brook reactor	2.2
1958	AEC Chairman McCone considers the ACRS overconservative	2.2
1958	Discussion of "maximum credible accident" concept by Clifford Beck	2.2
1958	ACRS reports unfavorably on first proposed site (and reactor concept) for Piqua, Ohio	2.2
1960	ACRS comments on nuclear power plants in California	2.2
1960	ACRS gives unfavorable report to proposed Point Loma reactor and to proposed site for Jamestown reactor	2.2
1960	ACRS report of October 22, 1960 to AEC Chairman McCone on Reactor Site Criteria	2.3
1960	ACRS report of December 13, 1960 to AEC Chairman McCone on Site Criteria for Nuclear Reactors . .	2.3
1961	AEC publication of proposed "Reactor Site Criteria" for public comment	2.3

Chronology of Some Matters Discussed in Text (Continued)

Date	Item	Section in Text
1962	Adoption by AEC of 10 CFR Part 100, Reactor Site Criteria,(without direct inclusion of two major recommendations made by ACRS in report of October 22, 1960).	2.3
1962	Review of several potential sites and reactor concepts for LADWP, culminating in ACRS letter of October 12, 1962	2.4
1962-64	The substitution of engineered safeguards for distance within the framework of Part 100 (San Onofre, Connecticut Yankee, Malibu, and Oyster Creek)	2.4, 2.6
1963	Summary of AEC siting philosophy by Clifford Beck	2.5
1963	Ravenswood	2.7
1964	ACRS report of November 18, 1964 to AEC on Engineered Safeguards	2.6
1964	The seismic design controversy over the Bodega Bay reactor	5.3
1964-65	The seismic design controversy over the Malibu (Corral Canyon) reactor	5.4
1965	Boston Edison's proposed Quincy, Massachusetts reactor, and the ACRS and AEC positions which placed a hold on metropolitan siting	2.7
1965	Dresden 2 and the ACRS report of November 24, 1965 on Reactor Pressure Vessels	2.8
1965	AEC publication of first version of General Design Criteria for comment	3.1
1966-67	Evolution of requirements for improved integrity of pressure vessels	2.8, 2.10, 3.2
1966	Recognition of the China Syndrome problem and the correlation of containment failure with core melt	2.8, 2.9
1966	ACRS reports of August 16, 1966 on Dresden 3 and Indian Point 2 recommending major improvements in primary system integrity to reduce the probability of a LOCA and major improvements in ECCS, leading to a revolution in LWR safety. .	2.9

Chronology of Some Matters Discussed in Text (Continued)

Date	Item	Section in Text
1966	ACRS decision not to issue report on Problems Arising From Primary System Rupture when AEC proposed establishment of Task Force to study the matter	2.13
1966	The ACRS safety research letter of October 12, 1966 on matters related to large scale core melt, LOCA-ECCS, and other concerns	2.13
1966	Speech by Clifford Beck reflecting AEC position on metropolitan siting in April, 1966	2.15
1966	Initiation of Heavy Section Steel Safety Research Program	3.2
1967	Revision of safety research program in LOFT as part of new emphasis on ECCS	6.3
1967	Brown's Ferry review and the asterisked items	2.10, 6.1
1967	Proposal for probabilistic approach to safety evaluation by F. R. Farmer	2.14
1967	AEC publication of 2nd version of General Design Criteria for comment	3.1
1967	A finding against the Burlington site by the ACRS and the Regulatory Staff	2.10
1967	AEC rejection of proposed limit on population density around reactor site	2.10
1967-68	Separation of Protection and Control	4.2
1967	Task Force Report on Power Reactor Emergency Cooling	2.13
1968	ACRS letter of February 26, 1968 on Task Force Report	2.13
1968	Removal of core-catcher from Indian Point 2	2.10, 2.11, 2.13
1968	Development of ACRS Site-Population Index	2.11, 2.12
1968	Zion	2.11
1968	Bolsa Island, Trap Rock, Bowline and Montrose	2.12

Chronology of Some Matters Discussed in Text (Continued)

Date	Item	Section in Text
1968	The Oyster Creek review and the development of more stringent quality assurance requirements	3.3
1968	The steam-line break accident	3.4
1969	The birth of ATWS	4.3
1969	ACRS letter of November 12, 1969 on Reactor Safety Research	6.3
1969	ACRS draft report of July, 1969 on Location of Power Reactors at Sites of Population Density Greater than Indian Point-Zion	2.12
1969-1973	Newbold Island	2.17
1970	Meetings of ACRS and Regulatory Staff with representatives of nuclear industry concerning metropolitan siting	2.12
1970	ACRS report on Use of Water-Cooled Power Reactors at Sites More Densely Populated than Those Employed to Date (never issued)	2.12
1970	Initiation of Safety Guides (later renamed Regulatory Guides) as means of defining technical positions acceptable to Regulatory Staff and ACRS	3.1
1970-71	Study by duPont on molten core retention and ACRS letter of January 11, 1971 to Milton Shaw on safety research program	2.13
1971	Adoption of General Design Criteria as Appendix A to 10 CRF Part 50	3.1
1971	Interim Acceptance Criteria for ECCS	6.2
1971-72	Revision of ACRS report on Acceptance Criteria for ECCS	6.2
1972	Letter of February 3, 1972 from Milton Shaw to ACRS opposing research and development on molten core retention	2.13
1972	ACRS letter of February 10, 1972 on Water Reactor Safety Research	6.3
1972	First ACRS letter on Generic Items	6.1

Chronology of Some Matters Discussed in Text (Continued)

Date	Item	Section in Text
1973	AEC Publication of Seismic and Geologic Siting Criteria	5.5
1973	Reorganization of safety research in AEC, placing LWR safety research under H. Kouts	6.3
1973	WASH-1270, A Staff Position on ATWS	4.3
1973	Final Acceptance Criteria for ECCS	6.2
1974	Establishment of NRC, including its own safety research program	6.3
1974	Issuance of WASH-1285, a report by the ACRS on pressure vessel integrity	3.2
1974	Issuance of draft WASH-1400	7.1
1975	Browns Ferry Fire	6.1
1975	ACRS letter of June 12, 1975 to R. V. Carlone on safety research	2.13
1975	Site population density guidelines adopted in NRC Standard Review Plan	2.18
1975	Final version of WASH-1400	7.1
1976-78	Consideration of Class 9 accidents in review of floating nuclear power plants	7.1
1977	ASLB Appeal Board Ruling on the matter of New England Power and on Seabrook concerning emergency evacuation and definition of exclusion and low population zone boundaries	2.19
1977	ASLB Hearing Board examination of Class 9 accidents in connection with Black Fox	2.19
1978	First annual report by ACRS to Congress on NRC Safety Research Program	7.1
1978	NUREG-0438, Report to Congress by NRC on Plan for Research to Improve the Safety of Light Water Nuclear Power Plants	7.1

CHAPTER I

1.1 SOME OPINIONS ON LWR SAFETY AND SOCIETAL RISK

This study represents an effort to examine in some detail the historical evolution of several aspects of light water reactor safety in the United States. The perspective is that of one with access to the records of the Advisory Committee on Reactor Safeguards (which are generally available with minor exception) but no access to the files of the Regulatory Staff, the nuclear utilities, the reactor vendors, the intervenors, etc. This study in no way represents an assessment of the safety of light water reactors, nor does it represent an attempt to comment on risks from nuclear reactors. Furthermore, this study does not attempt to examine the technical facets of most of the issues discussed, except as such an examination arises coincident to the historical discussion.

To provide some admittedly incomplete perspective on light water safety, the study begins with several excerpts from existing documents.*

1. An excerpt from UCLA-ENG-7777, Final Report in an NSF-funded study entitled, "A General Evaluation Approach to Risk Benefit for Large Technological Systems and its Application to Nuclear Power" (1977).

2. "The Generic Safety Issues of Pressurized Water Reactors," Health and Safety Executive, United Kingdom (1977).

3. F. R. Farmer, Accident Probability Criteria, Journal of the Institution of Nuclear Engineers, April 1975, Vol. 16.

4. D. Okrent, Testimony before the California Legislature Assembly Committee on Energy and Diminishing Materials, October 29, 1975, and U. S. Congressional Joint Committee on Atomic Energy (94th Congress, Second Session, 1976).

<u>Excerpt from UCLA-ENG-7777 (Okrent 1977)</u>

<u>Introductory Comments on Light Water Reactor Safety</u>

Nuclear power plant accidents differ from those in conventional power plants in that they can potentially release significant amounts of radioactivity to the environment. While very large amounts of radioactivity are generated by the fission process in the uranium dioxide fuel in a nuclear plant, the bulk of this radioactivity (about 98%) remains in the fuel as long as

*The first excerpt has been amplified, using published information from WASH-1400.

the fuel is adequately cooled. For large amounts of radioactivity to be released from the fuel, it must be severely overheated and essentially melt. Based on this knowledge, the major types of nuclear power plant accidents that have the potential to cause large releases of radioactivity to the environment, have for some time been identified. Attempting to prevent such accidents and to mitigate their potential consequences have been the primary objectives of nuclear power plant safety design.

Gross heat imbalance in the fuel in the reactor core might occur in the following ways:

(a) The occurrence of a loss-of-coolant event will allow the fuel to overheat (due to decay heat), unless emergency cooling water is supplied to the core.

(b) Overheating of the fuel can result from transient events that cause the reactor power to increase beyond the heat-removal capacity of the reactor cooling system, or the cooling rate.

In a loss-of-coolant accident (LOCA), the rupture would allow the high-pressure and high-temperature cooling water to flash to steam and blow down into the containment building. To cope with this event, a system, called engineered safety features (ESF), is provided in each plant. A number of the engineered safety features, as well as the physical processes, act to reduce the amount of radioactivity released to the environment, should either a LOCA or a transient event result in a significant release of radioactivity from the reactor core. For instance, a containment building is provided to contain the radioactivity released from the fuel and to delay and reduce the magnitude of release to the environment.

In early power reactors the power level was about one-tenth that of today's large reactors. It was thought that core melting in those low-power reactors would not lead to melt-through of the containment. Further, since the decay heat was low enough to be readily transferred through the steel containment walls to the outside atmosphere, it could not overpressurize and fail the containment. Thus, if a LOCA were to occur, and even if the core were to melt, the low-leakage containments that were provided would have permitted the release of only a small amount of radioactivity.

However, as reactors grew larger, several new considerations became apparent. The decay heat levels became so high that the heat could not be dissipated through the containment walls. Further, in the event of accidents, concrete shielding was required around the outside of the containment to prevent overexposure of persons in the vicinity of the plant. Finally, it became likely that a molten core could melt through the thick concrete containment base into the ground. Thus, new sets of requirements came into being.

Emergency core cooling systems were needed to prevent core melting which could, in turn, cause failure of all barriers to the release of radioactivity. Systems were needed to transfer the core decay heat from the containment to the outside environment, in order to prevent the heat from producing internal pressures high enough to rupture the containment. Finally, systems were needed to remove radioactivity from the containment atmosphere in order to reduce the amount that could leak into the environment.

The major goal behind these changes was to attempt to provide safety features designed so that the failure of any single barrier would not be likely to cause the failure of any of the other barriers. For example, emergency cooling systems were installed to prevent the fuel from melting, and thereby protect the integrity of the containment if the reactor coolant system were to rupture. Other features were added, e.g., additional piping restraints and protective shields, to lessen the likelihood of damage that could result from pipe whip following a large pipe break. Knowledge that large natural forces, such as earthquakes and tornadoes, could cause multiple failures led to design requirements that attempted to reduce the likelihood of dependent failures from such causes.

The net result of the addition of ESF's in current large reactors was to reduce the likelihood of accidents that could have significant public impact. Nevertheless, there remains some probability that an accident might lead to the melting of the core and to the subsequent release of large amounts of radioactivity from the containment.

Studies prior to WASH-1400 have indicated that a core meltdown in a large reactor would likely lead to failure of the containment.

WASH-1400* analyzed such relationships and found that the containment failure modes, their timing, and the potential radioactive release depend strongly on the operability of the various ESFs.

WASH-1400 predicts that the core could melt through the bottom of the reactor vessel and the thick, lower concrete structure of the containment about half to one day after the accident, thus providing considerable time for radioactive decay, washout, plateout, etc., to reduce the radioactivity in the containment atmosphere if containment integrity had not been previously violated. Furthermore, most of the gaseous and particulate radioactivity that might be released would, in this case, be discharged into the ground, which acts as a filter, thus reducing the radioactivity released to the above-ground environment. Accidents that would follow this path are thus characterized by lesser releases and consequences. In plants that have relatively large volume containments, the melt-through path was found to represent the most likely course of the accident.

Following this melt-through, there would be a possibility of ground water contamination through a long-term process of leaching the radioactivity from the solidifying mass of fuel, soil, etc. An estimate of the nature and timing of the leaching processes and the potential contamination levels that could result is presented in Appendix VII of WASH-1400. The leaching and contamination processes would occur over an extended period of time (few or many years, depending on the particular radioactive species) and hence the potential contamination levels should not be substantially larger than the maximum permissible concentrations (MPC). The concentrations could potentially be controllable to even lower levels. Therefore, the potential from ground water contamination has been assessed to have only a small contribution to the overall risk.

Containments may also fail by overpressure, resulting from various noncondensible gases released within the containment due to core melting. For small containments, the pressure due to the combinations of these gases would represent the most likely path to containment failure. Such failures would most likely occur in the above-ground portion of the containment, several hours from the time of core melt.

*U.S. Nuclear Regulatory Commission, "Reactor Safety Study, An Assessment of Accident Risks in the U.S. Nuclear Power Plants," WASH-1400 (NUREG-75/014), Washington, D.C., October 1975.

In the course of a potential core meltdown, there would be conditions which would have the potential to result in a steam explosion that could rupture the reactor vessel and/or the containment.* These conditions may occur: (i) when molten fuel would fall from the core region into water at the bottom of the reactor vessel, or (ii) when it would melt through the bottom of the reactor vessel and fall into water in the bottom of the containment. These modes of containment failure were predicted to have lower probabilities of occurrence.

Reactor Transients

In general, the term "reactor transient" aplies to any significant deviation of the key reactor operating parameters from their normal operating values.

Transients may occur as the consequence of an operator error or due to the malfunction or failure of equipment. Many transients are handled by the reactor control system, which returns the reactor to its normal operating condition. Others would be beyond the capability of the reactor control system and require reactor shutdown by the reactor protection system in order to avoid damage to the reactor fuel.

In safety analyses, the principal areas of interest are: (i) increases in reactor core power (heat generation), (ii) decreases in coolant flow (heat removal), and (iii) reactor coolant-system pressure increases. Any of these could potentially result from a malfunction or failure, and they represent a potential for damage to the reactor core and/or the pressure boundary of the reactor cooling system (RCS).

It should be noted that the kinetics of LWRs are relatively sluggish; also, cores are slightly undermoderated, and any further reduction in water density (as from increase in power and temperature rise) is strongly negative in reactivity effect. Further, fuel heating causes a negative Doppler effect. These factors all tend to reduce the vulnerability of LWRs to power excursion type transients, and thus make overpower conditions less important than inadequate heat removal as a possible cause of fuel melting.

*The term "steam explosion" refers to a phenomenon in which the fuel would have to be in finely divided form and intimately mixed with water, so that its thermal energy could be efficiently and rapidly deposited in the water, thus creating a large amount of steam.

Finally, it is noted that a light water reactor core cannot explode in the usual sense attributed to an atomic bomb.

Probability of Core Melt

The WASH-1400 study was performed on two actual plants, one PWR and one BWR, to assess accident probabilities. For each reactor, possible accident sequences were collected into radioactivity-release categories and the overall probability of each release category was estimated. A summary of the results obtained for release probabilities is given in Table 9.1.

The probability side of the study is based on the fault-tree/event-tree methodology. The authors of the study believe that some of the particular characteristics of the reactor case (e.g., no serious risk without core melt, knowledge of engineering design of the reactor) make it practical to handle the large number of theoretically possible accident sequences and to cut this array down to a manageable number. Further, according to the authors, it is not essential to include all accident sequences, but only to identify enough of the higher-probability sequences, so that the overall probability of a given level of release becomes insensitive to the addition of more low-probability sequences.

However, for the assessments of low-probability events, whether one is considering reactor core meltdown, failure of a specific large dam, release of hugh quantities of liquified natural gas in port, or possibly catastrophic, unanticipated effects from biological research or some new vaccine or drug, uncertainties do exist.

With regard to WASH-1400, the question remains, "Are there undiscovered accident sequences with large probabilities?" More specifically, questions have been raised concerning the WASH-1400 treatment of common-mode failures, reactor aging, human errors, fires, earthquakes, and sabotage, among others, in ascertaining the probability of core meltdown. Questions have also been raised concerning the quantification of the uncertainty in the best-estimate results, and there is a considerable body of opinion that the stated uncertainty range, plus or minus a factor of about five in the core-melt estimates, has not been validated and may be too small.

On the other hand, while it is impossible to demonstrate that the estimated occurrence frequency of core-melt of 1 in 20,000 per reactor year is correct, none of the critics has demonstrated that it is in error. Some reviewers have taken the point of view that this occurrence frequency cannot be low by more than a factor of ten, due to the absence of a LOCA, let alone a core melt, in about two thousand reactor years of experience with reactors in the Navy and commercial power reactors.

While one cannot say with absolute certainty that some, as yet undiscovered, accident sequence will not yield still far greater core-melt frequencies, the chance of overlooking such a large-probability event is small, according to others.

Overall probabilities of the release categories are usually determined by a small number (5 to 10 in each category) of dominant sequences (i.e., high probability sequences), and many of these are single-failure events. Also, as the summation of the event probabilities gets large, it tends to reduce the sensitivity of the probability calculations to unperceived common-mode failures.

Nevertheless, although the job was done in a workman-like way, many of the underlying facts, which must be known to accurately predict the course of an accident, are lacking. Therefore, the quantitative estimates of the probability of the various accident chains must be viewed with some reservations.

Consequences of Core Melt

On the consequence side, the first step is to characterize the nature and amount of radioactivity release for each release category. Table I covers the essentials. The most severe categories for both PWRs and BWRs involve release of about half of the total core inventory.

The next steps are transport of radioactivity through the atmosphere with associated plume spread, meander, and depletion processes, and calculation of inhalation, whole body, etc., doses. These transport and exposure calculations were done for siz hypothetical sites, for which the meteorology and demography were synthesized from the parameters of actual nuclear plant sites of the same type (ocean-front site, river-valley site, etc.). Descriptions of the methods are given in Appendix VI of WASH-1400, but essentially no dose results are included. Instead, the calculation is carried forward to health effects and land-contamination effects, and tnese final results are presented. The various probability/consequence plots can be shown as in Figures 9.1 and 9.2; i.e., for the single reactor case and with error-bands on the "average" curve shown.

The consequences reported in WASH-1400 can also be summarized as in Tables 9.2 and 9.3 for a single reactor or as in Tables 9.4 to 6 for 100 reactors. The estimates in the Tables have uncertainty ranges similar to those in Figures 9.1 and 9.2.

Among the principal questions raised with regard to the consequences presented in WASH-1400 are the following:

Is the estimate of health effects, due to low-level radiation, sufficiently conservative? Has the efficiency of evacuation and decontamination procedures been over-estimated? Should not the risk be given for each individual site rather than some weighted average, since, for the worst accident, the most highly populated sites could have 100 to 1,000 times as many early fatalities as the remote sites? Are the potential effects on drinking water and other long-term dose commitment effects treated adquately?

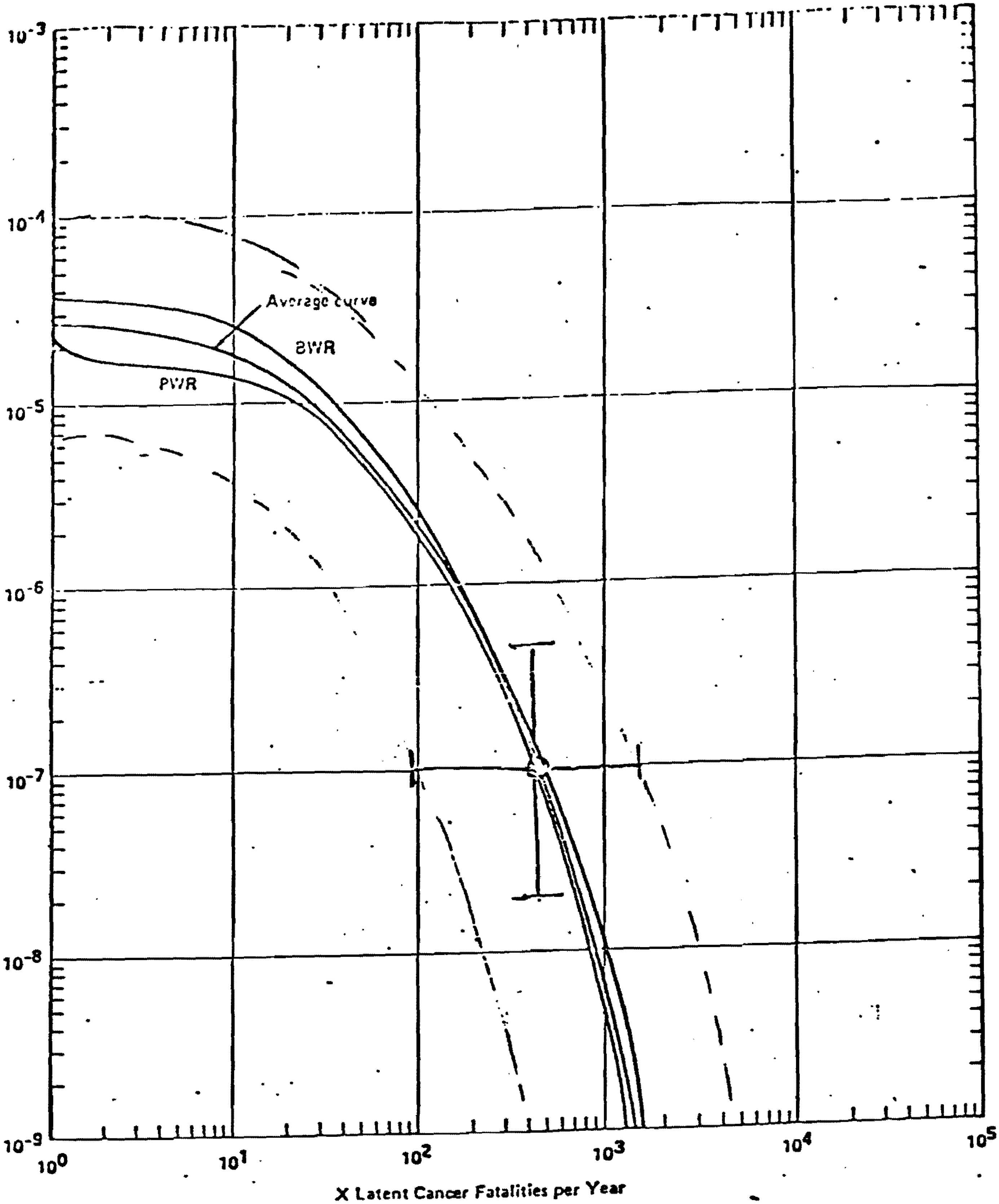
Taking a rough "weighted mean" of the various criticisms, it appears that the consequences might shift toward more severe consequences, particularly if the "linear" theory of radiation effects is valid; however, the shift may not be much outside the present error-band on consequences.

Addendum to discussion in UCLA ENG-7777

WASH-1400 included a considerable section on other risks in society, and some figures and tables are reproduced from the report, together with three figures which given the conditional probability of early or latent mortality as a function of distance from the reactor for the largest radioactivity release considered, a PWR release.

Comments and Criticism Regarding the Reactor Safety Study WASH-1400)

A large number of comments were made on Draft WASH-1400. According to Appendix XI of WASH-1400, about 90 organizations and individuals provided comments totaling about 1,800 pages. This Appendix is entitled "Analysis of Comments on the Draft WASH-1400 Report," and presents the views of its authors on the significance of the comments and how they were dealt



e: Approximate uncertainties are estimated to be represented by factors of 1/6 and 3 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

IGURE 1.1 Complementary cumulative distribution function for latent cancer fatalities per year.

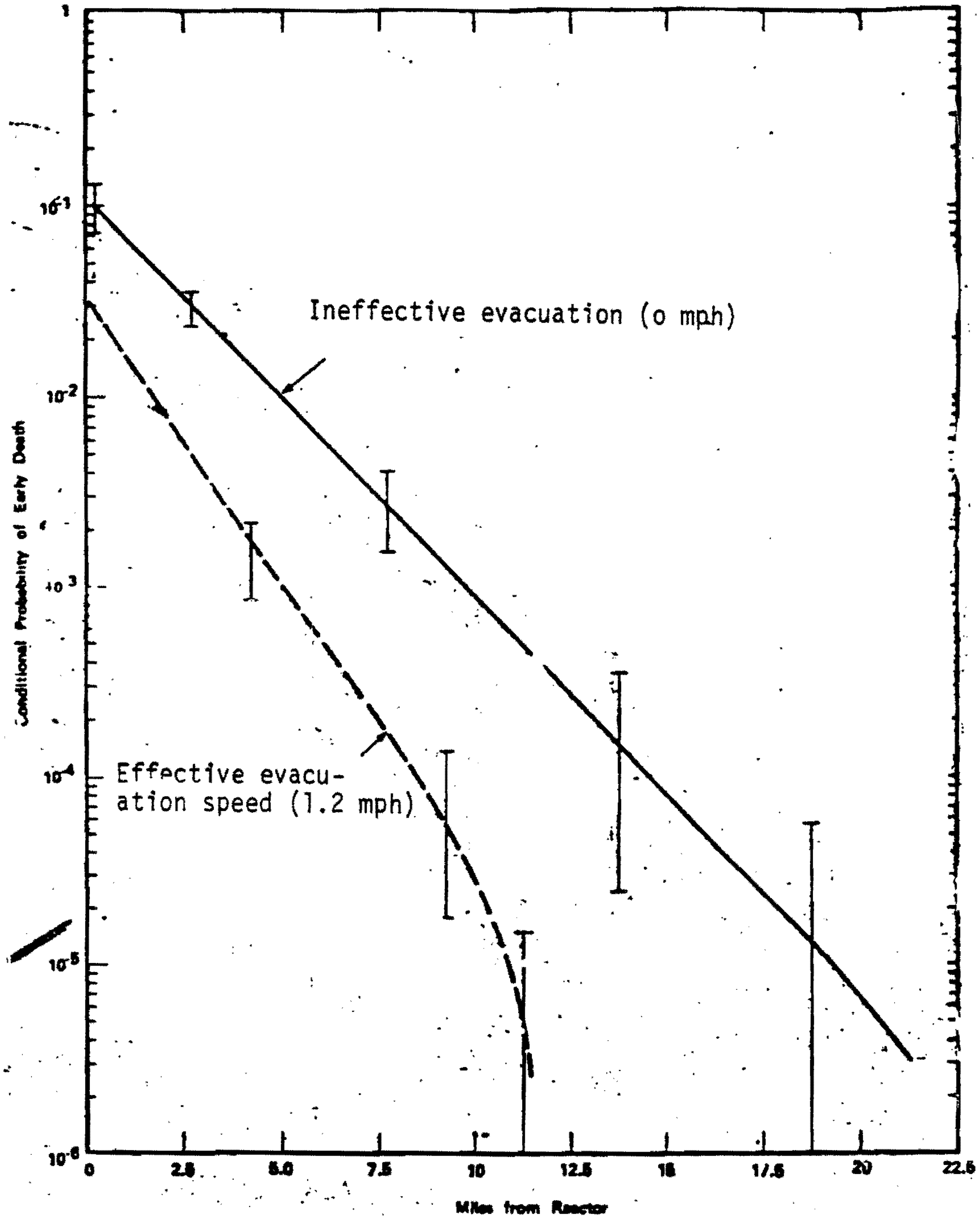


Figure 1.2 Conditional probability of early death as a function of distance from reactor for three effective evacuation speeds given a PWR-1A release.

(a) Approximately, absolute mortality probabilities are 10^{-6} per reactor year times quoted values.

(b) The error bars denote the variation in the mean values for the six meteorological data sets.

(c) For effective evacuation speeds of 4.7 and 7 mph, the conditional probability of early death is effectively zero within 25 miles.

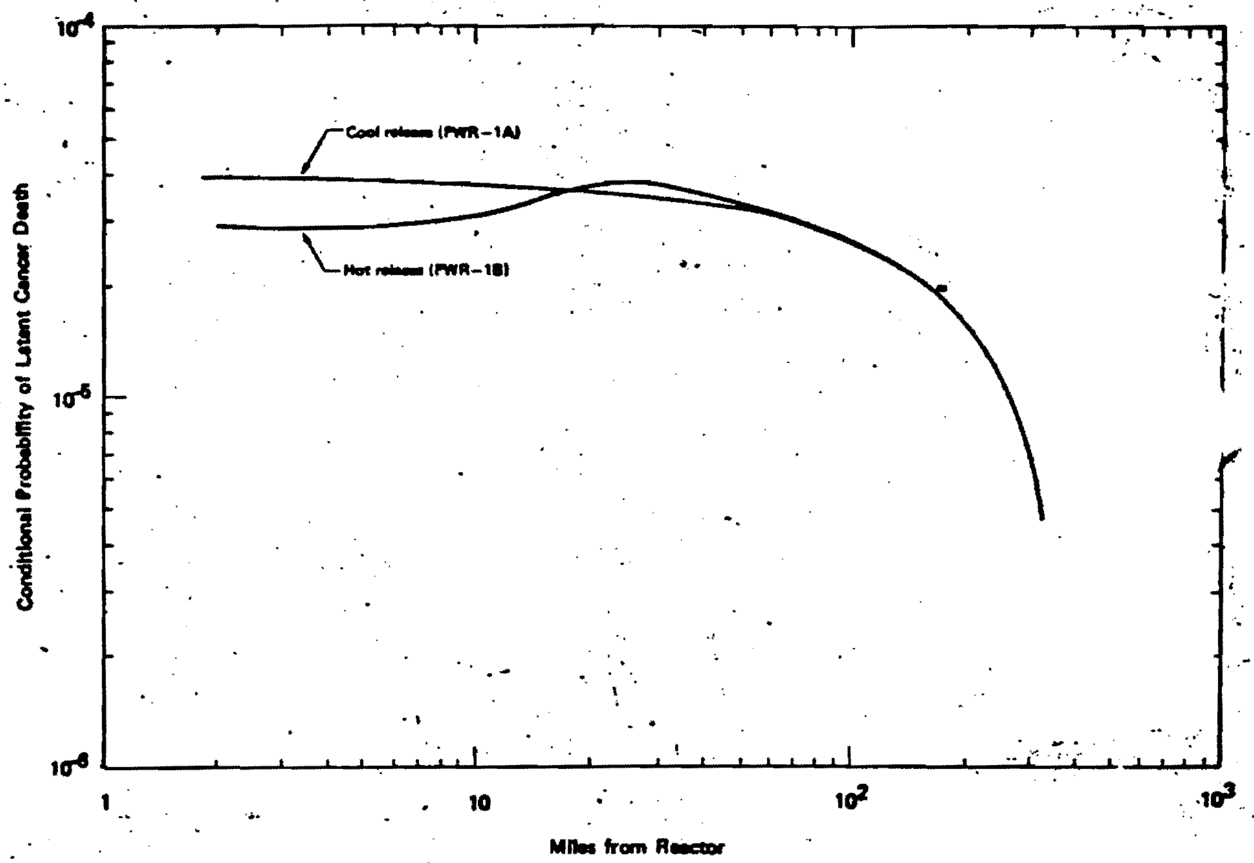


Figure 1.3 **Conditional probability of latent cancer death given a PWR-1A or PWR-1B release. (Approximately, absolute mortality probabilities are 10^{-6} per reactor year times stated ones).**

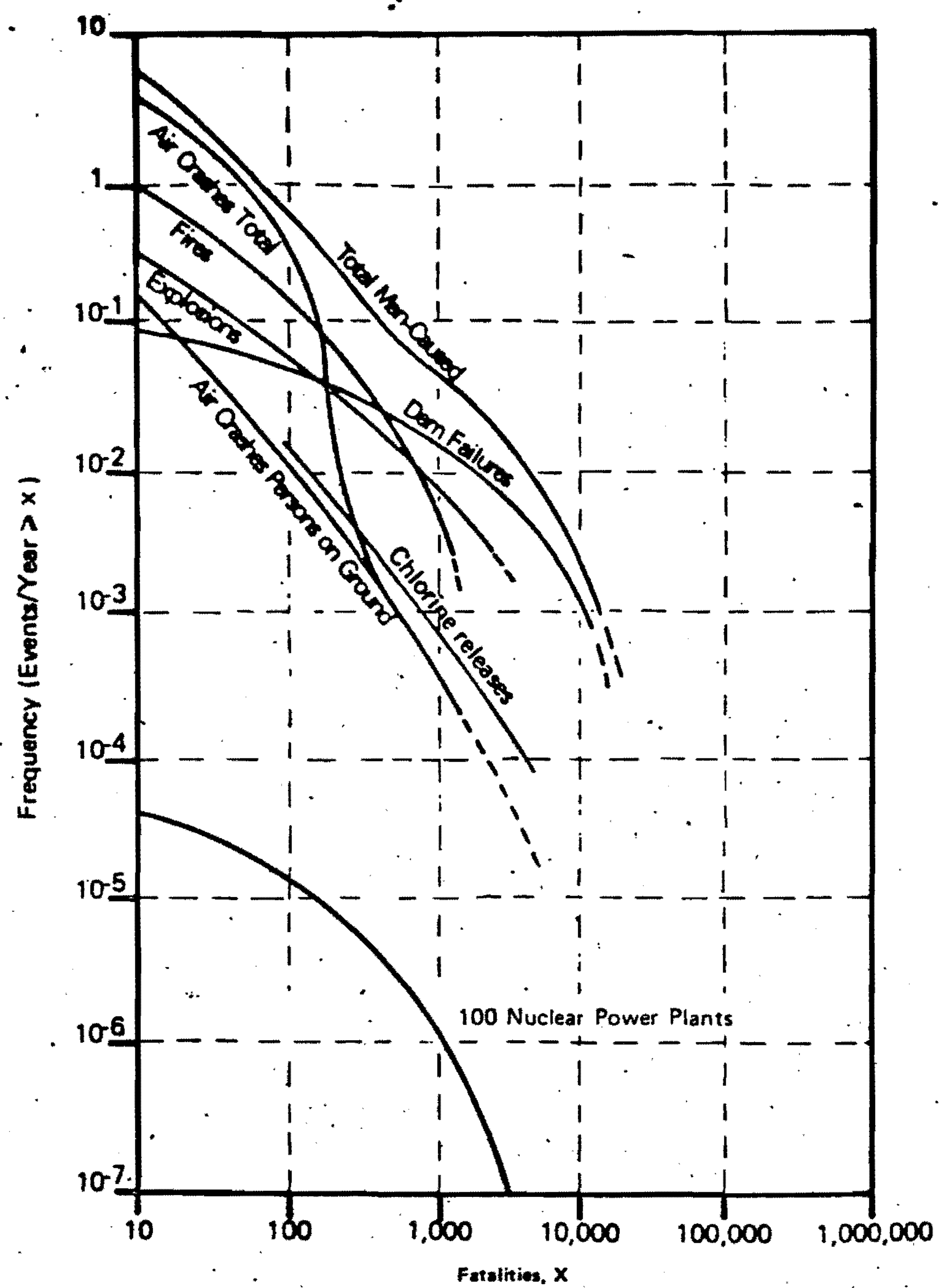


Figure 1.4 Frequency of Man-Caused Events Involving Fatalities.

Notes: 1. Fatalities due to auto accidents are not shown because data are not available for large consequence accidents. Auto accidents cause about 50,000 fatalities per year.

2. See section 6.4 for a discussion of confidence bounds applicable to the non nuclear curve. See section 5.5 for the confidence bounds on the nuclear curve.

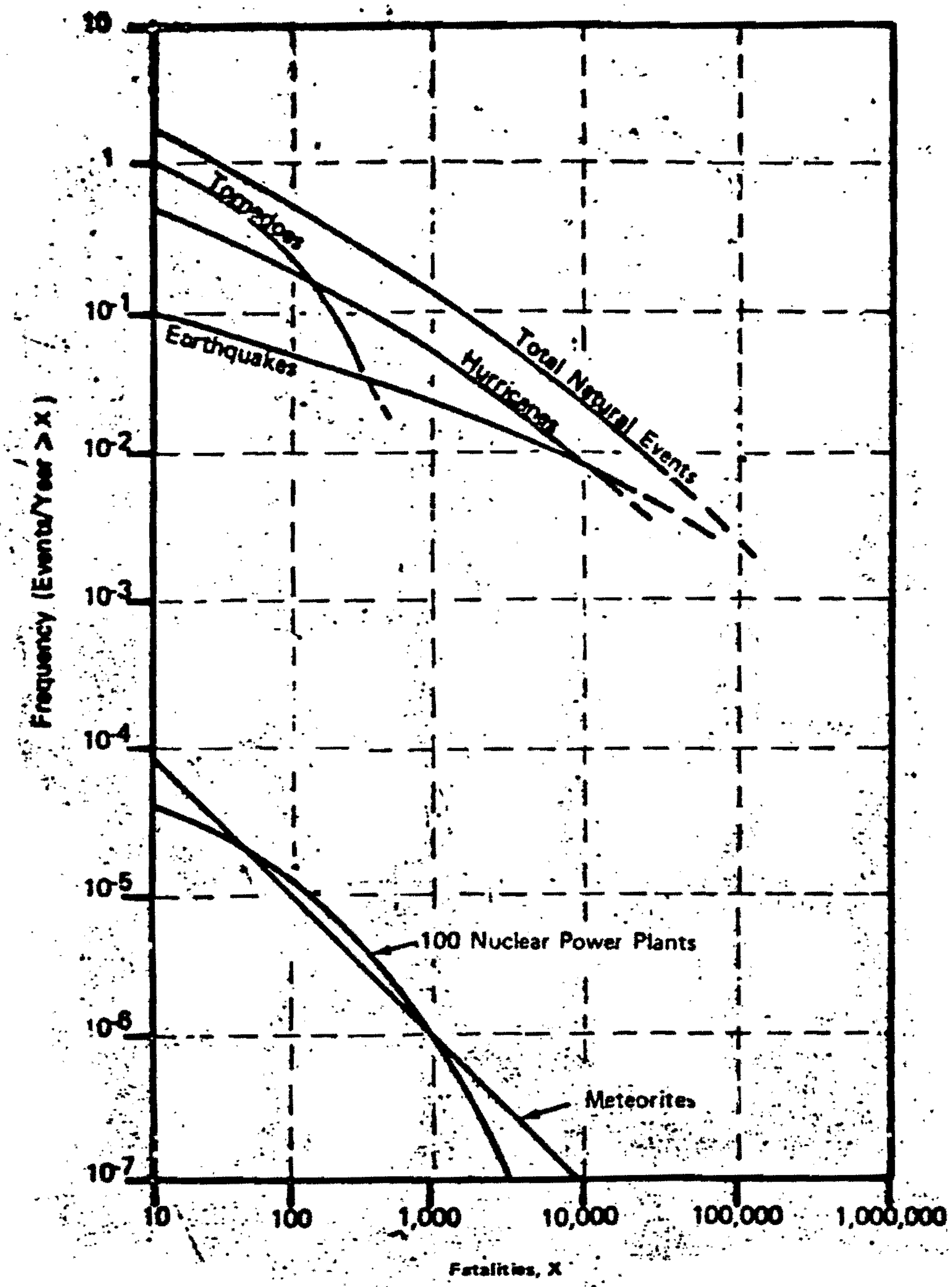


Figure 1.5 Frequency of Natural Events Involving Fatalities.

TABLE 1.1 SUMMARY OF ACCIDENTS INVOLVING CORE

RELEASE CATEGORY	PROBABILITY per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED[a] Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	9×10^{-7}	2.5	0.5	1.0	25	520[d]	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0	N/A	0.3	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	10.0	10.0	1.0	0	N/A	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A	0	N/A	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	N/A	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR 1	1×10^{-6}	2.0	2.0	1.5	25	130	1.0	7×10^{-3}	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR 2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR 3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	3×10^{-3}
BWR 4	2×10^{-6}	5.0	2.0	2.0	25	N/A	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR 5	1×10^{-4}	3.5	5.0	N/A	150	N/A	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

(a) A discussion of the isotopes used in the study is found in Appendix VI. Background on the isotope groups and release mechanisms is found in Appendix VII.

(b) Includes Mo, Rh, Tc, Co.

(c) Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.

(d) A lower energy release rate than this value applies to part of the period over which the radioactivity is being released. The effect of lower energy release rates on consequences is found in Appendix VI.

Table 1.2 Approximate Average Societal and Individual Risk Probabilities per Year from Potential Nuclear Plant Accidents[1].

Consequence	Societal	Individual
Early Fatalities[2]	3×10^{-3}	2×10^{-10}
Early Illness[2]	2×10^{-1}	1×10^{-8}
Latent Cancer Fatalities[3]	7×10^{-2}/yr	3×10^{-10}/yr
Thyroid Nodules[3]	7×10^{-1}/yr	3×10^{-9}/yr
Genetic Effects[4]	1×10^{-2}/yr	7×10^{-11}/yr
Property Damage ($)	2×10^{6}	---

Note: (1) Based on 100 reactors at 68 current sites.

(2) The individual risk value is based on the 15 million people living in the general vicinity of the first 100 nuclear plants.

(3) This value is the rate of occurrence per year for about a 30 year period following a potential accident. The individual rate is based on the total US population.

(4) This value is the rate of occurrence per year for the first generation born after a potential accident; subsequent generations would experience effects at a lower rate. The individual rate is based on the total US population.

Table 1.3 Immediate Consequences of Reactor Accidents for Various Probabilities for 100 Accidents.

	Consequences				
Chance Per Year	Early Fatalities	Early Illness	Total Property Damage $\$10^9$	Decontamination Area Square Miles	Relocation Area Square Miles
One in 200[a]	<1.0	<1.0	<0.1	<0.1	<0.1
One in 10,000	<1.0	300	0.9	2000	130
One in 100,000	110	300	3	3200	250
One in 1,000,000	900	14000	8	(b)	290
One in 10,000,000	3300	45000	14	(b)	(b)

(a) This is the predicted chance per year of core melt considering 100 reactors.
(b) No change from previously listed values.

Table 1.4 Delayed Consequences of Reactor Accidents for Various Probabilities for 100 Reactors.

Chance (per year)	Consequences		
	Latent Cancer Fatalities[2] (per year)	Thyroid Nodules[2] (per year)	Genetic Effects[3] (per year)
1 in 200[1]	< 1.0	< 1.0	< 1.0
1 in 10,000	170	1,400	25
1 in 100,000	460	3,500	60
1 in 1,000,000	860	6,000	110
1 in 10,000,000	1,500	8,000	170
Normal Incidence	17,000	8,000	8,000

Note: (1) This is the predicted chance per year of core melt for 100 reactors.

(2) This rate would occur approximately in the 10 to 40 year period after a potential accident.

(3) This rate would apply to the first generation born after the accident. Subsequent generations would experience effects at decreasing rates.

Table 1.5 1968 Annual Mortality Data for the 15 Leading Causes of Death in the United States.

Cause of Death, By Name For 15 Leading Causes Ranked by Number of Deaths	
All Causes	1,930,000
Heart Diseases	745,000
Fatal Neoplasms	324,000
Cerebrovascular Diseases	211,000
Accidents	115,000
Pneumonia	66,400
Certain Causes, Early Infancy	43,800
Diabetes Mellitus	38,400
Arteriosclerosis	33,600
Bronchitis, emphysema, asthma	33,100
Cirrhosis of Liver	29,200
Suicide	21,400
Congenital Anomalies	16,800
Homocide	14,700
Peptic Ulcer	9,500
Kidney Infection	9,400
All Others	217,000

TABLE 1.6 INDIVIDUAL RISK OF EARLY FATALITY BY VARIOUS CAUSES
(U.S. Population Average 1969)

Accident Type	Total Number for 1969	Approximate Individual Risk Early Fatality Probability/yr[a]
Motor Vehicle	55,791	3×10^{-4}
Falls	17,827	9×10^{-5}
Fires and Hot Substance	7,451	4×10^{-5}
Drowning	6,181	3×10^{-5}
Poison	4,516	2×10^{-5}
Firearms	2,309	1×10^{-5}
Machinery (1968)	2,054	1×10^{-5}
Water Transport	1,743	9×10^{-6}
Air Travel	1,778	9×10^{-6}
Falling Objects	1,271	6×10^{-6}
Electrocution	1,148	6×10^{-6}
Railway	884	4×10^{-6}
Lightning	160	5×10^{-7}
Tornadoes	118[b]	4×10^{-7}
Hurricanes	90[c]	4×10^{-7}
All Others	8,695	4×10^{-5}
All Accidents (from Table 6-1)	115,000	6×10^{-4}
Nuclear Accidents (100 reactors)	—	2×10^{-10}[d]

(a) Based on total U.S. population, except as noted.
(b) (1953-1971 avg.)
(c) (1901-1972 avg.)
(d) Based on a population at risk of 15×10^6.

with in preparation of the final report. In particular, Appendix XI specifically discusses comments received from the Environmental Protection Agency (EPA), The American Physical Society Study Group on Reactor Safety (APS), the Atomic Energy Commission (now Nuclear Regulatory Commission) Regulatory Staff, the Advisory Committee on Reactor Safeguards (ACRS), the Union of Concerned Scientists (UCS), and Resources for the Future (RFF). Among the other groups who have commented were the Department of the Interior, the United Kingdom Atomic Energy Authority (UKAEA), and the French Commissariat a l°Energie Atomique (CEA).

A principal defect in Draft WASH-1400, which was identified by the APS (and others), related to the consequence model employed. The final report includes major changes in the consequence model, developed in part as a result of this criticism.

A much smaller number of public comments and criticisms are available on the final version of WASH-1400. The report of the Oversight Hearing before the Subcommittee on Energy and the Environment of the Committee on Interior and Insular Affairs, House of Representatives, June 11, 1976* provides a major source of most such comments, including those by R. C. Erdmann, H. Kendall (UCS), W. Panofsky (APS Study), W. D. Rowe (EPA), F. Von Hippel, J. Yellin, and R. Wilson.

The principal comments of the EPA include the following:

1. On the average, the Reactor Safety Study (RSS) has underestimated latent cancer deaths by a factor of four; this is stated to be a judgmental decision. Also, EPA believes that the RSS results for acute radiation injury are too low; in particular, they feel that the claimed benefits of supportive therapy have not been justified adequately.

2. EPA also believes that their original comment concerning the evaluation of failure-probability for the BWR anticipated transient, without scram, has not been resolved.

*Oversight Hearing on Reactor Safety Study before the Subcommittee on Energy and the Environment of the Committee on Interior and Insular Affairs, House of Representatives, June 11, 1976.

The Department of the Interior continues to express concern about the adequacy of treatment of the impact or core-melt into the ground on ground water or streams, which could lead to a later drinking (or other ingestion) of radioactive water.

Yellin has raised questions concerning the adequacy of treatment of common-mode failures and the potential disparity between actual sites, having large population densities, and the "average site" used in the RSS*.

Yellin, Von Hippel and Panofsky all have expressed reservations concerning the stated uncertainties in WASH-1400. The ACRS has also expressed the opinion that the uncertainties may be larger than stated, and that additional independent work is required to ascertain the quantitative validity of the RSS.

It should not be assumed that the limited number of public comments made thus far on WASH-1400 comprise the entire spectrum of opinion or cover the specific questions that may be raised or re-raised in the future.

In general, the representatives of the Nuclear Regulatory Commission take the position that none of the comments on the final version provide a basis for their changing the values in the final report, and that either these comments are judgmental and cannot be demonstrated to be valid, or that they are encompassed within the stated uncertainty.

Another principal criticism, that the report should not be used for policy making, is endorsed by the representatives of the Reactor Safety Group; however, that the report affects policy cannot be realistically denied.

Summary on LWRs

A detailed study of nuclear reactor risk arising from accidents exists only in WASH-1400, which examined one PWR and one BWR for core-melt probability. The risks for a family of 100 reactors were then estimated by assuming that these two reactors were representatives in accident probability, and averaging the consequence calculations over several reactor sites. If the point estimates of WASH-1400 are accepted, the risk either to an individual or, statistically, to society, is small, and it is small compared to risks arising elsewhere in the generation of electricity.

*ACRS Letter to Chairman Anders, Nuclear Regulatory Commission, April 8, 1975; ACRS Letter to Congressman M. K. Udall, July 14, 1976, ACRS Letter to Congressman M. K. Udall, December 16, 1976

There is considerable controversy over the validity of the quantitative results in WASH-1400, both on probability of core-melt and on magnitude of consequencs. There is a considerable body of opinion that the range of uncertainty is larger than the factor of five estimated by the authors of the report.

However, there is no evidence which demonstrates that the actual results lie outside the uncertainty-band. It is essentially impossible to prove or disprove recurrence frequencies as low as one in 20,000 per year without actuarial data. The concensus is that the probability of core-melt is very unlikely to be more than a factor of ten higher than in WASH-1400, and the consequences are unlikely to be more than a factor of twenty times higher. It must be acknowledged that the probabilities may also be lower than the best estimate of WASH-1400.

The point has been made that there is a wide disparity in the estimated early deaths for a remote site and a relatively highly populated site. This risk is statistical, that is, the risk to the individuals is similar and low, but the total can be considerable if many people are exposed.

Whether the societal risk, imposed by reactors in more populated areas, should not be repeated in future reactors, remains to be judged. The risk may be large relative to a remote site, yet still be acceptably small. The risk may be small compared to many other aspects of society and the hazard may be no larger than that imposed by many large dams or large chemical storage facilites.

In Germany, where a considerable number of LWRs are in operation or under construction, the average site already in use resembles the most populated ones of the U.S. reactor sites. For a site which has a ten-times higher population density, the Germans have imposed a large number of additional safety features whose express purpose is to reduce the probability of core-melt by a factor of ten.

Various options exist in the U.S. These include the following:

(a) Continue the current practice, which has reduced the acceptable population density by about a factor of two from the highest previously used;

(b) Require still more remote siting;

(c) Require new forms of containment against core-melt; including the possibility of molten-core retention, controlled venting of filtered containment gases in the event of an excess pressure buildup, underground siting, measures to reduce the probability of core-melt, or some combination of the above, as proved desirable. Implicit in any such paths would be judgment that the risk from the existing approach was unacceptably large and/or the reduction in societal risk accomplished by the new steps was larger than might be gained by an equivalent expenditure elsewhere in society.

A possible example of the latter question is introduced by the current requirements on routine releases of radioactivity for nuclear power plants. A considerable difference of opinion exists as to the appropriate routine release level. It is possible that money is being spent to reduce releases which are already at a level low enough, and that the same money could be more effective in reducing other societal risks.

Excerpt From Health and Safety Executive (1977)

THE GENERIC SAFETY ISSUES OF PRESSURIZED WATER REACTORS

An account of the study carried out by the Nuclear Installations Inspectorate of the Health and Safety Executive

SUMMARY

This report indicates only the scope and main conclusions of the study carried out by the Nuclear Installations Inspectorate on the Generic Safety Issues of the Pressurized Water Reactor. It is intended to publish a more detailed report of the work later.

Conclusions

1. The Inspectorate consider that there is no fundamental reason for regarding safety as an obstacle to the selection of a Pressurized Water Reactor for commercial electricity generation in Britain.

2. Although there are some safety aspects about which present information and investigations are insufficient to allow final conclusions to be reached, and some areas where more

work would lead to greater confidence, the Inspectorate are satisfied that these issues are not such as to prejudice an immediate decision in principle about the suitability of the Pressurized Water Reactor for commercial use in Britain. Further progress would appropriately form part of the more detailed review of any specific design of reactor put forward for licensing.

3. More detailed conclusions, all of which are subject to General Conclusions 1 and 2, are listed under the following headings:

Introduction

1. The White Paper (Cmnd 5695) published in July 1974, entitled "Nuclear Reactor Systems for Power Generation," stated that the Nuclear Installations Inspectorate (NII) had been asked to bring to a conclusion their studies on the Generic Safety Issues of Light Water Reactors.

2. Two basic concepts of Light Water Reactors are commercially available, the Boiling Water Reactor (BWR) and the Pressurized Water Reactor (PWR). The studies have concentrated on the latter version, that is the Pressurized Water Reactor, since it is this type of reactor in which the Central Electricity Generating Board (CEGB) has declared an interest.

3. This report is a summary of the scope and conclusions of a study which will be reported more fully later in the year by the Health and Safety Executive. The objectives of this study have been to arrive at a view of the PWR concept and in particular to be what health and safety conditions would have to be satisfied before a PWR could be considered to be acceptable in principle for use as a commercial nuclear power plant in the United Kingdom. While a particular plant was selected as a reference, and much of the detailed study based on it, the report is not intended to be, and should not be regarded as, a commentary on a specific design or particular plant.

THE SCOPE AND ORGANIZATION OF THE STUDY

4. The generic safety aspects of any nuclear power plant can be defined as:

   (a) those safety features which can be regarded as inherent in the concept, and

Paragraph No

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(b) features which, while not necessarily being strictly inherent, are likely, in practice, to be common to any alternative options.

5. For the purpose of the study, a substantial volume of information was required and arrangements were made, with the aid of the CEGB and the National Nuclear Corporation (NNC), for the supply of data relating to a reactor, known as Trojan, designed by Westinghouse of America, and located in Oregon. The reports relating to this plant, and backed by a large volume of supporting technical material, have formed a large part of the reference information used in this review.

6. By agreement with the Kraftwerk Union Company in West Germany, discussions have also taken place between them and NII during which information regarding their design of PWR has been outlined. This information, although on a smaller scale than that supplied by Westinghouse, covered the same generic topics and was sufficient to enable an appreciation to be made of the important safety differences between the two designs.

7. Because of the international interest in the PWR, special attention has been paid to developments overseas. Discussions have been held with groups having a similar role to the Nuclear Installations Inspectorate in Germany, France and USA. In the conduct of the study, the Inspectorate have been assisted by consultants selected, not only for their professional specialism, but also, as far as possible, as individuals who could reasonably be expected to adopt an independent view. In addition, a supporting program of extramural work has been conducted, largely involving theoretical analytical studies but also including some experimental work.

Methodology of the study

8. The basic method adopted in the study was to take the Westinghouse safety documents and to subject them to a rigorous professional review. This resulted in a number of detailed technical questions which were addressed to Westinghouse and which gave rise to additional technical information. Particular attention was paid to any restrictions or assumptions implicit in the Westinghouse approach, and the Inspectorate needed to satisfy themselves that any such restrictions or assumptions did not invalidate, or significantly limit the generality of, the conclusions reached by Westinghouse. As is the case in any detailed safety assessment of a reactor or other plant,

there is a substantial element of professional judgment in deciding, first, whether the starting point for calculations and appraisals adequately represents all the possible situations and, secondly, whether the method of analysis represents an adequate interpretation of the physical events. There is no way of eliminating this dependence on professional judgment, and thus of making absolute statements concerning the safety of complex systems.

9. The associated work has involved the Inspectorate in a total effort of about 12 man-years, and has involved a further expenditure of £100,000 by HSE on consultancy and associated work. An additional and substantial contribution has been made by the staff of Westinghouse and NNC in providing information required by the Inspectorate.

<u>The intended system of reporting the details of this study</u>

10. This report is intended to do no more than indicate the scope of the study and its principal conclusions. A more detailed report is being prepared which the Health and Safety Commission will refer to its Advisory Committee on the Safety of Nuclear Installations. The Committee will be asked to comment, in particular, on any areas which have been identified as requiring further study.

THE BASIS OF JUDMENT

11. No human activity is entirely free from features that are potentially detrimental to health or involve risks to life. The basic policy of the Health and Safety Commission is to eliminate these risks so far as is reasonably practicable. In some cases, where risks would otherwise be high, absolute duties and absolute requirements are imposed. These may take the form of quantitative limits or specific design requirements but they do not remove the further statutory duty to achieve even safer conditions whenever this is reasonably practicable. These duties and requirements do not eliminate risk completely.

12. The criteria used by the Inspectorate as a basis for judging other reactor systems have been applied in this case.

These criteria have been developed from more than 20 years of experience and practice in nuclear safety in the United Kingdom. Account has also been taken of the criteria applied by the Nuclear Regulatory Commission (NRC) which is the body responsible for nuclear safety in the USA. Certain aspects of these criteria set out in the U.S. Code of Federal Regulations (10 CFR 50) have been used to supplement the NII criteria where this has been judged useful and appropriate.

13. Much of the work in this study is concerned with particular fault sequences. Each of the fault sequences considered in the assessment is representative of a range of possibilities and in each case there is a very small but finite possibility that a release of radioactive material to the environment might occur as a consequence. So far as individuals and the population at large are concerned it is the total probability that a given consequence will occur that is of concern. To arrive at an estimate of this risk it is necessary to combine the magnitude of the expected releases and the probabilities of those releases due to all identifiable causes. The result is a relationship between expected consequences and the overall probability of those consequences due to all causes. In common with most everyday experience, it is found that, as the consequences become more severe, the probability of their occurrence becomes less.

14. In the absence of any practical example of a reactor accident having severe consequences, such as injury or death, beyond the confines of the reactor building, risk assessments have to be based on a synthesis of experience of separate fault conditions. The results of one such study are discussed in a later section of the report, Risk Assessment. (Para. 53)

THE PRESSURIZED WATER REACTOR

15. A pressurized water reactor consists in essence of a nuclear core consisting of an array of cylindrical fuel pins in which enriched uranium oxide fuel is contained in Zirconium alloy tubes. The core is mounted inside a water-filled reactor vessel and external coolant pumps force the water upwards through the core, out to steam generators and back to the core, thus completing the primary circuit. The pressure in this circuit is kept sufficiently high to prevent the water from boiling. In the steam generators, the heat from the primary coolant water is transferred to a secondary circuit and the steam passed to the turbines for driving the electrical generators. The cooling systems have to operate even when the reactor is shut down in order to remove the heat released by the

decaying fission produced. The whole of the primary circuit, including the steam generators, is enclosed in a containment vessel. The arrangement is shown diagramatically in Figure I.

16. The nuclear reaction is controlled by the insertion of neutron-absorbing rods into the core or, on a longer timescale, by the addition of the neutron-absorbing material, boron, to the coolant.

THE PRINCIPAL HEALTH AND SAFETY ISSUES

17. The principal health and safety issues concerned with any reactor type can conveniently be grouped into three classes - the radiation doses to workers in normal operation; the generation and management of radioactive wastes in normal operation and the likelihood and consequences of fault conditions. If, in normal operation, there is direct irradiation of members of the public from within the plant rather than as a result of the management radioactive wastes, it is convenient to deal with this at the same time as irradiation of workers.

Radiation doses in normal operations

18. For most operations on any design of nuclear reactor, the problems of controlling the radiation exposure of workers in accordance with international recommendations and with the requirements of the European Community and UK regulations are fairly easily dealt with by appropriate design measures and by good standards of operational control. There are, however, a few operations, notably maintenance work, for which it is difficult to predict radiation doses at the design stage, and where operational aspects become more important. Experience suggests that water-cooled reactors pose rather more problems of this kind than do gas-cooled reactors, but that the problems are still tractable. On the basis of this experience, only limited detailed attention has been given to radiation doses in routine conditions in this study.

Radioactive wastes and their management

19. While not in itself a waste product, the irradiated nuclear fuel from a reactor contains almost all the radioactive

waste material produced. The amount of radioactive waste in irradiated fuel is almost independent of the particular type of thermal reactor. Essentially all of the waste is transferred to irradiated fuel storage facilities pending reprocessing to recover uranium and plutonium, or disposal if such recovery is not adopted. There are no particular issues affecting the pressurized water reactors as such, and this type of waste is not discussed further in the study.

20. Reactor operations and maintenance work also give rise to the production of wastes of a variety of characteristics, usually of much lower activity. Some of these consist of gases or airborne particulates, and these are released to atmosphere after suitable treatment. Liquid wastes also arise and, after treatment, are also discharged to the environment. Solid wastes, including the radioactive material extracted from liquid and gaseous streams, comprise a wide variety of materials and, although they are not specific to any particular reactor type, they do vary from one type to another. The problems of managing these wastes from a PWR can be dealt with by appropriate design measure and by good standards of operational control. They are given only limited consideration in the study.

<u>Fault conditions</u>

21. In any well-designed reactor system, most fault conditions fall into one of two classes - either the reactor continues to operate, relying, when necessary, on alternative or standby components, or, if this would result in too big a reduction in the margin of safety, the reactor automatically shuts down until the fault can be rectified. A futher, but numerically smaller, group of faults comprises those which are not automatically detected and which can be discovered only during routine inspection and maintenance procedures. The most important of these latter faults in the pressurized water reactors are those concerning defects in the pressure vessel and the associated primary coolant circuit. Finally, there are some very improbable combinations of fault conditions which are dealt with automatically by the reactor system, but not without some damage to the reactor and possibly some escape of radioactive material into the reactor containment and from there, in greater or lesser quantities, into the environment. Public attention tends to be concentrated on accidents which might cause radiation exposure of the public and, while it is one of the aims of

design work and safety analysis to ensure that such accidents are very rare, much of this work is concerned with ensuring that the less serious, and less rare, fault conditions are successfully detected and dealt with before they can give rise to danger.

22. Some of the possible fault conditions and the methods of dealing with them are specific to each design reactor, and this is particularly true of the control systems. Nevertheless, it has proved possible to review a wide range of fault conditions that are likely to be associated with most, if not all, designs of PWR. In particular, a great deal of work has been carried out and published on the possible faults in the pressure vessel and the primary coolant circuit.

23. This review of the PWR has therefore taken the following form:

i A search for potential faults.

ii A review of the processes following a wide variety of the most important faults, known as limiting faults.

iii An evaluation of the various protective systems employed in the design to prevent or intercept and control faults.

iv An assessment of the chance of various discrete fault sequences occurring and the likely consequences of each such sequence.

v A judgment as to the confidence that can be attached to the various aspects of the analyses with respect to both the physical processes and the chance of occurrences.

THE PRINCIPAL GENERIC ISSUES

The range of fault conditions

24. The principal objectives of this part of the study were:

i to judge the adequacy of the scope of the studies conducted by the designers:

and

ii to judge the adequacy of various protective features provided in a typical PWR.

25. On the first of these points, the Inspectorate have concluded that the scope of the studies already conducted by the designers is sufficiently wide for the present purposes, but that it is not completely comprehensive. Additional sensitivity studies to demonstrate that the faults selected for study were limiting cases would be necessary before a specific design could be licensed. Some additional work on the analytical models used for fault analysis would also be desirable.

26. Certain specific classes of faults were selected for more intensive study partly because these were judged to contribute significantly to the overall risk. Four classes were identified, although there is some overlap between them. The first class is the loss of coolant accidents in which the liquid cooling water is lost in tne immediate vicinity of the fuel elements. The resulting steam is a much less effective heat transfer medium than water, and this type of fault can lead to severe damage to the fuel elements. One cause is a sudden failure of part of the primary coolant circuit.

27. The second class comprises fault conditions which should cause the reactor to shut down automatically but which fail to do so because of a further fault.

28. The third class of faults involves a failure in the secondary cooling circuit which might, in turn, induce damaging temperature and pressure changes in the primary circuit. This class of fault could, therefore, be the cause of a loss of coolant accident. The fourth class, which has common features with the third, is the failure of a plant component in such a way as to interrupt the path for heat removal from the reactor.

29. In the following subsections an outline is given of the scope of the studies and the principal conclusions, under a series of headings, starting with a discussion of the loss of coolant accident, and then dealing with the various parts of the reactor system either because they might be affected by an accident or because they might contribute to the way in which an accident develops.

<u>The loss of coolant accident</u>

30. Although other possibilities exist, the term "loss of coolant accident" has come to mean accidents involving a breach in the primary coolant circuit. Thus, in addition to the implications for cooling the fuel, one stage of the multi-layer containment system of the reactor has inevitably been breached. It is therefore particularly important to re-establish emergency cooling before the core has been damaged to the point where it can no longer be adequately cooled. This is achieved by the injection of the emergency cooling water. The detailed course of events is extremely complex and will vary widely from accident to accident. It is therefore not practicable to explore, either experimentally or theoretically, all possible atlernatives, and the normal aproach is to identifying limiting cases. Studies are then carried out by computer calculation, validated as far as possible against experimental simulations and fundamental scientific studies of the behavior of fluids. In reviewing such work it is necessary, first, to assess the choice of conditions to be studied and, secondly, to assess the likelihood that the theoretical assessment accurately models the situations that would occur in practice.

31. The Inspectorate have reached a number of conclusions concerning the loss of coolant accident: these should be read in the light of General Conclusion 2.

    1. The evaluation models used correspond to adequately extreme cases, although it would be desirable to include consideration of a modest breach in the pressure vessel below core level.

    2. The models themselves are judged adequately conservative, although enhanced confidence in them is desirable in some respects, and can be achieved in the future by broadening the range of some of the assumptions, and by extending the scope of the analyses in a somewhat more systematic way. Correlation of the results with experimental simulation of plant conditions is an important part of validation and could provide further information on scaling factors. Attention should also be paid to the effects of a fault involving several steam generator tubes.

3. Mechanical stresses due to the discharge of coolant are adequately guarded against, and the effects of shock on the reactor system are judged to be within acceptable limits. However, on this latter point some additional tehoretical studies are indicated.

4. There are some questions as to whether physical swelling of the fuel elements during a loss of coolant accident might be sufficient to prevent emergency cooling of parts of the core. Further experimental work and analysis in this area is called for. (See also paras. 42 and 43)

5. Some residual uncertainties in the appraisal of the effectiveness of emergency coolant injection might be reduced by changes in design of the reactor - e.g, by providing for the injection of emergency coolant at the top and bottom of the core. Some additional uncertainties might be eliminated by a small extension in the scope of the analyses carried out.

<u>Other fault conditions</u>

32. All power reactor systems will experience fairly frequent perturbations. These include, for example, the accidental isolation of the generating plant from the supply system, or failures in electronic control systems. Most of these will be accommodated without the reactor having to shut down, but some faults are sufficiently serious that the design response is an automatic shutdown. The combined consequences of an initial fault followed by a failure to shut down as designed might then be serious. In general, at least two independent and diverse protective systems are required to prevent this situation from developing.

33. In the PWR, although there are multiple systems for initiating shutdown, the shutdown system itself involves the use of neutron-absorbing rods. These, while multiple, are not diverse. There is also a method of injecting neutron-absorber boron into the coolant system, but this is a slow acting system intended for long-term shutdown.

34. The study has included reviews of the consequences of a typical range of faults in which the design intent of automatic shutdown has not been achieved. The consequences of such faults may include over-heating of the fuel and over-pressurizing of the primary circuit. All these

faults are essentially self-limiting because the reactor has a negative reactivity coefficient with respect to power and temperature. In addition, if vapor voids are formed in a reactor core, there is also a negative effect on the reactivity of the system, so that faults giving rise to vapor voids also tend to be self-limiting. These features justify the use of a non-diverse system of rapid shutdown devices.

35. As in the case of loss of coolant accidents, the total range of possibilities is very large, and representative extreme cases have to be considered and analyzed in detail.

36. A further class of faults considered is typified by the possible break of a steam line between the steam generator and the turbine. Such an event could transiently reduce both the temperature and the pressure of the primary coolant. The temperature reduction may make the primary circuit more sensitive to existing defects and may give rise to thermal stresses. The pressure reduction may operate the emergency core system causing repressurization of the primary circuit. Of these factors, the last seems to be the most important and could probably be avoided by a redesign of the emergency logic of the reactor so that the emergency cooling was not initiated in the absence of the primary circuit failure.

37. The Inspectorate have reached the following conclusions on these aspects of the PWR: (see also General Conclusion 2.

   1. The range of faults anlayzed and the details of these analyses are sufficient for the present purposes, but better coverage, including the use of sensitivity analyses, and better validation of theoretical models would be necessary to provide adequate confidence for the acceptance of a specific plant.

   2. A review of faults which fail to shut down the reactor suggests that, provided steam generator tube integrity is maintained, such events are unlikely to give rise to a hazard. However, increased confidence is required that the analytical projections are consevative. This objective might be achieved by certain revisions, either to the design or to the operating mode of the plant.

3. Faults involving pressurization of the primary circuit, particularly at lower than normal operating temperatures, add to the risk of a failure of the primary coolant circuit, but such faults might well be eliminated by reconsideration of the operating mode or logic of the protective systems.

<u>The integrity of the primary coolant circuit</u>

38. The primary coolant circuit consists essentially of the pressure vessel housing, the main reactor core, the steam generator tubes, and the connecting pipework. A failure in the steam generators or pipework may produce a loss of coolant accident, but one which is within the design capabilities of the emergency core cooling system. This will also be true of some kinds of failures of the pressure vessel, but there may be pressure vessel failures of a kind so serious that it would not be possible to provide adequate cooling of the core. The likelihood of failures of this last kind has to be made exceedingly small.

39. To achieve the necessary integrity of the whole circuit, the standard of design, manufacture and operation of all parts has to be very high. There is relevant experience in the nuclear industry and more broadly in industry generally, and this leads to substantial confidence in the integrity of the circuit. Nevertheless, no complex welded steel structure can be made free of flaws, and pre-operational inspection and testing can detect flaws only above some limiting size. In use, such pre-existing flaws may grow in size, and if they continue to be undetected, could lead to a failure. Post-operational inspection is therefore needed, with the aim of detecting with very high reliability any flaws before they have grown to the size where they might initiate a sudden failure.

40. The Inspectorate have come to a number of conclusions: see also General Conclusion 2.

    1. It will be possible to design and construct the primary circuit pipework so that it has a suitably high degree of resistance to spontaneous failure. In many cases it is likely to leak before it fails, thus allowing remedial action.

2. The pressure vessel surrounding each of the steam generators and forming part of the secondary circuit should be dealt with by measures comparable to those used to ensure the integrity of the reactor pressure vessel.

3. The present analyses of certain faults is based on the failure under normal stresses of a single steam generator tube. It is desirable to extend these analyses to include the failure of several tubes, or alternatively to strengthen the evidence put forward to support the argument that only one can be expected to fail in practice.

4. It will be possible to design, construct and operate a PWR pressure vessel in a manner which makes the likelihood of sudden failure acceptably small. Design changes, such as the possibility of using forged elements, might offer improved confidence.

5. Conclusion 4 can be supported only if there is a searching and rigorous program of nondestructive examination both before operation and in service. A continuing program of work is also necessary to evaluate and improve the sensitivity of examination techniques.

6. Additional studies are needed to improve the data base for the theoretical fracture mechanics used in the evaluation of the safety of the reactor pressure vessel. This work, and the additional studies on nondestructive examination techniques, need not influence any decision regarding the adoption of the pressurized water reator system, but they would add to the confidence in the safety of the system and they would become important components in the detailed program of assessing a specific design for licensing purposes.

7. The arrangements necessary to ensure a suitable degree of independence of approach to the implementation of quality assurance is to be reviewed. This matter should be considered in a broader context than that of the reactor pressure vessel alone.

Fuel element behavior

41. There is extensive experience of the behavior of PWR fuel elements in normal operational conditions, and, as a result of this experience, a high standard of fuel reliability has now been achieved. Any failures in normal operation have implications for the exposure of workers on the reactor site and the generation of radioactive wastes. Because of the available experience, the necessary plans for dealing with damaged fuel can be made at the design stage.

42. Examination of the likely transient conditions shows that for most of them, even if the reactor fails to shut down automatically the fuel in the interval between the loss of coolant and the reestablishment of emergency cooling is a critical part of the reactor accident analysis. The study has given rise to one area of doubt. The combination of internal pressure in the fuel elements and increased temperatures results in some swelling of the fuel cladding. This swelling could interfere with the reestablishment of cooling under emergency conditions. Until recently, the experimental evidence has indicated that such swelling would be localized and not sufficiently serious to cause problems of this kind, but some limited experimental work in this country has now suggested that there may be conditions in which swelling might take place along an extended length of a fuel element and be of sufficient magnitude to prevent the reestablishment of cooling. Further experimental work and a reexamination of the analysis of transients are needed to estaolish whether the experimental conditions are representative of the conditions actually to be expected in the reactor.

43. The Inspectorate have reached the following conclusions: (see also General Conclusion 2.)

    1. The experience with PWR fuels shows that the fuel element defects in normal operations will not be a serious problem.

    2. In transient conditions short of loss of coolant, limiting operating conditions can be established to prevent serious fuel damage.

    3. In the event of a loss of coolant accident, the emergency cooling arrangements should be able to

re-establish cooling of the core. However, further experimental and analytical work is required to establish whether recent UK tests covering fuel cladding deformation are representative of in-reactor conditions during a LOCA. If it is established that the tests are representative, it is reasonable to expect that, following a more detailed examination of the technical issues, either the effect suggested by the UK experiments will be shown to be tolerable, or, by appropriate adjustment to design or operating conditions, the problem could be avoided.

<u>The reactor protection system</u>

44. Many of the details of a reactor protection system are specific to the design under review, and there is little to be said in a generic sense. In general, faults must be detected with a high reliability, and where they could lead to serious consequences there is need for a diversity in the methods of detection and in the initiation and conduct of the subsequent automatic action. Particular attention has to be paid to the avoidance of common mode failures associated, for example, with the use of the same type of component in systems which are otherwise apparently diverse.

45. The Inspectorate have identified no fundamental protection issues which would prejudice the adoption of the PWR, although there are some specific non-generic issues which would need further discussion if a PWR were put forward for licensing.

<u>Containment</u>

46. Containment should be regarded as the last of a series of defenses in depth against the loss of radioactive material from the fuel elements and thence from the primary coolant circuit into the reactor building and onwards to the environment. The containment should not be regarded only as a barrier of this kind, however. The maintenance of some pressure inside the containment following a loss of coolant accident improves the performance of the emergency core cooling system, but the containment may also restrict the space available for plant layout, and thus be a handicap in radiological protection during normal operations or maintenance. A containment structure also provides some protection against external events, such as crashing aircraft or the impact of missiles generated by the failure

of rotating machinery or explosions in nearby plants. Finally, even if the containment is not completely successful in retaining radioactivity, it provides a valuable delay during which emergency arrangements can be put into effect.

Programs of research and development on light water reactors

47. On a world-wide basis there exists a substantial volume of research and development work concerned with pressurized water reactors, and a substantial part of this work is aimed at safety, and particularly at improving the understanding of the complex processes, and thus of giving enhanced confidence in safety judgment. Some of the work is funded by the commercial design companies, and the results are then not necessarily widely available. However, other programs are established by government organizations and the resulting published body of knowledge is substantial.

48. The Inspectorate consider it desirable for there to be a continuing R&D program in PWR safety in the United Kingdom if there is to be any significant dependence on PWR as a commercial source of electricity in this country. Any UK R&D programs should be closely linked to other countries, particularly to the USA and Europe.

MAJOR VARIANTS FOR PWR DESIGNS

49. Although detailed differences exist between PWR designs marketed by various suppliers at variours times, these differences are not fundamental and would not affect any of the generic discussions of this study. However, one substantial variant has been considered for possible installation in a highly developed industrial and residential area in the Federal Republic in Germany. The most notable aspect of this design was the use of a secondary concrete structure closely surrounding the steel primary coolant circuit. The aim of the design was to take some of the stress off the steel structure and to reduce the severity of a loss of coolant accident should the primary circuit, nevertheless, fail.

50. The Inspectorate have not had access to detailed safety assessments for this proposed plant, which was never built, and the proposal was not sufficiently developed to call for a formal application for licensing by the German authorities.

51. The Inspectorate have concluded that there may be design changes which would improve the safety of the PWR, but that these aspects would require further study and probably further engineering development work. They should be taken into account in any consideration of a specific PWR design for publication in the United Kindgom, but do not significantly affect the review of generic issues.

RISK ASSESSMENT

52. Light water reactors, of which PWRs form a substantial fraction of the world's installed capacity, have been the subject of a very detailed analysis, the results of which have been published by Rasmussen*. The importance of this study, the most comprehensive of its kind, is that it produced two kinds of information; first, it enabled hitherto unidentified weak points in the designs to be revealed, and secondly, it has enabled the risk due to thermal power reactors to be seen in the context of the risks to health from other, non-nuclear human activities. The general conclusion reached by Rasmussen was that the risk due to nuclear power from water cooled reactors was significantly less than that due to other man-made hazards and to natural causes in the United States. These results, being based on conditions in the USA where population density is lower on average than in the UK, and the dangers of natural events higher, cannot be applied directly to this country. A reasonable adjustment still indicates nuclear hazards to be below other man-made hazards.

53. More specifically, the study led to the conclusion that the probability of a light water reactor giving rise to an accident large enough to cause a substantial number of deaths (ten or hundreds) in the surrounding population was no more than about 10^{-7} per reactor year. When account is taken of the inevitable uncertainties in this type of appraisal, this figure is probably not significantly different from that to be expected from other commercial reactor systems. However, also because of the inevitable uncertainties, such comparisons should not be regarded as definitive and any

*Reactor Safety Study. An Assessment of Accident Risk in U.S. Commercial Nuclear Power Plants. WASH-1400 (NUREG 75/014) available from the National Technical Information Service, Springfield, Virginia, USA consideration of licensing will depend far more on a detailed appraisal of the quality of the engineering input to the design manufacture, operation and maintenance of the reactor system.

GENERAL CONCLUSIONS

1. The Inspectorate consider that there is no fundamental reason for regarding safety as an obstacle to the selection of a Pressurized Water Reactor for commercial electricity generation in Britain.

2. Although there are some safety aspects about which present information and investigations are insufficient to allow final conclusions to be reached, and some areas where more work would lead to greater confidence, the Inspectorate are satisfied that these issues are not such as to prejudice an immediate decision in principle about the suitability of the Pressurized Water Reactor for commercial use in Britain. Further progress would appropriately form part of the more detailed review of any specific design of reactor put forward for licensing.

3. More detailed conclusions, all of which are subject to General Conclusions 1 and 2, are listed under the following headings:

   Loss of coolant accident (para 32)
   Other fault conditions (para 38)
   The integrity of the primary coolant circuit (para 41)
   Fuel element behavior (para 44)
   The reactor protection system (para 46)
   Program of research and development (para. 49) and
   Major variants of PWR designs (para. 52)

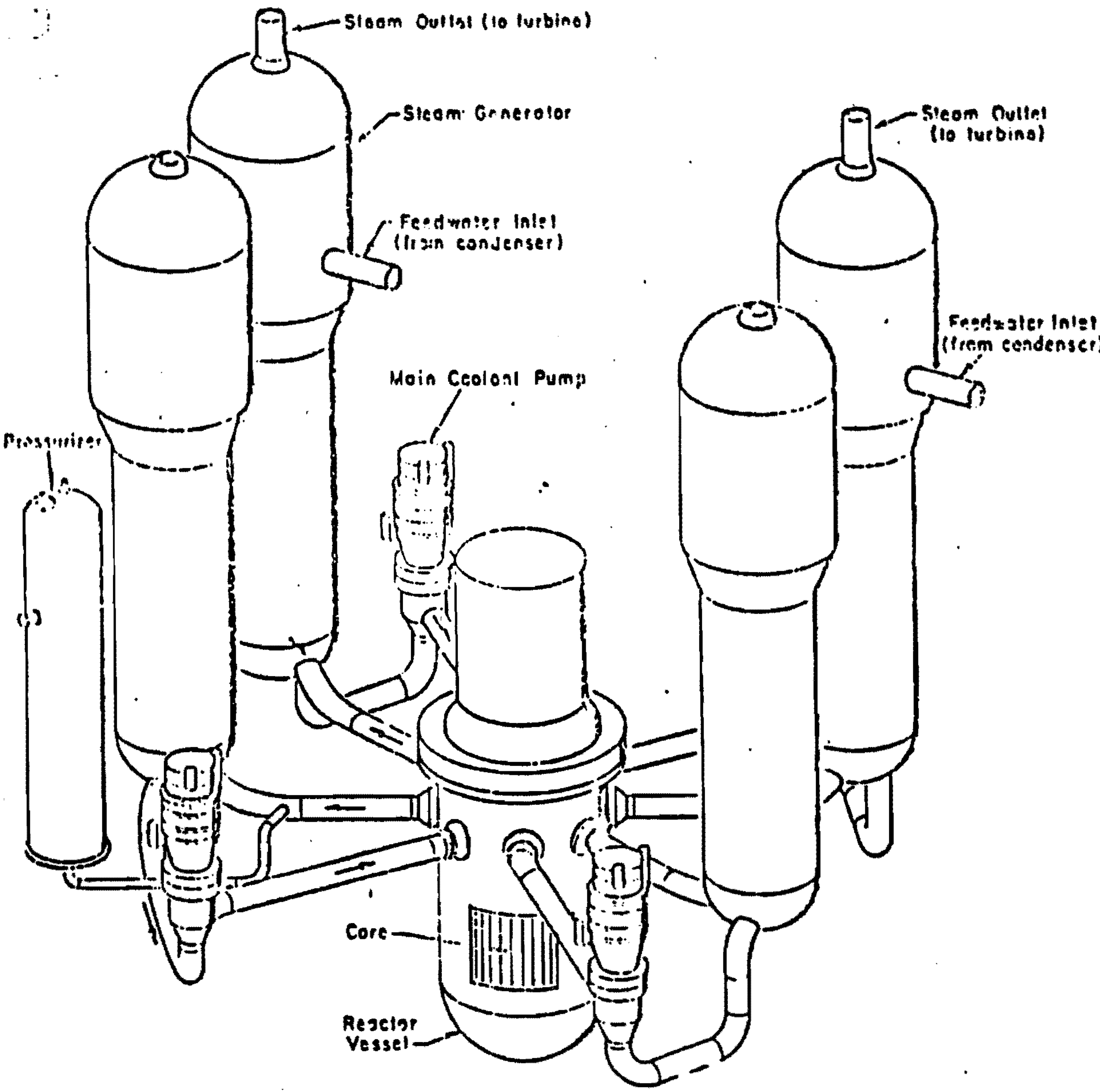

Figure 1
Schematic Arrangement of the PWR Components within the Containment.

Published in Journal of the Institution of Nuclear Engineers
April 1975, Vol. 16

ACCIDENT PROBABILITY CRITERIA

by

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INTRODUCTION

I shall be talking about non-nuclear as well as nuclear accidents and would like to introduce the subject in quite a general way. There are those who express the good intention of operating their plant in such a way as to create no hazard whatsoever, or others say that "we must do all that is necessary to make the plant safe". Obviously both of these statements are related to the degree of perception of the speaker, the amount of time devoted to accident studies, and the degree of credibility that is embodied in the assumptions. These statements follow a 'maximum credible accident' philosophy.

There are those who say that the basis of criteria should be on individual risk and we can recall a suggestion by Adams that the criterion might be around 10^{-5} per person per year, whereas recently it seems that the individual risk level might be as low as 10^{-7}. There have been many papers quoting risks from normal and abnormal occurrences. some based on straight probabilities per year and others on probabilities per unit of exposure, i.e., in time,or the like. Table I is extracted from Rasmussen's report (WASH-1400) and shows risks ranging from 10^{-3} to 10^{-4} down to 10^{-7}; the second is included in the report of Vinck 1973 (EUR 5001) and ranges from 10^{-2} to 10^{-7} (Table II).

There are those who say that nuclear risks are rather special - are they, in fact? At one time it was thought that the delayed action made them rather special, but we are well aware of delayed action in other industrial risks such as asbestosis, exposure to vinyl-chloride, or risks to which the public were thought to be exposed through lead, mercury, DDT and cyclamates. No doubt people who have spent their lives considering nuclear risks regard them as rather special, but does this justify more money per lives saved than other risks to which the community are exposed?

HIGH RISK SITUATIONS

I have been mainly interested in, and will be talking about, high-risk situations or, to be more precise, events which have a low probability of occurrence but which might well have serious consequences. I would like first to discuss the non-nuclear risks of this type and again made use of Rasmussen's work where he

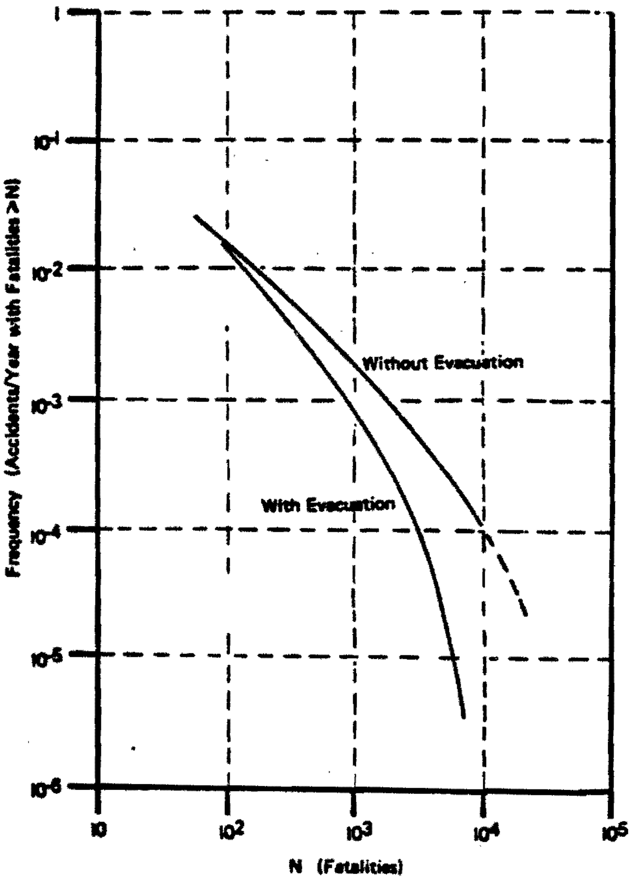

Figure 1. Frequency of Chlorine Accidents with Fatalities greater than N

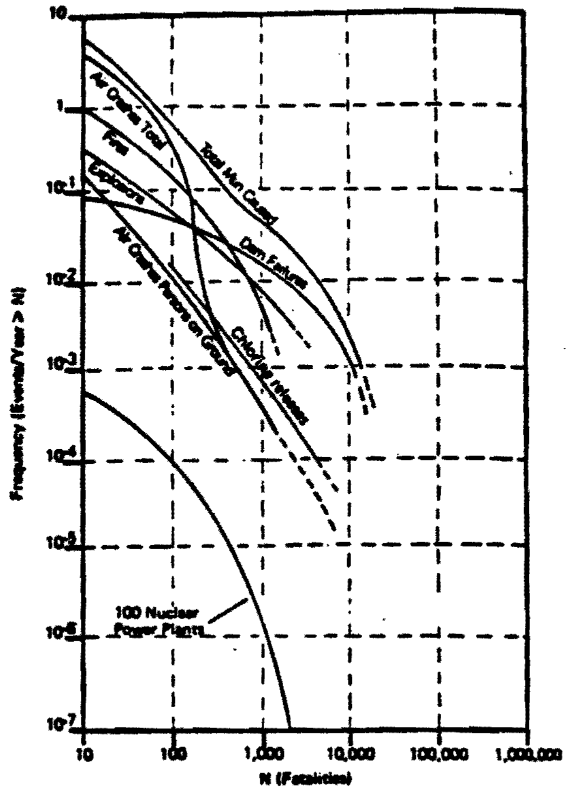

*Figure 2. Frequency of Man-caused Events with Fatalities greater than N**

TABLE I Individual Risk of Acute Fatality by Various Causes (U.S. Population Average 1969)

Accident Type	Total Number for 1969	Approximate Individual Risk Acute Fatality Probability/yr[1]
Motor Vehicle	55, 791	3×10^{-4}
Falls	17, 827	9×10^{-5}
Fires and Hot Substance	7, 451	4×10^{-5}
Drowning	6, 181	3×10^{-5}
Poison	4, 516	2×10^{-5}
Firearms	2, 309	1×10^{-5}
Machinery (1968)	2, 054	1×10^{-5}
Water Transport	1, 743	9×10^{-6}
Air Travel	1, 778	9×10^{-6}
Falling Objects	1, 271	6×10^{-6}
Electrocution	1, 148	6×10^{-6}
Railway	884	4×10^{-6}
Lightning	160	5×10^{-7}
Tornadoes	91[1]	4×10^{-7}
Hurricanes	93[2]	4×10^{-7}
All Others	8, 695	4×10^{-5}

[1] Based on total U.S. population, except as noted.
[2] (1953-1971 avg.)
[3] (1901-1972 avg.)

TABLE II Probabilities of individual fatal injury (casualty) through conventional activities and causes and through the effects of radiation

Type of risk	Individual probability of fatal injury per year of exposure (orders of magnitude)[1]	Remarks
Conventional (casualties only)		
— all diseases	10^{-2}	10^{-2} for light heavy *and* fatal injury
— motor accidents (automobiles)	10^{-4}	
— total mortality risk	10^{-3} (men) 10^{-4} (women)	
— accidents of all types	5×10^{-4}	
— smoking	5×10^{-4}	
— traffic accidents (in general)	2.5×10^{-4}	
— suicide	2×10^{-4}	
— falls	10^{-4}	
— air pollution	10^{-4}	
— industrial accidents	10^{-4} (all ages) 10^{-5} (age 20)	
— drowning	3×10^{-5}	
— firearms	2×10^{-5}	
— electricity	2×10^{-5}	
— leukemia (natural causes)	10^{-5}	
— poisoning	10^{-5}	
— coal and oil-fired power stations (pollution)	4×10^{-6}	
— cancer of thyroid (natural causes)	10^{-6}	
— natural disasters	2×10^{-6}	
— lightning	5×10^{-7}	

[1] Summary of data from various sources; with slight variations according to the country.

TABLE III Risk of Natural or Man-Made Disasters

Tornado	10^{-2}	> 100
	$10^{-3}/10^{-4}$	> 1,000
Earthquake	10^{-2}	≈10,000
Aircraft fatalities on ground	10^{-2}	100
	10^{-4}	> 1,000
	$\approx 10^{-6}$	10,000
Explosions	approx 10^{-1}	> 100
	$\approx 10^{-2}$	≈ 1,000
Dams	10^{-1}	10
	10^{-2}	1,000
	10^{-3}	10,000
Fire	10^{-1}	100
	10^{-2}	1,000

considers the risk that 10, 100, or 1,000 people may be killed as a result of natural or man-made disasters. We could reduce some of these to individual risks and, in many cases, the number would be below 10^{-6}, for example the risk of a person being killed through an aircraft falling upon them in the US is of the order of 10^{-7} per person per year in that the average number so killed per year during the last few years has been twenty people (Table III).

No doubt many of the risks arising in advanced societies have been reduced to the present level (as assumed to exist by extrapolation) by increasing the standard of Building Codes, as in houses or dams; providing better fire resistance; better warning and fire fighting; warning of tornadoes etc. It is my belief that action to decrease risk usually follows some severe event, and the accident which leads to action will depend upon the current background of risks endured by each society. There is an obvious difference in the consequential damage from earthquakes as between, say, Afghanistan and Japan. In Japan there has been much greater investment in the provision of buildings which will withstand earthquakes. There are some activities, as in building, transport, or even sport, in which poor practices, or practices carrying some risk continue until a spectacular event occurs, such as the collapse of the Rowan flats, recent problems with high aluminua concrete beams, the phosphate ship explosion in Mexico, the collapse of crash barriers on football grounds. When minor catastrophes occur, new standards are introduced, but I do not believe that these changes occur as a result of assessing the individual risk of harm - in most cases this risk would fall well below 10^{-6} per person per year - but as a result of societal pressures indicating that these various minor disasters are, at that point in time, intolorable.

Let me take another example - the transport of chlorine through major towns; this is only as an example of toxic material, not as an indication that chlorine is particularly worse than others which are transported. Rasmussen considers the transport of 90 ton tankers on normal routes in the US. He extrapolates past experience to deduce the likelihood of collision, derailment etc. Some events will be very damaging; others less so. Some will occur in open country, some in towns. There will be other variables, even the change of population density within a district depending on the time of day. Note, for example, on 2 June 1970 in Illinois, 10 tank cars of 33,000 gallon capacity were derailed; there was a vapour cloud explosion starting fires and subsequent explosions that destroyed the business section of the town, but there were no fatalities.

Additionally, the harm to people will depend upon wind direction, atmospheric stability etc. He concludes that in the US the likelihood that people might be killed as a result of transporting chlorine is as shown on Fig. 1.

If the more probable event (for transport of chemicals) were to occur, it would lead to casualties in the range 0 to 10 and I guess that there would be little change introduced into current practices. If the event led to casualties in the range 100 to 1,000 then I would expect some consideration of transport re-routing or improved design of tank cars or some other risk reducing factor. The difficulty lies in changing from present status where industry is currently dependent on the continuation of current practices as compared with the introduction of some new enterprise. This is also true in the introduction of reactors where it is probable that if a severe accident occurred early in their history, then it is possible that no more of that type might be built or there would be significant change in construction or siting. If the serious accident occurs arount the year 2,000 when there will be over 1,000 nuclear plants operating and many countries very strongly dependent upon them for their power supply, then what can be done other than to continue operation with some restrictions?

Both in respect of nuclear and non-nuclear risks, it has often been said that risk will be maintained at a low level because of the amount of insurance charges or the cost of replacing the plant or dealing with the damage and these act as a deterrent ensuring that safe plant and safe practices will evolve. There is no evidence that this is true. There have been many expensive accidents in industry and many of them have few casualties. Consider the record of vapour explosions which are reviewed by various writers. There have been 108 known cases of vapour explosion over forth years:

Up to 1950	Damage averaged less than $1/2M per year with the exception of the accident in Cleveland when an explosion of LNG in 1944 led to 136 dead, 77 missing, and a cost of several million dollars.
1950-64	Damage averaged less than $1M per year and since then there has been an increasing number of events and it would appear in recent times that the cost is running as high as several tens of millions of dollars per year.
1967	Failure of a 10 inch valve led to the release of 4,000 gallons of isobutylene; the vapour cloud explosion killed 7, with damage estimated at $35M.
1968	Accident at Perris cost $46M
1959	Freight car accident LPG killed 23
1962	7,000 gallon truck led to a vapour explosion. Killed 10.
1966	Propane explosion killed 17. Whereas, as reported earlier, 10 tank cars in Illinois caused major destruction but did not kill anyone.

My conclusion from these examples is that where an activity such as refining petroleum is a benefit to society, then we will continue to accept a fairly high frequency risk of events which are extremely damaging but seldom kill many people, and I presume that the damage is small in relation to the total value of the industry. I do not believe, then, that the cost of insurance is a general deterrent in situations which are thought to have a low frequency even if, in fact, frequency on a world-wide basis is still fairly high.

NUCLEAR PLANT

Turning now to nuclear plant, many of you will know of my early suggestions that there should be a risk criterion relating fission product releast to frequency of release on an inverse log/log basis. In 1967 we hesitated to suggest that there might be some risk of a nuclear accident releasing large quantities of fission products and we were content to talk about the release of I-131 and associated volatiles ranging from 10 to 10^{-6} curies. In later papers a number of people assessed the likely result of releasing 10% or more of the volatile fission products from a 1,000 MW(°) reactor. My assessment was that the consequences on a populated UK site could be described as:

- a 50% chance of no casualties with wind blowing to sea
- a 30% chance of between 100/1,000 cases of thyroid cancer
- a 20% chance of between 1,000/10,000 cases developing over the 10-20 years thereafter.

In addition there would be a smaller number of other cancers and one might conclude that the average result could be around some 200 deaths over a longish period of time.

Many of the possible accidents to reactors will result in damage; some in high cost to repair and high cost in lost output, but many could have a low probability of causing death. Rasmussen assesses that only 1 in 50 of reactor melt-out accidents would give the sort of casualty figures I have quoted above. This may well be so; but this number is not material, to my present argument.

I have been concerned at the views expressed in many places, including the US document WASH-1250 that the "worst" accident which can be foreseen would release only one part per million or thereabouts of the radioactive inventory of volatile fission products and that such release is a factor of 10 lower than the guide lines of US 10CFR Pt 100.

An accident which releases to atmosphere 100 curies or so of iodine and proportunate amounts of tellurium, caesium, etc, must obviously cause severe disruption to the plant, but is unlikely to hurt anyone away from the site. I exclude site staff as the accident is not sufficiently well defined either as to its source, duration, and whether associated with fire, mechanical damage etc. In fact, the release of 10^4 curies of I-131 and associated volatiles is unlikely on a statistical basis to cause one death in the surrounding population during the next 10-20 years. This leads me to the same conclusion as for non-nuclear accidents - that many can occur which may be very expensive but not injurious. This could lead us to the view that concern for these accidents should vest with those carrying financial responsibility - somewhat in line with the theory that insurance and financial interests will adequately protect. This is not my view. Many accidents causing small damage or even extensive damage but no casualties could have developed in some other way, at some other time, into a more serious accident. Accidents have started through local blockage through wrong fuel loading or misoperation of the charge machine as at Chapelcross and Saint Laurent. Those were costly - they could have been worse. Several reactors have depressurised when not under power; there have been problems with vibration of rotating machinery, of fuel and of heavy neutron shields. There have been maloperation of valves, cracks in valves, pipes etc. Many of these events have been such that had they occurred at some other time the result could have been disastrous.

In conclusion, then, I would not now concentrate too much on the target line concept, although it has limited uses. I am concerned with the broad objective that the chance of an accidental release in the range 10^5 to 10^7 curies of I-131 etc should be made very low - it is tempting to say as low as possible - but should this be 10^{-3} per year? This is about as low as some of the serious non-nuclear accidents. I do not think this is good enough:

(i) in view of the continued build-up of nuclear power stations

(ii) in view of the inability to change or switch off nuclear power in the year 2,000 when we are wholly or substantially dependent on this power.

As we can expect nuclear power of the western world to accumulate 10^4 reactor years based on present technology and if we wish to accept only a small chance of a major release, then we must aim to keep the chance of a serious accident as low as 10^{-5} to 10^{-6} per reactor per year.

Anyone who has spent much time seriously studying reliability of plant, including the interrelationship of equipment and people - whether manufacture, inspection, maintenance or operation - will seriously doubt whether we can yet achieve failure rates of complex composite systems as low as would be necessary to meet this overall target of 10^{-6} per reactor per year. This is in keeping with Rasmussen's conclusions about the two US plants he studied and, even so, he predicts the consequences of operating 100 nuclear plants in the US to be about 2 orders of magnitude less than several other man-made or natural disasters (Fig. 2).

I agree in general with his assessment, yet I am surprised to hear that there is a growing determination to try to protect against hazard which can be assessed on present evidence to be one or two orders of magnitude lower - ie $10^{-7}/10^{-8}$ per reactor per year - such as the random crash of a commercial aircraft.

This raises several questions:
(a) Should we protect against any event of any low frequency once it has been identified (even if there are others of greater likelihood - as through common mode failure, oversight etc) to deal with what lies beyond our present competence or organising ability?

(b) How much money should we pay for this - £10^5, £10^6 or £10^7?

(c) Should similar protection be provided to cope with non-nuclear hazards of comparable severity?

(d) Should the money be better spent; not as an investment against low frequency-high consequence risks, as there is a high probability that the investment will be wasted, or should it be spent on the less spectacular events of common occurrence?

Excerpt from Testimony of D. Okrent to California Legislature (and to Joint Committee on Atomic Energy)

Consequences of Accidents

Severity of accidents vs. risks and consequences.

I would like to conclude my testimony in this area; hence, I have placed it out of the original order in your agenda. Let me first define risk as the product obtained by multiplication of the magnitude of a hazard by the probability of the hazard.

I am not a medical or biological expert. I have read with interest comments from various groups concernign the treatment of health effects in draft WASH-1400 and am waiting to see what the final report says in this regard. I should note that I questioned the evacuation model used when I first saw the draft report.

One may reasonably anticipate that after issue of the final report there will again be a lack of agreement concerning its accuracy, its completeness and the stated uncertainties. The problem is not deterministic with all the input parameters specified. Matters of judgment are involved. And, with further detailed examination and with more knowledge, questions on the data, the analysis, and the assumptions employed will arise, suggesting that the risks calculated are either too small or too large. For purposes of discussion, let me assume that the overall estimates of risk will not be radically different in the final version of WASH-1400, and that semi-quantitative critiques of the new report by groups like the American Physical Study Group, the Union of Concerned Scientists, the Environmental Protection Agency, and the reactor designers will exhibit a spread in opinion not unlike that expressed earlier.

I must now ask some questions. How quantitatively does society know other risks to which it is exposed? From food additives and drugs? From earthquakes? From the failure of dams? From the shipment and storage of chemicals? From the burning of coal? From the discharge of mercury and other non-radioactive industrial wastes which may remain in our world forever.*

I must conclude that most, if not all, of the above are poorly quantified. And that most, if not all, pose threats to the public health and safety equal to or greater than nuclear power plants.

Let me raise some further questions.

What constitutes an acceptable hazard and an acceptable risk, either to society or to individuals who are directly exposed to some technological structure like a dam? What uncertainty in the estimates of the probabilities becomes unacceptable, whether the uncertainties arise from an absence of data or a difference in judgment among experts? Is there some number of people who might be killed in an improbable event which is so large as to be socially unacceptable?

At UCLA we have been trying to gain some perspective on such questions. We studied the risk to people on the ground near Los Angeles (LAX) and Hollywood Burbank Airport from the crash of commercial airliners.** Using a similar statistical approach to that employed in deciding whether a nuclear reactor requires protection against airplane crashes (because of exceeding the 10^{-7} guideline per cause), we calculated that the probability of a commercial aircraft crashing directly into the grandstand of Hollywood Park while it was occupied by a large crowd is about 10^{-5} per year. Postulating such a crash, we estimated the probable fatalities to be in the vicinity of 3,000 to 8,000; the maximum number of fatalities which might result was estimated to be about 30,000. Crashes into various shopping centers or hotels, etc. at either airport could produce lesser number of casualties with about the same probability. On the average it is estimated that about 5 people on the ground within 5 miles of each airport would be killed every ten years from such

**UCLA-ENG-7424 by K. Solomon, et al.

*The tragic mercury poisoning at Miramata, Japan is a graphic example.

crashes. Presumably a similar figure exists for most busy airports.

This was a statistical study. We may have no such casualties at either airport in the next 20 years. Or we may have more.

We also made a brief study of the potential effects of sudden complete failure of 10 dams in California and made crude estimates of their failure probabilities.* Sudden gross failure of a large dam has occurred in France in recent times; the equivalent of failure (overtopping due to a landslide) led to a few thousand deaths in Italy; and there is expert opinion that the Van Norman Dam would have undergone gross failure in the 1971 earthquake, had it been more nearly full.

The crude estimates of failure probabilities per year obtained in this study are given in Table 1. The estimated effects for sudden complete failure of these dams when filled to capacity are given in Table 2.

We think the estimated mortalities for sudden complete failure are fairly accurate. The probabilities of failure are clearly rough estimates, and could easily be wrong by a factor of more than 10.

In brief, the potential mortalities range from 14,000 to 260,000. The estimated probabilities of failure range from roughly 1 in 40 per year to one in 30,000 per year.

Are such hazards acceptable? If so, with what probability? What uncertainties in the estimates of such probabilities are acceptable? I understand that a law has been passed with regard to the safety of dams in California, that some steps have already been taken to reduce risk, and that further studies are underway. To my knowledge, there exists no acceptance limit on the maximum permissable deaths which postulated dam failure might cause. Nor do I know of quantitative criteria with regard to an acceptable probability of dam failure or uncertainites therein. Surely, that does not mean that the risk is zero.

I recognize that there is a panel of highly competent experts advising the State Office on such matters. However, based on a study performed by UCLA**, as well as listening to the opinions of various experts on the probability of exceeding some specific value of seismically-induced vibratory motion, I know that individual opinions on probabilities can vary by a factor of 10,000 (e.g. 10^{-2} per year versus 10^{-6} per year, or 10^{-4} per year versus 10^{-8} per year).

Does the California Legislature require that the opinions of only the most conservative seismologists and geologists be

*UCLA-ENG-7423 by P.Ayyaswamy, et al.
**UCLA-ENG-7515 by D. Okrent

TABLE 1

The results of the investigation could be summarized as follows:

Prediction of Earthquake in California by Fault Theory, Using a Computer

Name of Dam	Estimated Probability of MM VIII per Year*	Estimated Probability of MM IX per Year**	Estimated Probability of MM X per Year***
St. Francis[†]			
Van Norman	0.014	0.0049	.00003
San Andreas	0.044	0.014	.003
Lower Crystal Springs[†]			
Stone Canyon	0.012	0.0014	.0003
Encino	0.012	0.0013	.0002
San Pablo	0.076	0.032	.011
Folsom[1]			
Shasta[†]			
Chatsworth	0.013	0.0028	.00003
Mulholland[†]			
Upper San Leandro	0.12	0.063	.023
Lake Chabot	0.12	0.057	.021

*Signifcant Probability of Failure

**Substantial Probability of Failure

***High Probability of Failure

NOTE: Values for significant and substantial probabilities indicated in the above table correspond to the values at an order of magnitude higher as reflected in the computer output. Please see explanation in the text.

†These are Gravity type (concrete dams) for which the failure basis of Reference 2 is not applicable. Gast[22] has provided an estimate of 10^{-4} per year for world-wide concrete dams due to all causes.

TABLE 2

Estimated Effects of Total and Instantaneous

Failure of Dam Filled to Capacity

Name of Dam	MORTALITIES† Day	MORTALITIES† Night	DAMAGE ASSESSED IN U.S. DOLLARS
St Francis	- - - - Not calculated - - - -		
Van Norman	72,000	123,000	3×10^8
San Andreas / Lower Crystal Springs	21,000	33,000	1.1×10^8
Stone Canyon	125,000	207,000	5.3×10^8
Encino	11,000	18,000	5×10^7
San Pablo	24,000	36,000	7.7×10^7
Folsom	260,000	260,000	6.7×10^8
Chatsworth	14,000	22,000	6×10^7
Mulholland	180,000	180,000	7.2×10^8
Upper San Leandro / Lake Chabot	36,000	55,000	1.5×10^8
Shasta	34,000	34,000	1.4×10^8

†No allowance for evacuation. See Section 8 which indicates modest reductions for the two dams considered.

used in assessing dam safety? If questions concerning the uncertain source mechanisms of earthquakes, such as raised by Prof. Clarence Allen, are applicable to nuclear reactors, are they not applicable to dams, of which we have many in California?

Will the California Legislature by a two thirds vote, determine that the safety of dams has been demonstrated by comprehensive testing in actual operation substantially similar physical systems? How will the demonstration be done? In fact, can such a demonstration be done? What acceptance criteria will be used by the Legislature, especially for dams designed before soil liquefaction was a well-recognized phenomenon and for which little if any seismic design criteria existed. Will the Legislature require complete insurance coverage or close down the dams?

In passing, I might note that while Folsom Dam has Sacramento in its flood plain and was calculated to have the capability of inflicting about 250,000 deaths, UCLA is downstream of Stone Canyon Dam. We estimated Stone Canyon Dam has the potential for producing a flood height of 15 to 20 feet at Wilshire Boulevard and between 125,000 and 200,000 deaths.

Let me briefly discuss one more example from non-nuclear technology. At UCLA we tried unsuccessfully to obtain information from many of the largest chemical companies in the U.S. concerning the stored quantities of various chemicals having a considerable potential for inflicting harm on the public health and safety. We would have been interested in learning the sites of such storage and the safety criteria employed. We could then have done probabilistic studies of consequences. As I said, we received no information. I do know from informal discussion, however, that huge quantities of dangerous chemicals are stored within striking distance of population centers.

We have recently been trying to learn how the storage of dangerous chemicals is regulated in California. Our preliminary information indicates that both the knowledge and control of such risks is limited.

Does the Legislature know? Should it be active in this matter, since frequently an industrial park in one city is adjacent to a residential area in a neighboring city? Has the Legislature examined and found that there exists no risk to public health and safety from the shipment, storage, and use of chemicals employed in California's agricultural and food industires?

The shipment and storage of liquified natural gas (LNG) and similar hydrocarbons has been receiving increasing attention in recent years. Estimates of the potential hazard run into the many thousands of deaths for some postulated accidents. Accidents may occur at on-shore storage facilities, to the ship in or near port, or by grounding or collision of a tanker near a populated area away from port. I have heard estimates of the probability of such a collision in New York City harbor are about one in 50,000 per year for an LNG tanker, and one in 5000 per year for an LNG-laden barge. Such estimates are inferred and subjective, and critics can arrive at larger probabilities.

I believe that the Federal Power Commission has primary responsibility for the on-shore aspects of this technology, while the Coast Guard is responsible for the shipping aspect. I have, as yet, not been able to ascertain what quantitative hazard or risk criteria either group uses in reviewing such facilities. I have recently learned that in California, eight different agencies share the safety responsibility for energy facilities, and that criteria are now being developed which will address the issue of what is an acceptable risk to members of the public.

To my knowledge, none of the currently available sources of electric power is risk-free. And our preliminary studies at UCLA indicate that if controlled fusion power is actually developed, it may have a radioactivity hazard potential not vastly different from fission reactors and have to undergo similar safety precautions. For example, we expect that the actual tritium inventories will be much larger than the numbers usually mentioned; and we can postulate low probability accidents which might rupture the containment and volatilize some of the highly radioactive structural materials.

And coming from the Chicago area to Los Angeles, both with smog problems, I am painfully aware that essentially every study that I have seen, from various countries, indicates that the burning of coal for electricity has equal or greater overall effects on the public health and safety than nuclear power, mining accidents aside.

Why have I raised these examples? It is primarily to illustrate only some of the problems involved in judging acceptable risk. I have not brought in the question of benefits, which are an equally complex matter when one introduces effects on the national and international economy, and even potential political upheavals or war in the long term.

I have heard knowledgeable opinions that there are probably <u>not</u> enough resources available in the U.S. to raise the safety standards of existing non-nuclear technology so that it does not exceed either the risk from nuclear power reported in draft WASH-1400, or that estimated by most critics of the draft report. I am inclined to agree.

The problem is not simple. And the answer should not be simplistic. In fact, society is not risk-free.

I.2 A BRIEF PERSONAL INTRODUCTION

I was formaly initiated into the regulatory world of nuclear power reactors in November 1963 when I attended the 51st meeting of the Advisory Committee on Reactor Safeguards (or ACRS) in Washington, D.C. as a newly appointed member. Some years previously I had served the ACRS as a consultant, and for many years during the 1950's I had worked on problems concering the safety of fast reactors, particularly the EBR II (Experimental Fast Breeder) reactor built by the Argonne National Laboratory. And I had been heavily involved in developing the Atomic Energy Commission program in fast reactor safety during the 1950's. I considered myself knowledgeable about the reactor physics of most reactor types; however, the safety questions relating to reactors other than sodium-cooled fast reactors were only partially known to me, and I had little familiarity with the regulatory process or the matters which were receiving emphasis by regulatory groups in reviewing the power reactors, experimental power reactors, and engineering test reactors then under construction, under consideration for construction, or going into operation. Besides myself, the membrs of the ACRS present at the 51st meeting of the statutory ACRS were the chairman, David Hall, whose normal position was that of head of the K Division (or Reactor Division) at Los Alamos Scientific Laboratory; Charles Williams, Assistant Vice-President of Liberty Mutual Insurance Company in Boston; John Geyer, Professor of Sanitary Engineering at Johns Hopkins University; Herbert Kouts, an experimental reactor physicist at the Brookhaven National Laboratory; William Manly, a metallurgist from Oak Ridge National Laboratory who, like me, was a brave new member; Henry Newson, a Professor of Physics at Duke University; Kenneth Osborn, Chief Engineer for the General Chemical Division of Allied Chemical Corporation; Donald Rogers, Manager of Project Analysis of the Central Reserach Laboratory of Allied Chemical Corporation, and a man very knowledgeable in the design and behavior of large chemical-mechanical systems; Reuel Stratton, a consulting engineer who had formerly been with Travelers Insurance Company; and T. S. (Tommy) Thompson, Director of the Massachusetts Institute of Technology nuclear reactor and Professor of Nuclear Engineering at the Massachusetts Institute of Technology. Two other ACRS members who were not present at the 51st meeting were Franklin Gifford, a meteorologist with the U.S. Weather Bureau in Oak Ridge Tennessee, and Leslie Silverman, Professor of Engineering in Environmental Hygiene at the School of Public Health, Harvard University.

Actually the Advisory Committee on Reactor Safeguards had its inception in late 1947. In June of that year, the Atomic Energy Commission discussed with its General Advisory Committee the problem of evaluating the safety of nuclear reactors. It was concluded that a panel be established to advise the Commission on reactor safety matters. This was done by inviting highly qualified individuals with background and appropriate scientific discipline to serve on a Reactor Safeguards Committee reporting to the AEC General Manager. Beginning in the fall of 1947, meetings of this

Committee were held to consider and advise upon matters referred to it. In 1951 the Commission also established an Industrial Committee on Reactor Location Problems charged with responsibility of advising upon the siting of nuclear reactors. The Reactor Safeguards Committee and the Industrial Committee on Reactor Location Problems were combined by the AEC in July of 1953 and renamed the Advisory Committee on Reactor Safeguards. The September 1953 issue of the trade journal "Nucleonics" describes the formation of the new Committee, lists its then members, and quotes a statement made to the Joint Committee on Atomic Energy by Edward Teller, formerly Chairman of the Reactor Safeguards Committee.

The statement by Teller represents one of the earliest opinions concerning AEC philosophy on reactor safety, and is quoted below in the excerpt from Nucleonics.

EXCERPT FROM NUCLEONICS SEPTEMBER 1953

Reactor Advisory Committees Reorganized

Two advisory committees to the Atomic Energy Commission--the reactor safeguard committee and the industrial committee on reactor location problems--have been combined to form a new group known as the advisory committee on reactor safeguards.

The new committee will have the following functions:

1. Reviewing hazards summary reports prepared by organizations planning to build or operate reactor facilities, including criticality experiments.

2. Advising the AEC regarding the consistency of proposed reactor locations with accepted industrial safety standards, taking into account the proposed exclusion areas for reactors and the proximity of surrounding population and property.

Members are as follows: C. Rogers McCullough, Monsanto Chemical Co., chairman; Manson Benedict, Massachusetts Institute of Technology, Willard P. Conner, Jr., Hercules Power Co.; R. L. Doan, Phillips Petroleum Co.; Hymer Friedell, Western Reserve University; I. B. Johns, Monsanto Chemical Co.; Mark M. Mills, North American Aviation, Inc.; K. R. Osborn, Allied Chemical and Dye Corp.; D. A. Rogers, Allied Chemical and Dye Corp.; Reuel C. Stratton, Travelers Insurance Co.; Edward Teller, University of California; Abel Wolman, Johns Hopkins University; Harry Wexler, U.S. Weather Bureau, Department of Commerce; and C. R. Russell, secretary, AEC.

Very little has been said publicly on AEC philosophy regarding reactor safeguards. Therefore, the following statement, made to the Joint Committee on Atomic Energy by Edward Teller, formerly chairman, reactor safeguard committee, is of particular interest:

"Up to the present time we have been extremely fortunate in that accidents in nuclear reactors have not caused any fatalities. With expanding applications of nuclear reactions and nuclear power, it cannot be expected that this unbroken record will be maintained. It must be realized that this good record was achieved to a considerable extent because of safety measures which have necessarily retarded development.

"The main factors which influence reactor safety are, in my opinion, reasonably well understood. There have been in the past years a few minor incidents, all of which have been caused by neglect of clearly formulated safety rules. Such occasional accidents can not be avoided. It is rather remarkable that they have occurred in such a small number of instances. I want to emphasize in particular that the operation of nuclear reactors is not mysterious and that the irregularities are no more unexpected than accidents which happen on account of disregard of traffic regulations.

"In the popular opinion, the main danger of a nuclear pile is due to the possibility that it may explode. It should be pointed out, however, that such an explosion, although possible, is likely to be harmful only in the immediate surroundings and will probably be limited in its destructive effects to the operators. A much greater public hazard is due to the fact that nuclear plants contain radioactive poisons. In a nuclear accident, these poisons may be liberated into the atmosphere or into the water supply. In fact, the radioactive poisons produced in a powerful nuclear reactor will retain a dangerous concentration even after they have been carried downwind to a distance of ten miles. Some danger might possibly persist to distances as great as 100 miles.

"It would seem appropriate that Federal regulations should apply to a hazard which is not confined by state boundaries. The various committees dealing with reactor safety have come to the conclusion that none of the powerful reactors built or suggested up to the present time are absolutely safe. Though the possibility of an accident seems small, a release of the active products in a city or densely populated area would lead to disastrous results.

"It has been therefore the practice of these committees to recommend the observance of exclusion distances, that is, to exclude the public from areas around reactors, the size of

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"It would seem appropriate that Federal regulations should apply to a hazard which is not confined by state boundaries. The various committees dealing with reactor safety have come to the conclusion that none of the powerful reactors built or suggested up to the present time are absolutely safe. Though the possibility of an accident seems small, a release of the active products in a city or densely populated area would lead to disastrous results.

"It has been therefore the practice of these committees to recommend the observance of exclusion distances, that is, to exclude the public from areas around reactors, the size of the area varying in appropriate manner with the amount of radioactive poison that the reactor might release. Rigid enforcement of such exclusion distances might hamper future development of reactors to an unreasonable extent. In particular, the danger that a reactor might malfunction and release its radioactive poison differs for different kinds of reactors.

"It is my opinion that reactors of sufficiently safe types might be developed in the near future. Apart from the basic construction of the reactor, underground location or particularly thoughtfully constructed safety devices might be considered.

"It is clear that no legislation will be able to stop future accidents and avoid completely occasional loss of life. It is my opinion that the unavoidable danger which will remain after all reasonable controls have been employed must not stand in the way of rapid development of nuclear power. It also would seem that proper legislation at the present time might make provisions for safe construction and safe operation of nuclear reactors.

"It would seem reasonable to extend the AEC procedures on reviewing planned reactors and supervising functioning reactors to nuclear plants under the control of private enterprise. To what extent these functions should be advisory or regulatory is a difficult question. I feel that ultimate responsibility for safe operation will have to be placed on the shoulders of the men and the organizations most closely connected with the construction and the operation of the reactor."

With the passage of the 1954 Atomic Energy Act, it became possible for private companies to build and operate nuclear reactor facilities under license. At the same time this act assigned to the Commission the responsibility through licensing procedures of protecting the health and safety of the public. Applications for license as well as projects of the Commission°s power and military propulsion development programs were referred to the ACRS for advice. In 1955 the Commission established within its own organization a full time staff for the evaluation of reactor hazards. The workload of the Commission, its staff and the ACRS continued to increase.

In July of 1956, the Joint Committee on Atomic Energy requested that the Atomic Energy Commission make a study of the possible effects of an uncontrolled reactor accident. This resulted in the publication in March 1957 of a study entitled "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants" (WASH 740). This report, together with the demands by industry for indemnity, the intervention by the United Auto Workers in the Power Reactor Development case (that is, the Fermi reactor), and the hearings held by the Joint Committee on Atomic Energy, formed the basis for legislative action on September 2, 1957. By this amendment to the Atomic Energy Act, the Advisory Committee on Reactor Safeguards was established as a statutory Advisory Committee to the Atomic Energy Commission to review safety studies and facility license applications referred to it, and make reports thereon; advise the Commission with regard to the hazards of proposed or existing reactor facilities, and the adequacy of proposed reactor standards; and to perform other such duties as the Commission might request. Members are appointed by the Commission for a term of four years each. One member is designated by the Committee itself as its chairman. The law allows a maximum of 15 members. The members are appointed from private life, and since they are not dependent upon the AEC for their livelihood, are free of economic pressure from AEC. The reports on licensed facilities made by the Committee to the Commission are required by law to be made a part of the record of the application and available to the public. This last aspect of the law undoubtedly relates to the previous history of the construction permit review for the Fermi reactor, during which the non-statutory ACRS prepared a report which had several strong reservations concerning safety matters still to be resolved for the Fermi reactor. The Atomic Energy Commission issued a construction permit for the Fermi reactor without making public the ACRS report. Copies of the ACRS report became available to the Joint Committee on Atomic Energy and others some weeks later, however, and created considerable controversy.

The manner in which the ACRS functions is of some interest. Although changes have occurred through the years, much has remained the same, and during much of the 1960's the following procedures were followed: Safety Analysis Reports and other documents prepared by the Applicant, describing in varying detail the design of the reactor to be reviewed, are furnished to all members of the ACRS well in advance of the Committee meeting at which the project will be considered. There is an opportunity for individual study. A summary and evaluation of the project and its salient safety aspects is prepared by the permanent technical staff within the Atomic Energy Commission. This too is furnished to the members of the Committee in advance of the meeting. The Committee may hold one or more subcommittee meetings prior to the meeting of the full Committee. At the subcommittee meetings and the full Committee meetings, the reactor designed and/or reactor operator appears before the assembled Committee, and the designer describes his reactor, in particular its safety features. There is a free technical exchange between the ACRS members and the representatives of the Applicant. The Committee would then, in closed session, formulate its advice to the Commission. This advice is provided in the form of a Committee report which is written to the Chairman of the Atomic Energy Commission. This report is signed by the Chairman of the Committee but is prepared by the Committee as a whole. The ACRS report is almost always a concensus. An individual member has the right to issue his own separate opinion as part of the report.

It is important to note that the statutory ACRS elects its own Chairman, establishes its own agenda, gets its own consultants, and decides for itself when it is ready to write a report. And, while the ACRS has obligations to respond to specific requests for advice from the Commission, it can also take up matters falling in its general province of responsibility on its own initiative.

That this very considerable independence of function should exist is, in a sense, remarkable. That such independence did not occur automatically can be ascertained from a review of the efforts from within the AEC to limit the freedom and scope of the pre-statutory Reactor Safeguards Committee, to restrict the oeprations of the statutory ACRS in the first year of its existence, and then again to try to restrict it in 1966.

The agenda at the 51st meeting, November 7-8, 1963, covered various topics, including the following.

The Committee discussed the proposed dockside operation of the Nuclear Ship Savannah at Galveston, Texas and wrote a report to the Commission on this matter. The Committee heard a presentation concerning the possible operation of one of the "production" reactors at Savannah River for the production of a special isotope, Curium 244. This was a preliminary information session on curium production, and the Committee did not try to reach any conclusions during this meeting concerning hazards associated with the proposed method of operation. The Committee also reviewed with members of the Pathfinder plant their planned organization for full power operation of that relatively small boiling water reactor. The Committee prepared a letter report concerning aspects of the reactor safety research program, and a letter concerning aspects of design of shipment casks for irradiated fuel. In addition, there were discussions of various topics, including the proposed Ravenswood reactor at a site in the heart of New York City, for which Consolidated Edison had submitted a preliminary proposal. And there were discussions concerning various other things in progress, such as the Fermi reactor, the LOFT project and the Dresden reactor. This all took place during a meeting which lasted two days.

As a new member, it was for me a sudden immersion into a host of topics, many of which were completely strange and for which the Committee had little time to provide background. I assumed (rightly) that it was my principal responsibility to consider the reactor physics aspects of those things being discussed, and to raise any questions in this area that seemed to be important. I tried, also, to consider matters relating to the kinetic behavior or reactors (during transients). And, since I had a background in reactor safety research, I tried to become active quickly in that aspect of Committee activities. Similarly, Bill Manly, a metallurgist, was expected to provide insight into materials-related aspects of the topics under review.

For the large number of topics which were of a continuing nature, which mainly represented matters having been reviewed in previous months or even some years ago, new members just had to play catch-up as quickly as they could, and, depending on the nature of the matter, months or even years could be involved before one had achieved a relatively broad perspective on many of the topics. Looking back some 14 years later at this, my first meeting as a member of the ACRS, I can see that it was a rather brief discussion of the proposed Ravenswood reactor, and some long-since-forgotten paragraphs in the ACRS safety research report which were of the most long-term significance.

CHAPTER 2
ON REACTOR SITING

2.1 A BRIEF OVERVIEW

The complexity and great breadth of the field of nuclear reactor safety make it impossible to even attempt to review all of its history. Contributions to the evolutionary process have come from many countries and from many individuals and institutions within a country. Rather than try to do justice to all the participants in this process, we shall try in this semi-historical review, to look at several aspects of reactor safety from the particular vantage point of the U.S. Advisory Committee on Reactor Safeguards (ACRS). While opinion and work by the Regulatory Staff of the USAEC and USNRC will be referred to frequently, the files of the Staff are not generally available to the author to provide detailed insight into the pre-decisional thinking of that group. References to other groups in the U.S. and in other countries will be very limited; such material will be introduced, in general, only as it relates in a particularly strong way to ACRS action or opinion.

Reactor siting has been the subject of several International Atomic Energy Agency Conferences (International Atomic Energy Agency, 1962, 1963, 1967, 1973, 1975) and has been featured in the reactor safety discussions at the four United Nations "Atoms for Peace" Conferences held in Geneva (United Nations, 1956, 1958, 1965, 1972). Many countries have developed their own approaches to dealing with the siting of power reactors. However, we shall see that, in general, site criteria are far from definitive, tend to be comparative or relative, and are built around precedent, once some history of siting has occurred.

This examination of reactor siting will deal primarily with demographic (population) considerations. We will find it convenient to divide the discussion loosely into several time periods: Pre-1961; 1960-1962; 1962-late 1965; late 1965-1966; 1967-1973; and 1973-present.

We shall see that in 1950 the first AEC Reactor Safeguards Committee produced a very restrictive "rule of thumb" site criterion which related the reactor power to the required "exclusion distance" (or lightly populated region surrounding the reactor, under control of the reactor operator). This site criterion assumed gross release of radioactivity from an uncontained reactor, and required a large exclusion radius. However, pressures were already building in 1950 to construct reactors at sites which strongly violated the "rule of thumb", and the concept of reactor containment emerged quickly.

In addition to several test and experimental reactors, three light water power reactors (LWR) of substantial size, all with containment, were approved for construction in the mid 1950's, including the one at the Indian Point New York site, which remains to this day the most populated site used in the U. S. for an LWR. In the late fifties several other smaller reactors, all with containment, were approved. However, during the same period, a few small power reactors (~60 MWt) were proposed for sites within or at the very edge of small or medium size cities; these were rejected or forced to move to somewhat less populated sites, which were grudgingly accepted by the ACRS.

The review of these reactors was on a case-by-case basis, a judgment being reached concerning acceptability of each specific reactor-site combination. No AEC Reactor Site Criteria were published; and, from a review of the ACRS minutes and other documents, there does not appear to have been any single guiding philosophy.

The Atomic Energy Commissioners themselves seem not to have exercised strong direction on the development of siting criteria in the period prior to 1960. They may, however, have exercised strong or decisive influence on the favorable decisions for construction of Shippingport, Dresden, and Indian Point. (The detailed history of these early, precedent-setting reactor reviews is not available to the author). In the late 1950's, the Commissioners appeared particularly interested in the development of written criteria, as specific as possible, to help avert a recurrence of awkward situations wherein considerable work was first expended on reactor proposals involving sites which were then evaluated unfavorably by the ACRS and/or the AEC Regulatory Staff.

The first rather detailed expression of ACRS philosophy on power reactor siting came in its reports to the AEC on this subject in October and December of 1960. Independently, though in close association, the AEC Regulatory Staff (then called the Hazards Evaluation Branch) under the leadership of Dr. Clifford A. Beck, was developing site criteria. And in 1961, the AEC published for comment in the Federal Register proposed site criteria which included the concepts of an exclusion area, a low-population (or evacuation) zone, and a population center distance, and wherein, for purposes of site evaluation, an accident was postulated in which the noble gases and half the radioiodine were released to a containment assumed to maintain its integrity, and guideline doses of 25 rem whole body and 300 rem to the thyroid were not to be exceeded under specified conditions. This postulated accident, whose consequences were not to be exceeded by any credible accident (and which was called the maximum credible accident or MCA), became the focus of siting evaluation.

We shall see that the 10 CFR Part 100 Reactor Site Criteria, which were adopted by the AEC in 1962, led to a soemwhat stylized reactor safety review until 1966, during which primary emphasis was placed on containment design and engineered safety features which could enable the increasingly higher power reactors to meet the guideline doses. Reactors were receiving construction permits based on rather general safety criteria and sketchy design information. And most safety improvements which developed were related to the MCA.

We shall also see that the Part 100 Site Criteria of 1962 (which are reproduced at the end of this introductory section) were developed in a manner such that they lost the basic impact of two major philosophic recommendations made by the ACRS in its October, 1960 report, namely the integrated man-rem dose concept and, more specifically, the recommendation that there be no catastrophic effects to a population center from an uncontained accident much worse than the MCA. And we shall find the Regulatory Staff and the ACRS approving new reactors without direct consideration of these concepts during the period 1960-1965.

Beginning in late 1965, the ACRS began to give increased emphasis to events that might lead to consequences far worse than the stylized MCA. And in 1966, there occurred a revolution in LWR reactor licensing which for the next twelve years was to lead primarily to greatly increased measures to reduce the probability of serious accidents, and to greatly increased attention to safety features needed to prevent core meltdown in the event of transients or accidents.

The AEC, in general, opposed the metropolitan or the near-metropolitan siting of power reactors, even if Part 100 could be met; however, for those reactors it approved, the Regulatory Staff stayed rather stringently within the prescriptions of Part 100 and its MCA (later designated the Design Basis Accident), and declined to publicly discuss or examine safety aspects related to accidents which went beyond the MCA (e.g. in which containment integrity was lost). This trend of not examining Class 9 accidents (accidents exceeding Part 100 guidelines), and not evaluating measures which might ameliorate their impact, remained with the Regulatory Staff essentially until 1976, when limited formal consideration of such accidents was first introduced under ACRS pressure into the review of the Floating Nuclear Power plant.

The regulatory stance of the Atomic Energy Commissioners themselves during the period 1960-1974 seems to have depended on the role they were playing. When the highly populated Bolsa Island reactor was under active consideration in the mid 1960's, the Commissioners resisted the development of detailed siting criteria which might pose a hurdle to this project. Similarly, when in 1966 the ACRS was about to formally recommend the development and implementation of measures to deal with questions related to large scale core melt, the Commissioners proposed a Task Force to "study" the matter, thereby delaying (indefinitely) such a recommendation.

On the other hand, when required to accept a judicial role, such as that in connection with review of the Malibu hearing, the Commissioners supported the more conservative position of the Atomic Safety and Licensing Board over that of the Regulatory Staff. In their Malibu decision, the Commissioners, in a sense, provided the first quantitative guideline on acceptable risk, namely, they stated that the fact that the fault in question at Malibu had not moved in 14,000 years did not provide adequate assurance that the plant need not be designed for surface displacement.

It must be emphasized that there is little firm basis for judging that any approach or position adopted by individuals or groups during this era was "right" or "wrong". Up to the present, at least, the regulation of nuclear power reactors has led to much less adverse effect on the public health and safety than essentially any other equivalent technology. It has been argued by some that there has been more protection (and expense) than appropriate.

From 1966 on, while there has been no significant change in the Reactor Siting Criteria, and while they are pursued rather legalistically by the Regulatory Staff (and therefore by applicants), there have been major changes in reactor design criteria and in the depth of safety review aimed at reducing the probability of an accident exceeding the guidelines of Part 100. However, the unwillingness to "look" at Class 9 accidents by either the nuclear industry or the Regulatory Staff may have led to less than an optimum approach with regard to overall reduction in public risk for the same effort and cost. This matter is now getting active attention, as is the matter of site criteria. It remains to be seen whether the approaches used in the decade 1966-76 represented a near optimum in efficacy.

PART 100—REACTOR SITE CRITERIA

Pursuant to the Administrative Procedures Act and the Atomic Energy Act of 1954, as amended, the following guide is published as a document subject to codification, to be effective 30 days after publication in the FEDERAL REGISTER.

Statement of considerations. On February 11, 1961, the Atomic Energy Commission published in the FEDERAL REGISTER a notice of proposed rule making that set forth general criteria in the form of guides and factors to be considered in the evaluation of proposed sites for power and testing reactors. The Commission has received many comments from individuals and organizations, including several from foreign countries, reflecting the widespread sensitivity and importance of the subject of site selection for reactors. Formal communications have been received on the published guides, including a proposed comprehensive revision of the guides into an alternate form.

In these communications, there was almost unanimous support of the Commission's proposal to issue guidance in some form on site selections, and acceptance of the basic factors included in the proposed guides, particularly in the proposal to issue exposure dose values which could be used for reference in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence.

On the other hand, many features of the proposed guides were singled out for criticism by a large proportion of the correspondents. This was particularly the case for the appendix section of the proposed guides, in which was included an example calculation of environmental distance characteristics for a hypothetical reactor. In this appendix, specific numerical values were employed in the calculations. The choice of these numerical values, in some cases involving simplifying assumptions of highly complex phenomena, represent types of considerations presently applied in site calculations and result in environmental distance parameters in general accord with present siting practice. Nevertheles, these particular numerical values and the use of a single example calculation were widely objected to, basically on the grounds that they presented an aspect of inflexibility to the guides which otherwise appeared to possess considerable flexibility and tended to emphasize unduly the concept of environmental isolation for reactors with minimum possibility being extended for eventual substitution thereof of engineered safeguard.

In consequence of these many comments, criticisms and recommendations, the proposed guides have been rewritten, with incorporation of a number of suggestions for clarification and simplification, and elimination of the numerical values and example calculation formerly constituting the appendix to the guides. In lieu of the appendix, some guidance has been incorporated in the text itself to indicate the considerations that led to establishing the exposure values set forth. However, in recognition of the advantage of example calculations in providing preliminary guidance to application of the principles set forth, the AEC will publish separately in the form of a technical information document a discussion of these calculations.

These guides and the technical information document are intended to reflect past practice and current policy of the Commission of keeping stationary power and test reactors away from densely populated centers. It should be equally understood, however, that applicants are free and indeed encouraged to demonstrate to the Commission the applicability and significance of considerations other than those set forth in the guides.

One basic objective of the criteria is to assure that the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low in comparison with what might be considered reasonable for total population dose. Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible. Neither of these objectives were readily achievable by a single criterion. Hence, the population center distance was added as a site requirement when it was found for several projects evaluated that the specification of such a distance requirement would approximately fulfill the desired objectives and reflect a more accurate guide to current siting practices. In an effort to develop more specific guidance on the total man-dose concept, the Commission intends to give further study to the subject. Meanwhile, in some cases where very large cities are involved, the population center distance may have to be greater than those suggested by these guides.

A number of comments received pointed out that AEC siting factors included considerations of population distributions and land use surrounding proposed sites but did not indicate how future population growth might affect sites initially approved. To the extent possible, AEC review of the land use surrounding a proposed site includes considerations of potential residential growth. The guides tend toward requiring sufficient isolation to preclude any immediate problem. In the meantime, operating experience that will be acquired from plants already licensed to operate should provide a more definitive basis for weighing the effectiveness of engineered safeguards versus plant isolation as a public safeguard.

These criteria are based upon a weighing of factors characteristic of conditions in the United States and may not represent the most appropriate procedure nor optimum emphasis on the various interdependent factors involved in selection of sites for reactors in other countries where national needs, resources, policies and other factors may be greatly different.

AUTHORITY: §§ 100.1 to 100.11 issued under sec. 103, 68 Stat. 936, sec. 104, 68 Stat. 937, sec. 161, 68 Stat. 948, sec. 182, 68 Stat. 953; 42 U.S.C. 2133, 2134, 2201, 2232

SOURCE: §§ 100.1 to 100.11 appear at 27 F.R. 3509, Apr. 12, 1962.

§ 100.1 *Purpose.* (a) It is the purpose of this part to describe criteria which guide the Commission in its evaluation of the suitability of proposed sites for stationary power and testing reactors subject to Part 50 of this chapter.

(b) Insufficient experience has been accumulated to permit the writing of detailed standards that would provide a quantitative correlation of all factors significant to the question of acceptability of reactor sites. This part is intended as an interim guide to identify a number of factors considered by the Commission in the evaluation of reactor sites and the general criteria used at this time as guides in approving or disapproving proposed sites. Any applicant who believes that factors other than those set forth in the guide should be considered by the Commission will be expected to demonstrate the applicability and significance of such factors.

§ 100.2 *Scope.* (a) This part applies to applications filed under Part 50 and 115 of this chapter for stationary power and testing reactors.

(b) The site criteria contained in this part apply primarily to reactors of a general type and design on which experience has been developed, but can also be applied to other reactor types. In particular, for reactors that are novel in design and unproven as prototypes or pilot plants, it is expected that these basic criteria will be applied in a manner that takes into account the lack of experience. In the application of these criteria which are deliberately flexible, the safeguards provided—either site isolation or engineered features—should reflect the lack of certainty that only experience can provide.

§ 100.3 *Definitions.* As used in this part:

(a) "Exclusion area" means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

(b) "Low population zone" means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

(c) "Population center distance" means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

(d) "Power reactor" means a nuclear reactor of a type described in § 50.21(b) or 50.22 of this chapter designed to produce electrical or heat energy.

(e) "Testing reactor" means a "testing facility" as defined in § 50.2 of this chapter.

Site Evaluation Factors

§ 100.10 *Factors to be considered when evaluating sites.* Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

(a) Characteristics of reactor design and proposed operation including:

(1) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(2) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur.

(b) Population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.

(c) Physical characteristics of the site, including seismology, meteorology, geology and hydrology.

(1) The design for the facility should conform to accepted building codes or standards for areas having equivalent earthquake histories. No facility should be located closer than one-fourth mile from the surface location of a known active earthquake fault.

(2) Meteorological conditions at the site and in the surrounding area should be considered.

(3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow into nearby streams or rivers or might find ready access to underground water tables.

(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

§ 100.11 *Determination of exclusion area, low population zone, and population center distance.* (a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release[1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem[2] or a total radiation dose in excess of 300 rem[2] to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, due consideration should be given to the population distribution within the population center.

Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

[2] The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the AEC the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

NOTE: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.

Copies of Technical Information Document 14844 may be obtained from the Commission's Public Document Room, 1717 H Street NW., Washington, D.C., or by writing the Director, Division of Licensing and Regulation, U.S. Atomic Energy Commission, Washington 25, D.C.

2.2 REACTOR SITE CRITERIA - PRE-1961

In 1950 the USAEC Reactor Safeguards Committee issued the report WASH-3 (USAEC, 1950) containing the first publicly documented site guidance, in which it recommended an exclusion radius, R (miles) which depended on the reactor power, P(kw), according to the relationship

$$R = 0.01\sqrt{P}$$

Assuming gross uncontrolled release from an uncontained accident, and allowing in a rough way for meteorological conditions and the time requirements for evacuation, the Reactor Safeguards Committee estimated that beyond this minium distance, the radiation exposure should be 300 roentgen or less, or that evacuation should be practical. For a 30 MWt plant, this so-called "rule of thumb" led to an exclusion area (with no residents) of radius 1.7 miles.

For a 3000 MWt plant of the type commonly used today for electric power, the exclusion radius would be 17.3 miles.

In the book "Reactor Safeguards," (C. R. Russell, 1962) who served as Secretary of the Reactor Safeguards Committee, reviews the early history of reactor siting and discusses qualitatively the reasoning which went into formulation of the siting "rule of thumb". Russell also includes as Appendices several sections of WASH-3 which give technical backup for the siting formula, as well as an estimate of possible energy release from a reactor undergoing a large reactivity excursion with a resultant large sudden increase in power.

Russell recounts the very considerable attention given by the Reactor Safeguards Committee to reactor features pertaining to the potential for disruptive reactivity excursions, which could lead to gross core melting or pressure-driven disassembly. Not that the potential for core over-heating and melting from inadequate removal of fission product decay heat after shutdown of the chain reaction was ignored. But, reactivity excursions, and the potential for autocatalytic effects, that is, reactor features which inherently aggravate a reactivity excursion by adding reactivity due to the increase in power, received great emphasis. Considering the background of the Committee members, and the characteristics of some of the reactors then reviewed, this emphasis is not surprising.*

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*It is of some interest that back in 1950, the document WASH-3 called special attention to sabotage as a potentially important source of serious reactor accidents. WASH-3 was particularly concerned with sabotage by a knowledgeable person having access to the plant. About 16 years later, during the construction permit review of Turkey Point

(Continued)

The large exclusion distances required by the "rule of thumb" meant that rather few sites in the United States could qualify under this criterion for uncontained, large nuclear power plants. However, pressures built up very rapidly for the use of sites with smaller exclusion radii. Russell discusses the evolution of the concept of containment in connection with the not very high power, prototype Submarine Intermediate Reactor, eventually constructed within a large steel sphere near West Milton, New York, and the relatively low power Argonne Research Reactor, CP-5, located in a suburb of Chicago, Illinois.

The first "civilian" nuclear power plant was the Shippingport Atomic Power Station which was publically announced in October, 1953. This plant was designed, built and owned by the government, but operated by a utility (Duquesne Light Company) under the stringent guidance of the Division of Naval Reactors of the AEC. The Pressurized Water Reactor (PWR) clearly would not have met the 1950 "rule of thumb" at the Shippingport, Pennsylvania site which was about 420 acres in area and about 20 miles from Pittsburgh in a region having more population than had been characteristic of the remote sites employed for production and engineering test reactors; and a containment was provided around the reactor.

The proposal by Duquesne Electric Company to operate such a reactor on its grid had come in response to an invitation to utilities by the AEC in 1953 for proposals to construct a PWR.

The economic acceptability of the "rule of thumb" for power reactors had already been questioned because of the high cost of land for such large exclusion areas, and because of the desire or need of utilities to have their power plants near load centers. The decision leading to the selection of the Duquesne Light proposal appears to have been based in part on the interest in maintaining a technological program with a potential for application to future Naval Reactor projects and in part from the need to find a willing and able utility. It is not clear how matters such as surrounding population density entered into the decision making process on Shippingport.

Soon after Shippingport, proposals were made for utility companies for several privately owned, large power reactors and the AEC had to

(Continued)
3 and 4 in southern Florida, the question of attack on a reactor from an unfriendly country was raised by an intervenor, the AEC held that the Department of Defense and not the AEC was responsible for defense against any such aggression.

During the 1970's, the matter of sabotage, both by an insider and by armed intruders, was given much increased emphasis by the ACRS and the Regulatory Staff, and a considerable augmenmentation of protective requirements was initiated.

consider these for construction permits. All were to have containment vessels, and three of them were in the general vicinity of large cities: the Dresden 1 boiling water reactor (BWR), 35 miles southwest of Chicago, Illinois; the Indian Point 1 PWR, 24 miles north of New York City; and the Enrico Fermi fast reactor, 25 miles south of Detroit. Dresden 1 at 630 MWt was the largest light water reactor approved prior to 1961. Indian Point 1 at 585 MWt was at the most highly populated site approved for a reactor of that size, and ultimately has represented a landmark case. A letter from Harold Price, Director, AEC Division of Reactor Licensing to Vice President J. F. Fairman of Consolidated Edison dated May 16, 1955, states "at the meeting in New York on May 2, 1955, . . . it was agreed among the technical people present that there had not been assembled sufficient information on which to base an evaluation of the actual and potential hazard of operating a reactor of the type contemplated at the site you have chosen. Thus, while we have no reason to believe that such a reactor could not be operated safely at the chosen site, neither do we have sufficient information to allow us to state with reasonable assurance that it can be".

Actually no formal design criteria or siting criteria and rather little preliminary design information were available in 1956 when the ACRS reviewed the construction request. The applicant did submit five hazard reports dealing with the reactor site and with various design aspects in a conceptual fashion. The ACRS records note that there are about 16 one-family residences within a half-mile radius and about 45,000 inhabitants within a five-mile radius. There is no indication in the records of any emphasis on demographic considerations in the review of Indian Point, nor is there any indication of what criteria were used in reaching a judgment on the acceptability of this reactor which represented a very major step in the commercialization of nuclear power, and in which the promotional side of the AEC had great interest.

The ACRS report of January 6, 1956 on this project is on the following page. The minutes of the ACRS Subcommittee meetings held in August 29, 1955 and December 1, 1955 provide no information on the basis by which the ACRS judged this reactor-site combination to be acceptable. Nor is any insight on this particular question to be obtained from a paper co-authored by three ACRS members for the 1955 U.N. "Atoms for Peace" Conference (Geneva, 1955), although the paper does provide insight into what were then perceived to be the principal problems in reactor safety.

On February 15, 1956, while construction of the Shippingport reactor was being completed, and shortly after ACRS review of Indian Point, U.S. Senator Bourke Hickenlooper wrote to the ACRS asking questions concerning the siting of power reactors in populated areas. His letter is reproduced below, together with the response, which was made not by ACRS Chairman McCullough but by Dr. Willard F. Libby, Acting Chairman of the Atomic Energy Commission.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

C O P Y

January 6, 1956

Mr. K. E. Fields
General Manager
U. S. Atomic Energy Commission
Washington, D. C.

SUBJECT: Consolidated Edison Company

Dear Mr. Fields:

At the Sixteenth Meeting of the Advisory Committee on Reactor Safeguards the Consolidated Edison Company presented a review of their Nuclear Power Project. At this time they have calculated many of the nuclear characteristics and have outlined the design of the whole plant. A great deal of the experimental work remains to be done to verify the calculations and conduct tests of the essential features of the reactor. The Committee believes the approach is sound and sees no hazard problems which appear to be insoluble. From the hazard point of view, the Committee sees no reason to withhold a construction permit as long as it is thoroughly understood that the final design and Hazards Summary Report must be reviewed before the reactor goes into operation.

The Committee is favorably impressed with the competence of the staff and the programs being made on this reactor and the program which is being set up to solve the many problems connected with a reactor of this new design and type using thorium.

Sincerely yours,

C. Rogers McCullough
Chairman
Advisory Committee on
Reactor Safeguards

cc: H. L. Price, CA
C. A. Nelson, Inspection

C O P Y

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CONGRESS OF THE UNITED STATES
JOINT COMMITTEE ON ATOMIC ENERGY

February 15th, 1956

Advisory Committee on Reactor Safeguards
Atomic Energy Commission
Washington 25, D. C.

Attention: Dr. Roger McCullough, Chairman

Gentlemen:

As a member of the Joint Committee on Atomic Energy, also as a member of the Senate and as a citizen, I am, and have been, concerned about the question of safety of the public involved in the location and operation of atomic reactors. I would, therefore, like to ask the following questions, which to my knowledge have not been answered heretofore, and I shall appreciate categoric concise answers.

1. Will the operation of the Reactor presently under construction at Shippingport, Pennsylvania, create any potential dangers or the possibility of danger to the citizens in that densely populated area? (This question includes not only the possibility of hazards under so-called normal operation, but hazards or injury which could come from human error in operating the Reactor or from natural catastrophe, sabotage or enemy attack.)

2. I ask the same question about the operation of sizeable commercial type reactors so far as they might be located in any other area of average or concentrated density in the United States.

3. Is it and will it be, in the light of all present knowledge and that anticipated in the foreseeable future, more desirable, from the standpoint of safety to the public against actual or possible danger or injury, to locate such reactors in areas of practically no density of population, such as, for instance, the Arco area, rather than in areas of populations of normal to concentrated density?

Yours sincerely,

/s/ Bourke B. Hickenlooper

Bourke B. Hickenlooper

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D. C.

March 14, 1956

Dear Senator Hickenlooper:

In your letter of February 15, 1956, to the Commission's Advisory Committee on Reactor Safeguards, you asked for categoric, concise answers to three questions dealing solely with the relationship between reactor isolation and possible hazards to the public.

Regardless of location or isolation, there is no such thing as an absolutely safe nuclear reactor -- just as there is no such thing as an absolutely safe chemical plant or oil refinery. There is always present, regardless of the remoteness of its probability, a finite possibility of the occurrence of an event, or series of events, the result of which is the release of unsafe quantities of radioactive material to the surrounding area. Should such a release occur, the number of persons receiving excessive radioactive exposure would, of course, be directly related to the density and distribution of population in the area surrounding the reactor.

It follows that the answer to each of the three questions you raise is "yes". Thus, there is a remote possibility of danger to citizens in the vicinity of the Shippingport reactor or any other sizeable reactor, and, it is, therefore, more desirable from the standpoint of safety alone to locate reactors in areas of low, rather than normal, population density.

Such answers, standing by themselves can, however, be misleading and could result in misunderstanding and misconception. For this reason, I should like to expand my answers to include a discussion of some of the more important factors which we must consider in determining the location of any given reactor.

As previously stated, if considerations were given to safety factors alone, atomic reactors should be located in areas of lowest possible population density. However, the growth and development of an atomic energy industry cannot proceed under conditions of isolation which are significantly different from those which have been found to be applicable to most other industries.

While it is true that the potential danger to the public from a nuclear accident is only that of the release of radioactive materials and not that of an atomic explosion, still the maximum conceivable damage which can be caused by such an accident is far greater than that which can result from normal industrial accidents. Therefore, it is incumbent upon the new industry and the Government to make every effort to recognize

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every possible event or series of events which could result in the release of unsafe amounts of radioactive material to the surroundings and to take all steps necessary to reduce to a reasonable minimum the probability that such events will occur in a manner causing serious overexposure to the public.

The licensing provisions of the Atomic Energy Act of 1954 have made it possible for the Commission to establish a regulatory program designed to assure that these objectives will be achieved. Under our regulations no license will be issued for the operation of any reactor, regardless of size or intended use, until the scientists and engineers who conceived and designed the reactor have made a complete evaluation of all potential hazards of their particular reactor, and of the adequacy of the steps they have taken in design and operating procedures to minimize the probability of occurrence of an accident which would result in the release of unsafe quantities of radioactive materials to the surroundings. This evaluation, which is reported in a detailed "Hazard Summary Report" to the Commission, is used by the technical experts on the Commission's staff working in close collaboration with the experts of the Commission's Advisory Committee on Reactor Safeguards to determine whether or not the operation of the reactor can be carried out in a manner that gives reasonable assurance that the health and safety of the public will not be endangered.

The financial incentive of the owners of the reactor to take all steps necessary to protect their investment, as well as to decrease their potential public liability, and the legal and moral responsibilities of the Commission to protect the public from overexposure to radioactivity, are resulting in a system which is characterized by an attitude of caution and thoroughness of evaluation unique in industrial history. Every phase of the reactor design and operating procedure is reviewed separately and as a part of the whole. The inherent nuclear, chemical, metallurgical, physical and mechanical characteristics of the fuel, moderator, coolant, neutron absorbers and structural materials are carefully considered in connection with the electrical, mechanical, chemical, physical, metallurgical and nuclear characteristics of the control and safety systems, the heat removal systems, the pressure systems, and so on, to assure that the probability of an operating mishap has by adequate design and operating precautions been brought to an acceptably low level.

Not only must the evaluation show that the designers have taken all reasonable precautions to assure that the probability of a mishap is satisfactorily low, it must further show what steps have been taken to protect the public in the event the highly improbable incident did occur and unsafe quantities of radioactive materials were released from the reactor itself. It is in this evaluation of what is essentially a vital second line of defense for the public that the relationship of the characteristics of the location of the reactor to the ability of the building to contain radioactive materials which might be released becomes an important factor. It is during this phase of the study that the hydrology, meteorology, geology and seismology of the area; the existing and potential population density and distribution; the type of existing and

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potential activity in the area (i.e., agricultural, commercial, industrial, residential, etc.); the use of the surface and surface waters for industrial or personal consumption; and other factors pertinent to the specific location, are considered in order to be sure that the degree of containment is adequate for the location chosen.

If, for example, it is possible to show that under the most adverse set of circumstances which might occur, the structure of the building containing the reactor would not be expected to allow the release of any significant amount of radioactive materials into the surrounding area, such factors as the proximity of the reactor to densely populated areas would be less important than otherwise. Likewise, if the distance from densely populated areas were so great that under the most adverse conditions it would be reasonable to expect that there would be little exposure of the public, the degree of containment would not be so important.

It is expected that power reactors, such as that now under construction at Shippingport, Pennsylvania, will rely more upon the philosophy of containment than isolation as a means of protecting the public against the consequences of an improbable accident, but in each case there will be a reasonable distance between the reactor and major centers of population.

In summary, then, our safety philosophy assumes that the potential danger from an operating atomic reactor is very great and that the ultimate safety of the public is dependent upon three factors:

1. Recognizing all possible accidents which could release unsafe amounts of radioactive materials;

2. Designing and operating the reactor in such a way that the probability of such accidents is reduced to an acceptable minimum;

3. By appropriate combination of containment and isolation, protecting the public from the consequences of such an accident, should it occur.

Sincerely yours,

/s/

W. F. Libby
Acting Chairman

Honorable Bourke B. Hickenlooper
Joint Committee on Atomic Energy
Congress of the United States

It is interesting that in 1956, Senator Hickenlooper raised questions which once again were to be found on the forefront of discussion twenty years later. Chairman Libby points out that there is no absolutely safe reactor, just as there is no absolutely safe chemical plant. Libby discusses the principle of defense in depth and states "It is expected that power reactors such as that now under construction at Shippingport, Pa. will rely more upon the philosophy of containment than isolation as a means of protecting the public against the consequence of an improbable accident, but in each case there will be a reasonable distance between the reactor and major centers of population.

Libby suggests, in the previous paragraph, that under the most adverse circumstances, containment might be expected not to allow the release of any significant amount into the surrounding area. But he does not quarantee its infallibility.

In addition to the precedent setting Shippingport, Dresden, and Indian Point reactors, several other commercial and public power reactors received approval for construction in the time period up through 1960, as shown in Table 1. And several test reactors were approved.

TABLE 1. COMMERCIAL AND PUBLIC POWER REACTORS

Name	Type	Power (mwt)	Exclusion Dist.(mi.)	City Dist.	Site Approved
Shippingport	Pressurized Water	231	.4	7.5	before '55
Indian Point	Pressurized Water	585	.3	17.0	5/55
Dresden 1	Boiling Water	630	.5	14.0	7/55
Fermi 1 - PRDC	Fast Reactor	300	.75	7.5	7/55
Yankee	Pressurized Water	485	.5	21.0	7/57
Elk River	Boiling Water	58	.23	220.0	8/58
Piqua	Organic	48	.14	27.0	1/59
Carolinas-VA	D_2O; Tube type	63	.50	25.0	1/59
Hallam	Sodium Graphite	240	.25	17.0	7/59
Saxton	Pressurized Water	20	.17	20.0	9/59
Pathfinder	Boiling Water	203	.5	3.5	12/59
Big Rock	Boiling Water	240	.5	135	3/60
Humboldt Bay	Boiling Water	202	.25	3.5	3/60
Bonus	Boiling Water	50	.25	12.0	3/60
Peach Bottom	Gas-Cooled	115	.57	21.0	3/60

However, not all reactor proposals received approval. In the following, we shall examine some specific cases for such insight as may be obtained from the meeting minutes of the statutory ACRS and its letter reports to the AEC.

At its second meeting, Nov. 1-3, 1957, the ACRS wrote reports favorable to operation of the Shippingport PWR, to construction of the 30 MWt

General Electric Test Reactor*, and to construction of the 60 MWt NASA test reactor (Plum Brook), 3 miles from Sandusky, Ohio. It is of some interest to note that the Plum Brook report was the first to contain additional remarks by an ACRS member. The ACRS report itself noted the potential risks imposed by proposed experiments for defense purposes which would carry fuel elements to failures; and, while accepting the proposed site, the ACRS indicated that a less populated site would be preferable. In his additional remarks member Abel Wolman stated:

> While I agree with all that the Committee has stated, I feel that I must add some remarks for purposes of clarifying my own position. In view of the prospect of future continuing debates as to the safety of conducting essential experiments at this site, I would recommend against the site on the information presently available. I believe that the applicant should be required to consider the availability of other sites at which operation of the reactor would be feasible and which would afford a higher degree of protection to the health and safety of the public.
>
> It is unrealistic to permit operation at this site if experiments of importance to the national defense are likely to have to be curtailed because of the site. The realities of human behavior are such that operation of experiments, the hazards of which may be uncertain, are likely to be permitted if they are important to the national defense.
>
> I do not believe that we should freeze on a site in a situation like this merely because an applicant has chosen it.

At its 4th meeting, January 9-11, 1958, the ACRS reported favorably on the proposed operation of the MIT test reactor. This reactor, which was to operate at a fairly modest power, was located in the heart of an urban area in Cambridge, Mass., with essentially no exclusion area. The estimated dose to a person standing 65 feet from the building for one hour, if one postulated the "maximum credible accident", and assumed a gross release of fission products to the containment, was 28 R whole body. And the 60 day integrated internal dose to the thyroid was 1750 R for a person exposed one hour in the path of the cloud at a distance of 65 feet from the reactor. Dr. Dunham and Dr. Pack of the AEC Staff indicated that such an exposure did not represent a serious risk for an individual adult, but that if the maximum credible accident were to occur, every effort should be made to prevent such exposures to occupants of the neighborhood.

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*It is also of interest to note that twenty years later, in October, 1977 the NRC Regulatory Staff halted operation of the General Electric Test Reactor because recent studies had indicated the existence of a capable fault (a fault defined to be capable of producing permanent, relative displacement at the ground surface) at the reactor site.

At its 9th meeting, August 4-5, 1958, the ACRS reviewed the proposed construction of the 58 MWt Elk River boiling water reactor. In its report the Committee noted that, as a matter of policy, they considered it not desirable to locate a nuclear reactor of this power level so close to a growing community, and that, in the event of a major accident, a few people might be exposed to higher radiation dosage that was considered acceptable. (A dose of 4800 rem to the thyroid is discussed in the meeting minutes). The ACRS concluded, however, that, subject to the containment meeting its specified leak rate, the reactor would not represent an undue hazard to the public.

At the same meeting the ACRS heard a presentation concerning a proposed organic-cooled power reactor for the city of Piqua, Ohio. The reactor thermal power was only 45 megawatts. However, the site was just outside the city limits and only a few hundred feet from a temporary residential area. In its report of August 5, 1958, the ACRS concluded tentatively that the proposed site was not a suitable one.

The minutes of the 10th meeting, October 15-17, 1958 report considerable discussion about site selection criteria. It was noted that the probability of the maximum credible accident had not been analyzed or discussed in detail. The question was raised whether radiation doses like 4800 R to the thyroid might be acceptable if a very limited number of people (say 25) were involved. And there was a search for analogies in other aspects of industrial safety, such as release of noxious chemicals. At this meeting the ACRS passed a resolution which would require an exclusion radius of a quarter of a mile or more for reactors of power equal to 10 thermal megawatts or greater.

The minutes of the 10th meeting also say that the ACRS Chairman "reported a meeting with Mr. McCone, the new Chairman of the AEC, who stated that he considered the ACRS over conservative and contributing to the slowdown of nuclear power development due to financial and safety restrictions."

At the 10th meeting there was also further discussion of the proposed Piqua reactor. The reactor designer, Atomics International, stated they had not found an accident which could release significant quantities of radioactive materials and was not proposing containment. The AEC Regulatory Staff, on the other hand, said experience to date was inadequate to support this position, even if it might be true. The Regulatory Staff felt that containment was mandatory in a moderately populated region. Dr. Pittman, Acting Director of Reactor Development said that, in the future, sites for AEC-supported reactors would receive early review to help preclude situations like Piqua.

At its 11th and 12th meetings the ACRS continued discussions of the Piqua reactor. A new site was proposed and an unconventional form of containment was proposed. The third Piqua site was stated to have 80 people within 1/4 of a mile, 450 within 1/2 and about 7,000 within 1 mile of the reactor. The first site had 485 within 1/4 mile, 3,000 within 1/2 mile, 8,000 within 1 mile, and 20,000 within 5 miles. By way of comparison, Elk River was stated to have no people within 1/4 mile, 60 within 1/2 mile and 700 within 1 mile. At Shippingport, the PWR had 20,000 within 5 miles and 130,000 within 10 miles. Indian Point had no people within 1/4 mile, about 50 within 1/2 mile, 2,100 within 1 mile and 45,000 within 5 miles.

The ACRS wrote a report in which it stated that the newer site was more suitable than the original one. However, "the Committee does not consider the installation at this site of a nuclear power plant of this capacity of a relatively untried type to be without undue public hazard until the proposed unconventional type of containment is replaced by a more substantial and dependable system."

As a point of interest, during many of the first dozen ACRS meetings of the statutory ACRS, there were reports of incidents and accidents which were occurring at reactors. One example was the Windscale reactor accident (Windscale No. 1). Also during this period the ACRS was reviewing the safety of the rather high power production reactors, and was trying to give advice to the AEC concerning a more acceptable future mode of operation for these reactors, which were built during World War II and which posed some major safety questions.

During the 12th meeting, December 11-12, 1958, the ACRS submitted a report to the AEC responding to a request from the Commissioners for a comparison of the standards applied in evaluating the sites for the Elk River reactor, the Vallecitos BWR, the Sodium Reactor Experiment (SRE), and the Shippingport PWR. The complete letter is on the following page.

At a meeting of the ACRS Subcommittee on Site Selection Criteria, November 26, 1958, ACRS member Conner pointed out that to expect significant consistency among approved sites would be unwarranted, since the technique used was to ask the ACRS to review a proposal and comment on the safety of that proposal. It would be unwise for the ACRS to assert in any given case that too large an exclusion area had been selected.

Dr. Conner went on to say that the real guide should be that almost everybody in the vicinity should have a reasonable chance of escaping serious injury in the event of a reactor accident. To assume that this is the case, one must, aside from estimates of possible accidents, be provided with radiation dosage criteria which represent acceptabe emergency doses below the threshold of serious injury.

At the 13th ACRS meeting, January 8-10, 1959, a more conventional containment was proposed for the Piqua reactor, and the Committee wrote a report favorable to the new site.

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December 15, 1958

Honorable John A. McCone
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: COMPARISON OF SAFETY FEATURES OF:
GE-Vallecitos Boiling Water Reactor (VBWR)
Sodium Graphite Reactor Experiment (SRE)
Shippingport Pressurized Water Reactor (PWR)
Rural Cooperative Power Association Reactor, Elk River, Minnesota

Dear Mr. McCone:

The following is in reply to your request for comparison of the standards applied in evaluating the Elk River site with those that were applied to the VBWR, SRE, and PWR sites. Pertinent data tabulations are attached. These were furnished by the Hazards Evaluation Branch.

It need scarcely be emphasized that the question of site evaluation is complex. A large number of variable factors, many not strictly comparable from site to site, must be considered. Exact, completely objective, numerical site criteria are difficult to formulate, however convenient and desirable these might be. But the Committee attempts to bring a consistent philosophy to the reactor hazards problem and to provide a common basis for site judgments.

Three distinct types of reactor are involved in the group in question. These are of the sodium graphite, pressurized water, and boiling water types.

SRE, a low power (5 Mw thermal) reactor of the sodium graphite type, operates at atmospheric pressure in an underground location. The primary coolant is contained in a stainless steel shell which is in turn contained in a sealed concrete structure. Secondary coolant from primary heat exchangers located within the containment structures gives up its heat in external steam boilers. A rupture of the primary system will not cause melting of fuel or release of fission products

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therefrom. For these reasons, and because the SRE is located in a relatively large exclusion area (1.4 miles minimum radius), immediately surrounded by a sparsely populated district no containment vessel of the type used for pressurized reactors if employed.

The PWR reactor of the pressurized water type is provided with an exclusion distance of approximately 0.5 miles. It is fully contained and provided with biological shielding of the containment structures. It is designed to contain the vapor and energy released in the event of a rupture of the primary water system and one steam generator. In addition, the interconnected containment vessels are designed to contain the energy resulting from significant metal-water reactions.

The VBWR and Elk River reactors are of the boiling water type. In the VBWR, the coolant is vaporized and is used for the direct drive of turbo-generators. In the Elk River reactor, radioactive steam is taken to a heat exchanger, providing a barrier. Both are provided with containers designed to prevent release of vapors resulting from a break in the cooling system. The VBWR has been designed to contain the results of a metal water reaction. The Elk River reactor containment vessel is provided with significant missile and biological shielding.

In attemtping to decide for a particular reactor whether a given exclusion distance provides adequate protection for public safety, the Committee evaluates design features such as containment vessels, missile shields, biological shields, hydrology, meteorology, and geology, all of which affect reactor safety, particularly when a reactor is located near a populous area. Thus it was felt that the Elk River site would provide an acceptable degree of protection to the public, in view of the isolated primary system and the vapor containment provided. Like considerations were applied in the case of the PWR reactor. The SRE has somewhat less containment, but has a greater exclusion radius than the others mentioned.

Population density is of concern to the Committee. Consequently, the relatively low population density nearby the Elk River site was considered to be a generally favorable element. On the other hand potential growth of the community needs also to be taken into account. The extent of this growth at Elk River is problematical; but the Committee felt it appropriate to express its consideration for the eventuality and to recognize dependence on engineering features for minimizing risk.

The fact that a highway and a railroad run comparatively near the Elk River site is not considered to increase the risk significantly, and conversely acceptance of these features does not imply a reduction in standards of population protection. Since highway and railroad occupancy is transient and intermittent, both the probability and the intensity of the risk are greatly reduced over those applying to stationary, permanent populations at the same distance. Moreover, access to a highway or railroad can be restricted in the event of an accident.

Sincerely yours,

/s/
C. Rogers McCullough
Chairman

cc: Alvin R. Luedecke, GM
Harold L. Price, DLR

Attachment
a/s

Dist: Orig & 4 copies to Mr. McCone
1 copy to Gen Luedecke
1 " " Harold L. Price
1 " " H. H. Plaine (blind)

During a discussion on site criteria, Dr. Beck of the AEC Staff stated that there were three primary problems:

(1) Determination of adequate exclusion distances;

(2) The population distribution outside of the exclusion radius, and the proximity of the site to large centers of population; and

(3) The characteristics of the reactor and its effect on 1) and 2) above.

ACRS member Newson presented an independent approach in which site undesirability was proportional to an integration over reactor power and fission product escape fraction, population density, and probability of fission product escape. ACRS member Conner suggested that a numerical evaluation would not occur in their generation.

In the continuing discussion on site criteria at the 14th meeting, March 12-14, 1959, ACRS member (and then Chairman) McCullough hypothesized an approach in which a total societal risk in terms of a total damage dose (20,000 Roentgens/year) would be accepted for the nuclear industry, and a portion of this total acceptable dose would be allocated to each reactor. It was estimated (in 1959) that this might mean 20 statistical deaths per year attributable to the nuclear industry, whereas the electric industry was estimated as experiencing a fatal accident rate of 150/year.

At the same meeting, Dr. Beck advanced the proposal of the Regulatory Staff that the maximum credible accident not produce more than 25 Roentgens whole body radiation at the site boundary. The Staff also proposed a minimum exclusion radius of 1/4 mile for reactors less than 100 MWt. Dr. Forest Western of the AEC Division of Biology and Medicine advised against the inclusion of any arbitrary radiation levels in any public regulations. Dr. Western also stated that if an individual had accumulated twenty Roentgens, they would not propose evacuation, even with a probable additional dose of fifteen Roentgens, due to disruption of family and business activities.

At the 15th ACRS meeting, April 16-18, 1959, it was reported that at a recent symposium, members of the AEC Biology Division indicated that no threshold existed for biological damage from radiation, which tended to confirm previous ACRS interests in limiting integrated population dose from accidents (or routine emissions).

In a memorandum of April 23, 1959 from McCullough to H. L. Price, Director, Division of Licensing and Regulation, the ACRS provided a suggested regulation for the AEC, the impact of which would have been to require formal site review as the first step in requiring a reactor license.

On May 23, 1959 the AEC published in the Federal Register a notice of proposed rule making concerning site criteria for comment. The comments, which were received largely from the nuclear industry, were highly unfavorable.

The 1959 draft and some of the comments made directly to the AEC or in meetings of the Atomic Industrial Forum (AIF) or Edison Electric Institute (EEI) are reproduced below:

Proposed Rule Making

"Factors consideredin site evaluation for power and test reactors:

a. General. The construction of a proposed power or test reactor facility at a proposed site will be approved if analysis of the site in relation to the hazards associated with the facility gives reasonable assurance that the potential radioactive affluents therefrom, as a result of normal operation or the occurrence of any credible accident, will not create undue hazard to the health and safety of the public.

b. Exclusion distance around power and test reactors. Each power and test reactor should be surrounded by an exclusion area under the complete control of the licensee. The size of this exclusion area will depend upon many factors including among other things reactor power level, design features and containment, and site characteristics. The power level of the reactor alone does not determine the size of the exclusion area. For any power or test reactor, a minimum radius on the order of one-quarter mile will usually be found necessary. For large power reactors a minimum exclusion radius on the order of one-half to three-quarter miles may be required. Test reactors may require a larger exclusion area than power reactors of the same power.

c. Population density in surrounding areas. Power and test reactors should be so located that the population density in surrounding areas, outside the exclusion zone, is small. It is usually desirable that the reactor should be several miles distant from the nearest town or city and for large reactors a distance of 10 to 20 miles from large cities. Where there is a prevailing wind direction it is usually desirable to avoid locating a power or test reactor within several miles upwind from centers of population. Nearness of the reactor to air fields, arterial highways and factories is discouraged.

d. Meteorological consideration. The site meteorology is important in evaluating the degree of vulnerability of surrounding areas to the release of air-borne radioactivity to the environment. Capabilities of the atmosphere for diffusion and dispersion of air-borne release are considered in assessing the vulnerability to risk of the area surrounding the site. Thus a high probability of good diffusion conditions and a wind direction away from vulnerability areas during periods of slow diffusion would enhance the suitability of the site. If the site is in a region noted for hurricanes or tornadoes, the design of the facility must include safeguards which would prevent significant radioactivity releases should these events occur.

e. Seismological considerations. The earthquake history of the area in which the reactor is to be located is important. The magnitude and frequency of seismic disturbances to be expected determine the specifications which must be met in design and construction of the facility and its protective components. A site should not be located on a fault.

f. Hydrology and geology. The hydrology and geology of a site should be favorable for the management of the liquid and solid effluents, (including possible leaks from the process equipment). Deposits of relatively impermeable soils over ground water courses are desirable because they offer varying degrees of protection to the ground waters depending on the depth of the soils, their permeability, and their capacities for removing and retaining the noxious components of the effluents. The hydrology of the ground waters is important in assessing the effect that travel time may have on the contaminants which might accidentally reach them to the point of their nearest usage. Site drainage and surface water hydrology is important in determining the vulnerability of surface water courses to radioactive contamination. The characteristics and usage of the water courses indicate the degree of risk involved and determine safety precautions that must be observed at the facility in effluent control and management. The hydrology of the surface water course and its physical, chemical and biological characteristics are important factors in evaluating the degree of risk involved.

g. Interrelation of factors. All of the factors described in paragraph b through f of the section are interrelated and dictate in varying degrees the engineered protective devices for the particular nuclear facility under consideration, and dependence which can be placed on such devices. It is necessary to analyze each of the environmental factors to ascertain the character of protection it might afford for operation of the proposed facility and of the kind of restrictions it might impose on the proposed design and operation.

Dated at Germantown, Md., this 19th day of May 1959.

A. R. Luedecke
General Manager

SOME REACTIONS TO THE AEC'S PROPOSED REACTOR SITE RULE-MAKING

1. Philip Sporn, President, American Electric Power Service Corp., and Chairman of the AIF meeting, stated: "Any standard set up today, no matter how unreasonable and unnecessarily broad and supersafe, is going to be hard to re-do in the years to come.

"Whatever finally comes out in lieu of this particular rule should be clearly marked as not being a rule or regulation. It should be broad and not get into cold statements such as setting distances from large cities. Regulations will be millstones around the neck of an industry which is just starting. This particular set of rules should be suspended in the interim. It has already been a real service by bringing out the things it was designed to do."

2. Louis H. Roddis, Jr. President, Pennsylvania Electric Co., told the AEF meeting: "Anything as definitive as the issuance of a formal rule of the Commission is going to pose to the industry a problem. We do need a statement of what is needed in order to arrive at a balance of all these different factors, but it should not be formal."

3. Titus LeClair, Manager of Research and Development, Commonwealth Edison Co., told the AIF: "Is Dresden a large power reactor? It is today, but it is pretty small when compared to a plant of 500 megawatt capacity. We don't know what is large or small. These words in a regulation lead to considerable problems."

4. R. M. Casper, General Manager, Atomic Energy Division, Allis-Chalmers Manufacturing Co., wrote to the AEC: "...we feel strongly that it is too early to state quantitative rules which may be subject to misinterpretation by members of the general public. The wide difference between reactors, types of containment, etc., makes it particularly difficult to establish numerical rules, and we believe there will be a tendency to regard quantitative criteria as minimal safety requirements.

"...We believe it would be most helpful if the Commission would issue a policy statement on site evaluation, outlining the information necessary and indicating when it will be required with respect to the project schedule."

5. James F. Fairman, Senior Vice President of Consolidated Edison and Chairman of the Technical Appraisal Task Force on Nuclear Power of EEI told the AEF: "I would much prefer it if the AEC would come up with a general statement of principles or guides, or with a list of things which need to be done, rather than a set of rules. Rules with set numbers could be too restrictive, and hold back progress. The objectives of the operators of nuclear power plants are no different from those of government: we want reactors which are safe, and we don't want to be tied down to figures which may quickly become outdated."

. Francis K. McCune, Vice President, General Electric Co., said at the AIF meeting: "If you don't put numbers down, you get into real trouble. There will not be a nuclear industry until things like this are done. There is a way to say this - the exclusion area should be large enough to provide for one, two or three specific requirements."

7. Chauncey Starr, Vice President of North American Aviation Inc. and General Manager of the Atomics International Division, wrote to the AEC: "...we feel that to proceed with the hurried enactment of regulations such as the proposed rule could effectively smother the infant industry... Until such time as a logical, long-term power plant site plan is developed which does not hinder the industry's growth, the pattern of reviewing and evaluating each proposed reactor and its site on an individual basis whould be continued."

8. Leonard F. C. Reichle, Nuclear Engineering Director, Ebasco Services, Inc.: "The proposed rules emphasize only the characteristics of the site and environs. They virtually ignore the other two aspects which determine suitability, namely, the characteristics of the facility itself, including the state of knowledge and past experience, and the safeguard features which are incorporated in the facility. It is probably true that, with sufficient knowledge of the potential hazards, any facility can be designed with appropriate safeguards to permit operation anywhere with acceptable risk.

"As a rule, the prospective licensee selects a site for economic reasons and balances the cost of safeguard provisions against the added cost and inconvenience of a more isolated site. The AEC must similarly evaluate all factors to determine whether the overall hazard is acceptable. Because of the complex interplay of the many factors concerned, it is probably not practical to expect definitive standards. Some guide to the important factors considered by AEC and, if possible, the probably relative weights to be applied would be welcomed by industry."

9. Jack K. Busby, President of Pennsylvania Power & Light Co., wrote to the AEC: "We believe it most desireable that the Commission formulate and publish general site selection guides but, in our opinion, it is undesirable to designate minimum exclusion distances around power and test reactors, minimum distances of such reactors to the nearest town and city, and maximum offsite population densities. The problem is to establish reasonable assurance that there will be no hazards to the public ... We suggest that all minimum distances and maximum population densities be eliminated from the proposed regulation and that such factors be given consideration only in relation to the proposed type, design and safeguards of the particular reactor."

10. R. D. Welch. Florida West Coast Nuclear Group, wrote to AEC: " ... it would be better to avoid using distance measurements such as 1/2, 3/4 miles exclusion radii and 10-20 miles from cities for large power reactors. Such distances tend to become fixed in the public mind despite words of flexibility used in connection with them.

"The proposed regulation does not indicate that improvement in reactor design and safety experience may reduce the distances mentioned."

11. Robert L. Wells, of Westinghouse, quoted above, wrote to the AEC: "... we are quite concerned about the proposed rules pertaining to required exclusion areas. The safety of the public is a function of many factors, of which exclusion area is only one. Specifically, we feel that the safety of the public can best be maintained by proper engineering design consideration of all the important variables including reactor size and type, core safety cooling system, possibilities of release of fission products from the reactor system and subsequent leakage from the vapor container, exclusion area and meteorological conditions, to name but a few.

"The safety of the public cannot be insured by any single condition such as exclusion area, but rather is the certain result of the optimum combination of many interrelated factors. To specify minimum exclusion area is neither necessary nor sufficient."

12. Philip Sporn, of American Electric Power Service Corp., quoted above, remarked at the AIF committee meeting: "The British and the French keep their power reactor sites away from centers of population and we are trying to build on the periphery or on the very outside boundaries of our cities and towns. But the aim should be to come as close as possible to the heart of cities. Of course, a power reactor quite close to, or in, a city may require expensive additional safety structures as opposed to one in a wide exclusion are."

13. James F. Fairman, of Consolidated Edison and Chairman of the Technical Appraisal Task Force on Nuclear Power of EEI, told the EEI:

"Indian Point, which is 24 miles north of New York City and on the east bank of the Hudson, was the most remote location we could find in our operating area. It is not only extremely difficult to acquire power plant sites within the area of New York City and Westchester County, but also expensive.

"In the long term Con Ed will want to put nuclear power plants as close to its load centers as possible, which means, of course, right in the city limits. The setting of any arbitrary exclusion area limits would place a high cost premium on power plants in metropolitan areas and discourage the use of engineering ingenuity to find the most practical solutions to safety problems in built-up areas.

"Engineering design measures can meet safety requirements at a cost, for example, in the case of the Indian Point plant there was the problem of 'sky shine' if the containment sphere were filled with contaminated gases as a result of an 'incident.' This problem was solved by building an exterior biological shield to prevent atmospheric reflection of radiation emanating from the top of the containment vessel down on the surrounding area.

"Con Ed intends to build another facility adjacent to the Indian Point plant and wants to avoid the necessity of having to evacuate personnel from the site in the event of a nuclear incident. We believe an atomic power station can be designed safe enough to be located in a heavy populated area although such a design would increase the cost."

14. C. T. Cheve, Chief Engineer, Nuclear Projects, Stone & Webster Engineering Corp., wrote to AEC: "We agree, in general, with the idea of making these rules, since there has been some chaos because of the lack of them. The only matter we see which might cause a serious hardship is covered in Paragraph (c) in which it is suggested that large reactors should be 10-20 miles from large cities. This may give a sense of security, but the point requires perhaps a little more careful review, because the economics of nuclear power are going to be adversely affected by such a rule. One of the advantages of nuclear power plants which might overcome somewhat higher generating costs than obtained from combustible fuel-fired plants is that the nuclear plant might be located closer to load centers because of its lack of dependence on railroad shipment of coal."

15. Richard H. Peterson, of Pacific Gas and Electric Co., quoted above, to the AEC: "With regard to seismological considerations the proposal provides that a site should not be located on a fault. In the West Coast area, where earthquakes are more common than in other parts of the country, the strict application of this proposal to an entire site area could eliminate many desirable locations. Structures can be built adjacent or near to earthquake faults to withstand severe shocks without failure. In California we know of no structure which has been severely damaged by an earthquake for which the designer and builder took earthquake forces into consideration. For these reasons if a prohibition against location on a fault be included at all, we suggest that it be limited to location of the reactor and auxiliaries."

16. Patrick J. Selak, Manager, Nuclear Engineering Development Kaiser Engineers, proposed to the AEC: "Rather than establish a minimum distance from a 'large city', perhaps a better criterion would be to establish a maximum number of people who might receive an overexposure in the event of a 'maximum credible accident.' Then the reactor builder could determine, subject to AEC approval, the optimum combination of exclusion zone, distance from populated areas, containment features, and inherent safety features in the reactor-- which would provide adequate safety to the public at minimum costs."

The proposed rule, which was quite general, included the concept of credible accident." A letter dated May 11, 1959 from H. L. Price, Director, Division of Licensing and Regulation, to Chairman McCullough of the ACRS

The proposed rule, which was quite general, included the concept of "credible accident." A letter dated May 11, 1959 from H. L. Price, Director, Division of Licensing and Regulation, to Chairman McCullough of the ACRS states that these site criteria reflect the discussions between the Regulatory Staff and the ACRS at the 13th and 14th meetings; however, it is not clear from the available files to what extent the proposed rule reflected ACRS opinion, and at least one ACRS member commented adversely to the AEC. It is also not clear who originated the idea of "credible" or maximum credible accident.

However, in June, 1959 Dr. Clifford Beck of the Regulatory Staff gave a paper entitled "Safety Factors to be considered in Reactor Siting" at a nuclear congress in Rome, in which he discussed credible accidents (and other things) as follows.

> "It is well established by many studies that release to the atmosphere of the radioactivity inventory of any large reactor could cause death or injury to thousands of people over distances of many miles. Evaluation of a site for such reactors cannot be made, except for sites widely removed from populated areas, unless information is available on the radioactivity inventory of the reactor and the likelihood of various amounts of this inventory being released from the facility. Thus evaluation of the suitability of sites for a reactor leads at once to an analysis of the reactor proposed for the site and to an evaluation of radioactivity effluents normally expected from the facility and of potential accidents which might cause the unintended release of radioactivity to the environment.
>
> "Routine radioactive effluents to the environment must be restricted to such levels that potential radiation exposures would be within established limits for more or less continuous exposure. When the plant is operating it can be easily established whether or not permissible levels are being exceeded and if required it should be possible to make facility or procedural alterations which would bring the activity levels in normal plant effluents to the desired value. Thus, advance judgments of the adequacy of a site-facility combination with respect to normally expected effluents can be verified by direct observation after operation is underway.
>
> "It is an entirely different matter to evaluate the adequacy of a site-reactor combination for the accidental release of radioactivity which potentially could occur. Here a difficult dilemma is encountered. If the worst conceivable accidents are considered no site except one

removed from populated areas by hundreds of miles would offer sufficient protection. On the other hand, if safeguards are included in the facility design against all possible accidents having unacceptable consequences, then it could be argued that any site, however crowded, would be satisfactory... assuming of course that the safeguards would not fail and some dangerous potential accidents had not been overlooked. In practice a compromise position between these two extremes is taken. Sufficient reliance is placed on the protective features to remove most of the concern about the worst conceivable accidents, though there is seldomly sufficient confidence in the facility safeguards to be sure that all hazards have been eliminated. Thus, a possible reactor site is reviewed against the possibility of credible accidents, and their consequences, which might occur despite the safeguards present.

"it is inherently impossible to give an objective definition or specification for 'credible accidents' and thus the attempt to identify these for a given reactor entails some sense of futility and frustration, and, further, it is never entirely assured that all potential accidents have been examined.

"It should be noted parenthetically, however, that this systematic search for credible accidents often contributes substantially to the safety of a facility. Potential accidents having substantial consequences and clearly credible possibility of occurrence may be discovered in this search. If such are found, safeguards against them of course are incorporated ... and the evaluation then proceeds for the potential accidents remaining. In the plants finally approved for operation, there are no really credible potential accidents remaining against which safeguards have not been provided to such extent that the calculated consequences to the public would be unacceptable.

"In general, accidents would be considered credible if their occurrence might be caused by one single equipment failure or operational error, though clearly some consideration must be given to the likelihood of this failure or error. It has been suggested that this criterion might be extended to the assignment of decreasing probabilities to accidents which would be occasioned only by 2, 3 or more independent and simultaneous errors or malfuctions, with the possibility that accidents requiring more than 3 or 4 such independent faults would be considered incredible. In practice, this suggestion has not been found useful, largely because of the difficulty of isolating and identifying the independent factors which have contributed to an accident, or more to the point, of deciding in advance what probability can be assigned to various combinations of errors or malfuctions which are worthy of consideration.

CREDIBLE ACCIDENTS

In the past, it has appeared reasonable to expect that accidents could occur which would result in release of a a portion of the fission product inventory at least from the fuel elements into the coolant stream, if not from the succeeding containment barriers.

The likelihood of fuel element failures by overheating or cladding defects is usually placed quite high.

The interruption of primary coolant circulation is a credible possibility in almost every large reactor.

In all high power density, long fuel-burnup reactors, loss of coolant circulation probably and loss of coolant certainly would lead to fuel meltdown by decay heat, even though the reactor had been scrammed immediately. Hence, there are always provided emergency cooling systems, sometimes as many as three or more independent systems.

If the possibility of fuel meltdown cannot be completely excluded, and if metals such as aluminum and zirconium in contact with water are present, then the possibility of a violent water-metal chemical reaction cannot be excluded. With uranium oxide fuel and stainless steel cladding, the likelihood of such a reaction is not considered credible. An if water is replaced by organic or sodium as coolant, the concern with liquid-metal reactions is removed.

On the other hand, when organic liquid or molten sodium are used, the possibility of their becoming involved in serious chemical reactions with ambient atmospheric components or other materials cannot be excluded.

Thus in every reactor, accidents having potentially serious consequences can be identified. Against these, safeguard features and precautionary measures, to the extent considered necessary for the serious consequences to be avoided, are incorporated. Despite these precautions, however, there always remains an uneasiness that these safety devices will not operate as expected or that something has been overlooked. After all, our cumulated reactor experience is quite small and, more significantly, our experience with any one type of reactor built to a standardized design, is a great deal smaller. Hardly any two reactor facilities are alike and the carry-over of detailed safety analysis from a component of one reactor to an approximately similar component in another is often surprisingly small. Finally, the consequences of a major accidental release of fission products are so great that the degree of confidence in the safeguards must far exceed that required in ordinary industrial processes.

"These factors have led in the United States to the widespread use of external 'vapor' containment vessels for power and test reactors built near populated areas. This practice originated in part from the characteristics of the earlier types of reactors which were built: pressurized water reactors

> with large excess reactivities and metallic fuel elements having aluminum or zirconium cladding. But the initiation of this practice and its extension to other types of reactors also reflects a special degree of concern for the protection of the public from the hazards of potentially dangerous devices which were not very well understood or extensively proven.
>
> "The external containment vessel, as a barrier of last resort against releases of radioactivity to public areas, offers a unique protection, completely independent of all other safety devices and engineering safeguards and its dependability is unaffected by errors in safety analyses and judgment of the reactor assembly. It stands as a visually obvious and intuitively attractive bulwark against the possible consequences of errors in reactor design, malfunction and misoperation which are admittedly present in any human undertaking."

At the 17th ACRS meeting, July 23-25, 1959, there was the beginning of what was to be a very long and diffcult review process concerning problems associated with proposals to bring the nuclear merchant ship Savannah into highly populated port areas. We shall not go into this topic because of its unique nature. However, one can summarize (too briefly) by saying there was very considerable opposition to bringing the Savannah into the heart of New York City or equivalent ports and depending on its engineered safety features to protect against the "maximum credible accident."

At the 19th meeting, September 10-12, 1959, the ACRS put into writing regarding the Pathfinder reactor and the Carolinas-Virginia Tube Reactor (CVTR) a position it had taken on several previous cases, namely that it lacked sufficient information with regard to certain design features to arrive at a conclusion concerning construction of the plants.

The ACRS held a special meeting on March 5, 1960 to consider a request by A. R. Luedecke, General Manager of the AEC, for advice concerning the possibility of siting some relatively large LWRs (1000 MWt) in California. Major excerpts follow from the ACRS letter of March 6, 1960 to AEC Chairman McCone:

> "In reply to the request for an advisory report from the Advisory Committee on Reactor Safeguards on the feasibility and acceptability of locating the proposed reactors in the Los Angeles area and in an area within a fifty-mile radius of San Francisco in terms of the possible hazards associated with inversion and earthquake conditions, the following advice is given.

With respect to seismic considerations, we understand that it is present utility industry practice in California to locate generating stations at least one mile from known surface faults; and to design and construct these stations using local codes supplemented by special analyses and increased seismic design factors for those critical plant components necessary to maintain the station on the line. In addition, in the case of a nuclear reactor facility, special analyses and increased seismic design factors are needed for those reactor plant systems whose failure could result in a release of radioactive material. With these precautions, the Committee believes the reactor facility would be adequately protected against seismic disturbance.

With respect to the question, raised on page 2 of the subject letter, concerning the specific consideration given to the inversion question in connection with various reactor projects, inversion frequency information is invariably included in hazards summary reports by the applicant and considered by the ACRS in addition to other pertinent factors affecting site selection and safety. The attached appendix is a tabulation of inversion frequencies for a number of sites, culled from these reports and United States Weather Bureau sources.

Referring to the frequency of inversion conditions, the situation of the Southern California coastal strip (south of San Francisco) is essentially unique in the United States. The semipermanent Pacific high pressure area induces a slow, large-scale, persistent subsiding motion in the atmosphere there. Air, warmed by this descent, contacts the coastal water surface which is cold as a result of upwelling. By this mechanism an inversion is formed; and the air layer extending up to a few thousand feet above the surface becomes a trap for air pollution.

Whereas persistent poor dispersion (stagnation) conditions of meteorology, lasting several days, may be expected on the average once per year anywhere east fo the Rockies, the frequency of such episodes in the Southern California coastal strip is of the order of several per month. For example, during a two-year period, from July 1956 through June 1958, the Los Angeles weather was of the "smog warning" type 164 days.

For the Southern California Edison Electric Company reactor, the three locations given in the letter of February 27, 1960, cover a very considerable area. These locations have meteorological conditions varying from those approaching the area east of the Rockies to those characteristic of the Southern California coastal strip. On the basis of rather meager information it appears that the reactors proposed in the letter can be so designed and constructed that suitable sites can be found within the locations given. A specific reactor and its site should be given a detailed review at the earliest opportunity.

In selecting a site for a high power reactor, consideration should be given to an adequate exclusion radius and the population density, not only in the immediate vicinity, five to ten miles, but also for greater distances. Obviously, the lower the population density, the better. The meteorology of the Southern California coastal strip is so unfavorable for dissipating pollutants that this area should be avoided if it is coupled with a high population density. In theory a reactor can be designed, constructed, and operated that it will offset the unfavorable meteorology and high population density. Because of the present limited experience with the operation of power reactors and the large power level of the proposed reactors, the provision of an adequate degree of safety in practice may require an extreme of conservative design and containment."

Not surprisingly, the record of discussion within the ACRS itself indicates divided opinion as to how satisfactory the Southern California coastal area was for reactors of appreciable power. Some members believed that very good contaimnemt, together with waste retention such that routine releases would occur only under ideal weather conditions, would eliminate restrictions due to unfavorable meteorology. However, this did not appear to be the consensus of the Committee, which felt that meteorology was a principal environmental consideration for a reactor accident in Southern California.

Soon after, at the 24th meeting, March 10-12, 1960, the ACRS reviewed the proposed 40 MWt reactor at Point Loma (San Diego) California. It appears from the minutes of this meeting and of the 23rd ACRS meeting that Dr. Beck of the Regulatory Staff did not feel that the Point Loma site needed to be rejectd, although he conceded that it was not a very good site. Beck appeared to feel that the unfavorable meteorology and the unfavorable hydrology (which related to the limited rate of ocean flow to remove routine radioactivity releases) could be dealt with by appropriate containment. However, the ACRS wrote a report unfavorable to the proposed reactor, stating:

"The Committee considers Point Loma to be a poor site because of unfavorable meteorology and high population density, aggravated by recreational and fisheries aspects, and lack of ocean dilution. The close proximity of the San Cabrillo Monument area with its numerous visitors and its proposed enlargement with the probability of an increased number of visitors add to the unfavorable features. The experimental nature of the proposed installation contributes to our lack of assurance.

The Committee believes it would be unwise at the present time from the safety point of view to locate this reactor at this site."

At the same meeting the ACRS also wrote a report unfavorable to the proposed 60 MWt Jamestown, New York reactor. In this case the AEC Regulatory Staff took the same position as the Committee. The ACRS report is on the following page.

At its next meeting, the 25th, held May 5-7, 1960, the ACRS once again reviewed the same Point Loma reactor, but this time to be located at another site in California, namely Point Mugu. This time the ACRS wrote:

> "The proposed site is subject to the generally limited atmospheric dilution regimes characteristic of the Southern California coastal strip. However, this site appears to have afforded the degree of atmospheric dilution comparable with the most favorable that can be found in this coastal area. Because of the isolation afforded as a result of the site location relative to terrain features and Government-owned land, the Committee believes that the indicated site is acceptable for an appropriately designed and operated reactor having the proposed general characteristics."

There was renewed discussion of the Jamestown reactor at the Special ACRS Meeting, June 7, 1960 and at the 26th meeting, June 22-24, 1960. The applicant advised that a larger site could be provided, and that 0.1% per day containment leakage rate would be the new design specification. In a letter to Chairman McCone dated June 30, 1960, the ACRS once again advised against the small PWR at the Jamestown, New York site. In a rather strong policy type statement, the ACRS stated "The Committee deplores the tendency on the part of some of those proposing reactor sites to place power reactors containing large quantities of stored energy in or near centers of population at this time to duplicate conditions for conventional power plants for the sake of demonstrating how near a population center such a reactor can be located. We believe that the Jamestown reactor is a case of this kind. We wish to point out that the proximity to a population center would require more rigid specifications of all safety features including containment, leakage rate, power densities, ultimate power, shielding, etc." The ACRS went on to say, "The Committee can find no serious technical fault with the reactor, the containment, and the safety features proposed, insofar as the partial information supplied to date has presented the case. The Committee emphasizes, however, that power reactors are relatively new and untried, and that there exists a considerable degree of uncertainty in our knowledge of their longterm safe behavior. Accordingly, the Committee doubts that the new and relatively untried technical features for improved safety proposed by the applicant, since our last report, are a satisfactory substitute for the inherent safety implied by a greater distance from population centers."

In passing, it is noted that at the 26th meeting, the ACRS wrote a letter favorable to the construction of the Humboldt Bay Boiling Water Reactor after a protracted review in which the Committee insisted on having a full

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C.

March 14, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: SMALL SIZE PRESSURIZED WATER REACTOR, JAMESTOWN, NEW YORK, SITE

Dear Mr. McCone:

At its twenty-fourth meeting, March 10-12, 1960, the Advisory Committee on Reactor Safeguards considered the site proposed for a Small Size Pressurized Water Reactor to be located in the City of Jamestown, New York. The data furnished in the Site Report (referenced below) provided only general information on the reactor which is in the conceptual stage. In addition to the site report, the ACRS had the benefit of comments from the AEC Staff and others as well as a visit to the site by a Subcommittee.

This 60 MW (thermal) pressurized light water moderated reactor is to be built and operated by the Commission on a site furnished by the City of Jamestown, New York, which will also provide the generating plant. The proposed site comprises thirty-five acres of city owned land located in the northwest corner of the city approximately 1.75 miles from the center.

The ACRS believes that such factors as the small size of the site; proximity to the City of Jamestown with its high population density; unfavorable meteorology; lack of control by the City of Jamestown over the area contiguous to the south and west boundaries of the site, which is located within the limits of the town of Celeron; and the long periods of low flow in the Chadakoin River with consequent adverse effects on liquid waste disposal all indicate that this site is not suitable for a power reactor of this size in the present stage of technology.

Sincerely yours,

/s/

Leslie Silverman
Chairman

cc: A.R.Luedecke, GM
W.F.Finan,OGM
H.L.Price, DL&R

scale test section experiment performed on the proposed suppression pool containment concept.

The final item in this section records that at the 27th meeting, July 20-22, 1960, the ACRS reviewed a site proposed by Southern California Edison for location of an 1150 MWt PWR. The site was relatively remote, midway between the ends of Camp Pendleton Marine Reservation; the Committee, in approving the site, said that the design of the engineering safety features of the proposed reactor should take into account the leak rate of the containment versus the unfavorble site meteorology, especially the poor atmospheric dilution.

We shall end here this partial recounting of light water reactor siting, as it took place, case by case, during the 1950's. Two decades later it is not straightforward to find a consistent thread, or all the factors which went into each decision. It is not clear that reactor-site combinations which were rejected would have posed greater societal risks than others which were accepted. Nor is it clear what level of societal risk was sought, except that it be less than from similar technologies. The advent of containment was clearly a decisive step in moving large reactors away from highly remote sites to populated areas. Just how much additional safety containment was providing was not quantified (or quantifiable) at that time. And rather little was known about accident probabilities, both those for which the containment would function and those for which it would be violated.

Natural phenomena and external events received rather little attention regarding their influence on safety. Meteorology was a big factor in California; hydrology seemed not to enter in any important way.

We shall next look at the development of 10 CFR Part 100, Reactor Site Criteria.

2.3 1960-62 THE DEVELOPMENT OF 10 CFR PART 100

Beginning in the spring of 1960, the reactor siting criteria, as they appear in 10 CFR Part 100, began to take shape. Earlier ACRS discussions had included a review of published concepts from Great Britain and elsewhere, and an exchange of various possible U. S. approaches between the members. There was considerabe pressure on both the Regulatory Staff and the ACRS, arising from the Commissioners and from the nuclear industry, to provide some form of quantitative guidance by which the reactor designers could judge site suitability prior to Regulatory review.

At an April 8, 1960 meeting of the ACRS Environmental Subcommittee, both the Regulatory Staff and the ACRS seemed to lean toward a "maximum credible accident" in which much or all of the fission products are released to a containment which is assumed to remain intact. Both considered use of an exclusion area and an evacuation (low population) area. There were differences of opinion concerning the dose limits to be specified, but these were not large differences.

At the 25th ACRS meeting, May 5-7, 1960, Dr. Beck described his proposed philosophy (or assumptions): the probability of a major accident is relatively small; an upper limit of fission product release can be estimated; reactors are expected to be in inhabited areas; and the containment holds. The exclusion radius is defined as that at which not more than 25 Roentgen would be received in two hours. A limit of 50 Roentgen was proposed for the evacuation area. The city distance was set by 10 Roentgen exposure during the entire accident (MCA). There would be a limit of 10,000 persons in the evacuation radius, etc. It was noted that Indian Point 1 would not meet several of these criteria, while the rejected Point Loma site was satisfactory. Beck stated that the basis for acceptance of Indian Point 1 was its double containment.

The question of pressure vessel failure as a possible consideration was discussed at the April 8 Subcommittee meeting. However, notes by ACRS member Gifford of July 7, 1960 state that Beck's most recent draft is "tied to a rupture of a major pipe." No technical reasons for this decision are given.

The minutes of the Environmental Subcommittee meeting held on August 23, 1960 show a continuing emphasis by Dr. Beck of the approach discussed at the 25th ACRS meeting. The dose limit for the evacuation area was now down to 25 R whole body. There would be a limit of 10 R for the entire accident at the edge of a substantial city (10,0000). Interestingly, the minutes of the Subcommittee meeting state, "The real basis, however, for this criterion is an uncontained 'puff' release resulting in an LD-50 (50% chance of death) dose at the city boundary."

On September 21, 1960, the ACRS received a letter from W. F. Finan, the AEC Assistant General Manager for Regulation and Safety, as follows:

> "You will recall that in the spring of 1959 the Commission published for public comment proposed criteria for the evaluation of sites for power and testing reactors. Since those proposed criteria were published, the matter of site criteria has been receiving almost continuous attention from the Commission staff and the Advisory Committee on Reactor Safeguards. A new approach to site criteria has gradually emerged and has been embodied in several drafts prepared by the staff and made available informally to the ACRS during recent months. These drafts have been discussed by the staff on several ocassions with members of the Environmental Subcommittee headed by Dr. C. Rogers McCullough.

As the Chairman has publicly stated, the Commission desires to establish site criteria as soon as possible. In view of that fact we believe that the new approach to site criteria, along the lines of the draft attached to this letter, should be laid before the Commission with a view to publication for public comment. The General Manager has asked that I send this draft to you with the request that the Commission be given the comments of the Advisory Committee on Reactor Safeguards, as soon as may be convenient, regarding he suitability of the draft for publication for public comment.

I am sending you this draft at this time in order that it may be available to the Committee during its meetings this week."

The draft criteria forwarded by Finan were similar to those previously discussed and incorporated a detailed "sample calculation" which included possible (worst 20%) meteorological input data and conversion factors from iodine concentration in air to thyroid dose. It was assumed that 75%, 25% and 1% of the noble gases, halogens and non-volatile fission products respecively, were released to the containment, but the criteria permitted these amounts to be reduced if special engineered safety features were available.

Instead of a fixed number of people in the evacuation area, a maximum population density of 100/square mile was now proposed, with additional limits on density in an angular sector.

A table was prepared and included in the memo, showing how well various reactors previously reviewed compared with the proposed criteria. The table is reproduced on the following page.*

There was extensive discussion of these site criteria at the 28th meeting, September 22-24, 1960, and the ACRS prepared a long letter in which it reviewed its philosophy on reactor siting. That letter, reproduced in its entirety, follows the table mentioned above.

*Reminiscing almost twenty years later, Regulatory Staff members who had worked on this draft, recalled trying to find a set of parameters and assumptions which would fit essentially all the previously approved reactor-site combinations, within some broader, generally acceptable framework.

REACTOR SITE SUMMARY (CONTAINER LEAK RATE = 0.1% PER DAY

			Exclusion Distance		Evacuation Area		City Distance	
	Exposure Limit		300 rem		300 rem		50 rem	
	Exposure Time		2 hr		0		0	
	Halogen Release		25 %		25%		25%	
	Pop. Velocity		none		none		.2 cm/sec	
MWt	Reactor	Assumed Leak Rate	Criteria	Actual	Criteria Distance	Actual Pop. density	Criteria	Actual
630	Dresden	0.1%	.40	.5	7.7	41.1	9.6	14
585	Con. Ed.	0.1%	.38	.3*	7.4	384	9.2	1.3*
485	Yankee	0.1%	.34	.5	6.5	31.8	8.4	21
300	PRDC	0.1%	.25	.75	4.8	29.7	6.6	7.5
270	PWR	0.1	.23	.4	4.4	265	6.2	7.5
240	Consumers	0.1	.23S	.5	4.1	32.8	5.8	135
240	Hallam	0.1	.23S	.25	4.1	10.7	5.8	17
203	Pathfinder	0.1	.22S	.5	3.7	26.1	5.3	3.5*
202	PG&E	0.1	.22S	.25	3.7	201	5.3	3*
200	LCBWR	0.1%	.22S	.2*	3.7	81.3	5.3	10
153	FWCNG	0.1%	.21S	.42	3.1	2.8	4.6	15
115	Phil. Elec.	0.1%	.20S	.57	2.5	31.2	3.9	21
60	NASA	0.1%	.16S	.57	1.7	33.3	3.3	3*
60	CVTR	0.1%	.16S	.5	1.7	6.8	3.3	25
60	Jamestown	0.1%	.16S	.3	1.7	836	3.3	0.5*
58	Elk River	0.1%	.16S	.23	1.7	20.6	3.2	20
50	VBWR	0.1	.15S	.4	1.5	22.3	2.5	15
48	Pique	0.1%	.15S	.14*	1.5	1020	2.5	27
40	Pt. Loma	0.1%	.14S	.25	1.3	0	2.3	3

*does not pass criteria

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

October 22, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: REACTOR SITE CRITERIA

Dear Mr. McCone:

You have asked that we supply you with criteria which could be used for judging the adequacy of proposed sites for reactors. The Advisory Committee on Reactor Safeguards has devoted considerable time to this problem. A large part of our delay in submitting site criteria stems from the fact that we believe it is premature to establish quantitative limits on the variables involved in site evaluations - especially if such limits will appear in Federal regulations, or otherwise be announced as Commission policy. We recognize that the correctness of the numbers which could be selected now cannot be proved by experimental or empirical data, and, therefore, these numbers would give a false sense of positiveness which could not be supported upon detailed scrutiny. Numbers chosen now will be expected to change as more information develops. For example, a quantitative calculation of dosage must include some estimate of the fraction of the total fission product inventory which may be air-borne. This fraction is currently under experimental examination and the estimate may be subject to change.

The Committee believes that the officially endorsed numbers could stifle progress toward a better selection of numbers. The ideas and interpretations from applicants themselves have played a major part in the formulation of the current bases for site evaluation. It would be a significant loss to stop the flow of new ideas from the applicants. The Committee also believes that it is possible that the appearance of quantitative numbers in a Federal regulation or policy statement will reduce the continual awareness of the applicant that he has assumed a responsibility to be alert to and to act on unforeseen disadvantages of a site even after the site has been approved. The Committee, therefore, advises that a quantitative statement of site criteria not be included in Federal regulations.

These comments do not mean that the ACRS has no bases for judging the adequacy of sites. They merely emphasize that site selection is still largely a matter of judgment. Inasmuch as the ACRS has been making site and reactor evaluations, it may be helpful to review the framework on which these judgments are being made. It is a prerequisite, of course, that the reactor be carefully and competently designed, constructed, and operated. It should be inspected during all these stages in a manner to assure preservation of the intended protection of the public. Also, these factors are applicable only to those reactors on which experience has been developed. Reactors which are novel in design, unproven as prototypes, or which do not have adequate theoretical and experimental or pilot plant experience belong at isolated sites - the degree of isolation required depending on the amount of experience which exists.

Our site evaluations stem from several concepts. These are overlapping, but not conflicting:

1) Everyone off-site must have a reasonably good chance of not being seriously hurt if an unlikely but credible reactor accident should occur.

2) The exposure of a large segment of society in terms of integrated man-rems should not be such as to cause a significant shortening of the average individual lifetime or a significant genetic damage or a significant increase in leukemia - should a credible reactor accident occur.

3) There should be an advantage to society resulting from locating a plant at the proposed site rather than in a more isolated area.

4) Even if the most serious accident possible (not normally considered credible) should occur, the numbers of people killed should not be catastrophic.

Incidentally, the concept has been proposed by others that the damage to people from reactor accidents can be accepted if it is no greater than that experienced in other industries. We reject this suggestion as premature, and follow rather the concept that the consequences of reactor accidents must be less than this. The reasons for this rejection are twofold: First, we do not have sufficient information on the probability of reactor accidents to make use of this concept in site evaluations. We do use, of course, the fact that the probability of a serious accident is very low. Second, we recognize that the atomic power business has not yet reached the status of supplying an economic need in a manner similar to that of more mature industries; and, therefore, arguments for taking conventional risks for the greater good of the public are somewhat weak. At the same time, we do not want to imply that the restrictions placed on site locations during the developmental period of atomic power will necessarily be carried over to the period of maturity of the atomic power industry.

The reduction of these concepts to a judgment as to the adequacy of a proposed site requires further logic and the introduction of some numerical estimates. We believe that the searching analysis which is necessary at this stage should be done independently by the owner of the reactor, using the characteristics which are peculiar to his site and to his specific reactor. This step, we believe, is essential in developing his continuing alertness to his responsibility to the community surrounding the site. However, in Committee deliberation, we balance his analysis against a generalized accident which serves as a reference point from which we can better understand the analysis submitted by the applicant.

Our generalized accident analysis assumes that a serious accident has occurred and predicts in rough terms the consequences of such an accident. It is obvious that the generalized accident is an arbitrary artifact subject to change and has value only so far as it aids judgment. As a matter of fact, for certain reactors and conditions judgment will indicate that the generalized accident is too severe. In the generalized accident, we must make numerical assumptions as to the amount, type and rate of radioactivity release (the source term), the dispersal of the radioactivity in the air and in the hydrosphere, and the effect of this radioactivity on people.

<u>Source Term</u>

An arbitrary accident is assumed to occur which results in the release of fission products into the outermost building or containment shell. About 100% of the total inventory of noble gases, 50% of the halogens, and 1% of the non-volatile products are assumed to be so released. It is then assumed that this mixture leaks out of the outermost barrier at a rate defined by the designed and confirmed leak rate. The reasoning back of this source term is admittedly loose. It stems primarily from a present inability to be convinced that coolant cannot be lost somehow from the reactor core, either by spontaneous fracture of some element in the primary system a fracture caused by maloperation (instrumental or human) of the control rods. Admittedly, this assumed source term is large, but it thereby affords a factor of safety. In some cases it is justifiable to reduce this source term. It is also tacitly assumed that in this accident the outermost barrier will not be breached. The logic behind this assumption is that we require all of the components restraining the pressure of the primary system to be operating at temperatures above their nil-ductility temperature. We are, therefore, more confident, but not certain, that failure will occur by tearing rather than by brittle fracture and that the probability of ejection of missiles which penetrate the outermost barrier is low. The necessary supporting structures and shielding also protect against missile damage.

Dispersal of the Radioactivity

1) Meteorology

We assume a dilution of air-borne activity using atmospheric diffusion parameters which reflect poor, rather than average, meteorological conditions. Choice of specific parameter values follows from a survey of meteorological conditions expected to apply at the site, primarily wind and stability distributions. To analyze the generalized accident, we use the standard diffusion calculation methodology outlined, for example, in AECU-3066 and WASH-740. The atmospheric diffusion phenomena is the subject of active research, and new results can be expected to firm up and improve the present methods, although we do not anticipate major revisions in this area.

2) Hydrology

Considerations of hydrology are based on characteristics of surface and sub-surface flow as they are related to the possible release of contaminated liquids to the off-site environment. Thus, the rate and volume of surface flow and the possible presence or absence of absorbing barriers of soil between the reactor complex and important underground aquifers should be taken into consideration. These factors must be favorable for restraining the flow of radioactive materials in case of accident. Design factors, including the capability of providing adequate hold-up in the event of adverse hydrology, are also significant.

Effect of Radioactivity on People

The upper limit to the exposure to a member of the public in the generalized accident should be no higher than the maximum once-in-a-lifetime emergency dose. Such a level has not been established by AEC. We are arbitrarily using a figure of about 25 r whole body or equivalent integrated dose for this level. This figure is mentioned in Handbook 59 of the National Bureau of Standards, pages 69-70. Since the iodine dose is often controlling, we are tentatively considering a thyroid dose limitation of 200-300 rads. The dosage so far mentioned refers to limits to people when the people are considered as independent individuals. We believe that it is essential that the Atomic Energy Commission attempt to confirm through its staff or its advisors in this field that this suggested value of 25 r whole body or equivalent is without significant biological effect on the individuals who might be subjected to this dose from the generalized accident.

When large numbers of individuals are exposed to radiation, another limit also exists because of genetic effects and because of the statistical nature of induced leukemia and the shortening of the life span. The limits of exposure to large groups of people are better expressed in terms of integrated man-rems. We are considering using a figure of 4×10^6 man-rems for this limit for the people who might be exposed to radiation doses falling between 1 and 25 rems. This figure of 4×10^6 man-rems is roughly equal to the dose received from natural background by a million people during their reproductive lifetime.

The implication of these numbers is this. About a reactor site, there should be an exclusion radius in which no one resides. Surrounding this, there should be a region of low population density, so low that individuals can be evacuated if the need arises in a time which will prevent their receiving more than a dose of 25 r. Beyond this evacuation area, there should be no cities (above 10,000 to 20,000 population) sufficiently close so that the individuals in these cities might receive more than the lower of the following: (1) 4×10^6 man-rems in the generalized accident, and (2) 200 rems under the extremely improbable accident in which the outermost barrier fails completely to restrain all of the radioactivity of the generalized accident.

The Committee wishes to emphasize again that the numbers which have been used in discussion of the generalized accident should not be formalized into regulations or Commission policy. The Committee wishes to acknowledge the help it has received from the Hazards Evaluation Branch in this matter and suggests that these individuals be encouraged to present as technical papers, but not as regulations, a complete description of their working approach to making judgments on the adequacy of proposed reactor sites. Such a paper, of course, would have the status of the opinion of an informed technical individual, but would not imply Committee approval, nor would it have the rigidity of a Commission policy statement.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, Dir., DL&R

The letter presents the basic ACRS concepts as follows:

1. Everyone offsite must have a reasonably good chance of not being seriously hurt if an unlikely but credible reactor accident should occur.

2. The exposure to a large segment of society in terms of integrated man-rems should not be such as to cause significant shortening of an individual lifetime, or significant genetic damage or significant increase in luekemia, should a credible reactor accident occur.

3. There should be an advantage to society resulting from locating a plant at a proposed site rather than in a more isolated area.

4. Even if the most serious possible (not normally considered credible) accident should occur, the numbers of people killed should not be catastrophic.

The ACRS was very reluctant to have the AEC establish firm numbers in its regulations at a time when information was still developing. But, the Committee did give guideline numbers that they thought provided a basis for consideration. The ACRS said that they had arbitrarily been using a figure of about 25 R whole body and 300 rads to the thyroid as a limiting dose for people in the low population zone who could be evacuated, or those beyond the low population zone who might stay in place during the course of the accident. They suggested using a figure of 4×10^6 man-rem as a maximum exposure integral (to be computed by counting all exposures falling between 1 and 25 rems) from the generalized accident involving release to the intact containment. With regard to cities and the question of no catastrophic effects, the Committee recommends that beyond the evacuation area there should be no cities, having a population above 10 or 20 thousand, so close that the individuals in these cities might receive more than 200 rems under the extremely improbable accident in which the outermost barrier fails completely to restrain all the radioactivity of the generalized accident.

Clearly the population density outside the low population zone (i.e., the nearness of population centers) would also affect the chance of staying above or below the proposed integrated man-rem dose.

It is not clear from the minutes whether the ACRS looked in detail at the reactors then approved for construction or going into operation such as Indian Point 1, Dresden 1 or perhaps the Shippingport PWR, to see whether the criterion of not exceeding 200 rem at a large population center under the worst accident conditions would be met. And, it is difficult to tell from the minutes of previous ACRS meetings the extent to which the numerical guidelines presented in the letter of October 22 had actually been applied by the ACRS.

In passing, it is noted that at its 29th meeting, Nov 3-5, 1960, the ACRS wrote a report favorable to either of two newly proposed sites for the proposed small pressurized water reactor for Jamestown, New York. The new sites were located east of the city and were stated to have adequate exclusion radii and low population density.

Following its letter of October 22, 1960, the ACRS received a memorandum dated December 7, 1960, subject "Criteria fo Judging the Adequacy of Proposed Sites for Reactors," from AEC Commissioner Olson. A copy of that memorandum follows this page.

At its 29th meeting, December 8-10, 1960, the ACRS* prepared a reply, a copy of which follows the above mentioned memorandum from Commissioner Olson.

There are no major changes between the ACRS positions and recommendations in the letter of October 22, 1960 and those in the letter of December 13, 1960.

On February 1, 1961, the Atomic Energy Commission issued notice of its proposed "Reactor Site Criteria" for public comment. The complete notice, including both the Statement of Considerations and the Site Criteria is attached following the ACRS reply to Commissioner Olson.

The proposed criteria do include an exclusion area and a low population zone, with dose limits of 25 rem whole body and 300 rem to the thyroid, as proposed by both the Regulatory Staff and the ACRS. A population center distance of at least 1 1/3 times the distance from the reactor to the outer boundary of the low population zone is proposed, together with a relatively vague statement that, "When very large cities are involved, a greater distance may be necessary because of total integrated population dose considerations."

Not in the criteria themselves but in the Statement of Considerations is found the statement, "Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic." This is similar to but less restrictive than the general ACRS recommendation on this point. And there was included the corresponding numerical criterion of 200-300 R at the edge of a population center for an uncontained accident.

The 1961 proposed Reactor Site Criteria include an Appendix which spells out a sample calculation, using what the Regulatory Staff considered to be reasonable assumptions. The 1961 Site Criteria do not mention multiple reactors at one site.

*The ACRS members at this time were the following: L. Silverman, Chairman; R. L. Doan, W. P. Connor, W. K. Ergen, D. A. Rogers, R. C. Stratton, T. J. Thompson, C. R. Williams, and A. Wolman.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

December 7, 1960

Memorandum for: Dr. Leslie Silverman, Chairman
Advisory Committee on Reactor Safeguards

Subject: CRITERIA FOR JUDGING THE ADEQUACY OF PROPOSED SITES FOR REACTORS

In the letter from the ACRS to Chairman McCone, dated October 22, 1960, you make the point that criteria for reactor sites cannot be given in precise quantitative terms. YOu also make the point that the Committee believes it would be unfortunate if numbers or specific quantity were put out as part of regulations dealing with site criteria at this time. The suggestion was made that some of the quantitative aspects of site criteria might be discussed by members of the AEC staff in the form of technical papers.

As I read the Committee letter, it actually contains site guidance but not clearly identified as criteria. While recognizing the difficulty of writing detailed criteria at an early stage of a technology, it is in the interests of sound regulatory practice to have criteria to the extent possible to work from. In particular, the problem of the selection of suitable sites for nuclear reactors is and has been a troublesome point.

It is my understanding that the Advisory Committee on Reactor Safeguards is holding a meeting on December 8, 9, and 10, 1960. It would be most helpful if the Committee would try to summarize whatever general site criteria it considers appropriate for the selection of reactor sites. It should be borne in mind that criteria should be as definitive as possible, but a clear expression in general terms of the concept used would be a useful step forward. It is my feeling that it is important that we make any criteria we can available to the public in order to provide guidance to the extent possible to the public. The Commission would appreciate anything the Committee can do along this line.

L. K. Olson

L. K. Olson
Commissioner

COPY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

December 13, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: SITE CRITERIA FOR NUCLEAR REACTORS

Dear Mr. McCone:

Commissioner Olson has observed that in our letter of advice to you dated October 22, 1960, we have furnished guidance relating to the selection of reactor sites but have not clearly identified the criteria contained therein. He has suggested that the Committee summarize such general site criteria as may be appropriate in order that guidance can be provided to the public.

While the Advisory Committee on Reactor Safeguards believes that it would be unwise to publish detailed quantitative site criteria in a regulation at this early stage of technology, we have provided in an attachment to this letter criteria which should be useful in the selection of sites for nuclear reactors.

Sincerely yours,

Sgd/LESLIE SILVERMAN

Leslie Silverman
Chairman

Enclosure:
Site Criteria for Nuclear Reactors
dated Dec. 13, 1960

COPY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

December 13, 1960

SITE CRITERIA FOR NUCLEAR REACTORS

The following site criteria are applicable only to reactors of a type and design on which experience has been developed. For reactors which are novel in design, unproven as prototypes, or which do not have adequate theoretical and experimental or pilot plant experience, these criteria will need to be modified in the direction of specifying isolated sites -- the degree of isolation required depending upon the lack of certainty as to the safe behavior of the reactor. It is a prerequisite, of course, that the reactor be carefully and competently designed, constructed and operated, and inspected during all these stages in a manner to assure preservation of the intended protection of the public.

General Concepts

1. The location must be such that everyone off-site has a reasonable chance of escaping serious injury if an unlikely but still credible accident should occur.

2. The exposure of many persons in terms of man-rems should not be such as to cause significant somatic or genetic effects should a credible accident occur. The somatic effects presently recognized are: the occurrence of leukemia and life shortening. In order to be significant, the effect must be statistically great enough to be recognizable among the variations which occur without assignable cause.

3. Even if the most serious accident possible (not normally considered credible) should occur, the numbers of people killed should not be catastrophic.

4. A site which has characteristics not clearly conforming to the foregoing criteria can only be considered if it is possible to show that there is an advantage to society in locating the reactor at this site rather than in a more isolated area. This is a matter of degree, and no site is acceptable for a non-military reactor which imposes a foreseen risk of serious injury to anyone off-site.

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Specific Criteria

1. The applicant must show that his particular reactor at the chosen site does not violate the general criteria.

2. The demonstration must be based upon the actual barriers provided to contain the radioactive material, upon the means of spreading the radioactive material (meteorology, hydrology, etc.), and upon the actual distribution of population surrounding the site.

3. There must be three or more independent physical barriers unless it can be shown that the particular reactor cannot break through a lesser number. For example, if there were a reactor type with no stored energy, an "incredible" possibility for a nuclear excursion, and either such a low specific power or such a reliable cooling system that the loss of coolant accident need not be considered, one or two barriers might be considered acceptable.

4. The analysis of the applicant presenting the kind of accidents which can happen, the provisions against such accidents, and the estimate of dosage to persons off-site based upon actual meteorology, hydrology and population distributin will be compared to a generalized source term for release of radioactive material. This will be an arbitrary release of a certain fraction of the fission products in the reactor into the outermost building or container. The maximum arbitrary value will be used unless the applicant can show good reason to use a lesser quantity. The results to the persons off-site shall not be worse than those set forth in the general criteria.

5. There shall be an area surrounding the reactor, known as the exclusion area, which will be under the control of the applicant and in which no one will reside. Credit can be taken for special geographical characteristics such as a seaside location, the bank of a wide river, the bank of a lake, etc., in reducing the size of the exclusion area if the actual location warrants. Location underground would also be considered as a means of reducing the exclusion area.

Numerical Values

The Advisory Committee on Reactor Safeguards believes strongly that there has not yet been a sufficient critical review of the data available to set such numbers as part of a formal regulation. The ACRS recommended a study of the data applicable to the safety problems and the derivation of criteria for all parts of the reactor systems in a letter dated November 16, 1959. As far as the Committee is aware, there has been no such study. Data and numbers applicable to site criteria were suggested as a part of the proposed study. Such a study would permit numbers to be used in defining criteria for site selection. The following numerical values are given as examples to aid in understanding the problem even though their validity is open to question until the study is made.

1. Under the extreme conditions of a serious reactor accident, it should be reasonably possible for persons off-site to take protective steps, such as evacuation and retirement to shelters, within a period of two hours so that within the two hours they will not receive more than a 25 rem whole body gamma dose or the inhalation of radioactive material which will give a dose of 300 rem to the thyroid, or 25 rem to the bones or lung.

2. The integrated man-rem dose for all people off-site receiving a radiation dose above 1 rem whole body, or equivalent thyroid, bone or lung dose, shall not exceed 4×10^6 man-rems.

3. The reactor should be located sufficiently distance from cities (metropolitan areas) of above 10,000 to 25,000 population so that no inhabitant receives more than 300 rems in the extremely improbably accident defined by a complete failure of all confinement barriers and a source strength equal to most of the fission product inventory.

It is of some interest to note that in his testimony at the JCAE Hearings, on Radiation Safety and Regulation, June 12-15, 1961, Mr. Robert Loewenstein, Acting Director, AEC Division of Licensing and Regulations specifically discussed the population center distance as follows:

> "If one could be absolutely certain that no accident greater than the "maximum credible accident" would occur, then the 'exclusion area' and 'low population' zone would provide reasonable protection to the public under all circumstances. There does exist, however, a theoretical possibility that substantially larger accidents could occur. It is believed prudent at present, when the practice of nuclear technology does not rest on a solid foundation of extended experience, to provide protection against the most serious consequences of such theoretically possible accidents. Consideration of a 'population center distance' is therefore prescribed: This is a distance by which the reactor would be so removed from the nearest major concentration of people that lethal exposures would not occur in the population center even from an accident in which the containment is breached."

The AEC received a wide range of comments on the proposed site criteria. On July 31, 1961 a meeting was held between representatives of the Atomic Industrial Forum (AIF) and of the AEC, including Commissioner Graham. Mr. W. K. Davis made several comments in reviewing the position of the AIF:

"1) The example given in Appendix A should be deleted

"2) The population center distance should be deleted since the 1-1/3 number is without technological basis

"3) If the AEC's policy is against the location of reactors in cities, it should be so stated as a matter of policy and not inferred by calculation."

In succeeding drafts of the Reactor Site Criteria, words like catastrophic, evacuation, and fall-out were avoided, and the appendix was deleted, to be replaced by a new AEC report, TID-14844, which provided considerable detail on the methodology and parameters to be used in calculating accident doses per the requirements of the criteria.

On April 12, 1962, the USAEC published Part 100, Reactor Site Criteria, to be effective one month later. The Statements of Considerations and Criteria, which are also given in the Section entitled "A Brief Overview" are repeated here for ready reference.

The Statement of Considerations discusses the population center distance as a way to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be

ATOMIC ENERGY COMMISSION

[10 CFR Part 100]

REACTOR SITE CRITERIA

Notice of Proposed Guides

Statement of considerations. On May 23, 1959, the Atomic Energy Commission published in the FEDERAL REGISTER a notice of proposed rule making that set forth general criteria for the evaluation of proposed sites for power and testing reactors. Many comments were received from interested persons reflecting, generally, opposition to the publication of site criteria, as an AEC regulation, both because such a regulation would, to some extent, incorporate arbitrary limitations and because it appeared that in view of the lack of available experimental and empirical data specific criteria could not be established.

Judgment of suitability of a reactor site for a nuclear plant is a complex task. In addition to normal factors considered for any industrial activity, the possibility of release of radioactive effluents requires that special attention be paid to physical characteristics of the site, which may cause an incident or be of significant importance in increasing or decreasing the hazard resulting from an incident. Moreover, the inherent characteristics and the specifically designed safeguard features of the reactor are of paramount importance in reducing the possibility and consequences of accidents which might result in the release of radioactive materials. All of these features of the reactor plus its purpose and method of operation must be considered in determining whether location of a proposed reactor at any specific site would create an undue hazard to the health and safety of the public.

Recognizing that it is not possible at the present time to define site criteria with sufficient definiteness to eliminate the exercise of agency judgment, the proposed guides set forth below are designed primarily to identify a number of factors considered by the Commission and the general criteria which are utilized as guides in evaluating proposed sites.

The basic objectives which it is believed can be achieved under the criteria set forth in the proposed guides, are:

(a) Serious injury to individuals off-site should be avoided if an unlikely, but still credible, accident should occur.

(b) Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic.

(c) The exposure of large numbers of people in terms of total population dose should be low. The Commission intends to give further study to this problem in an effort to develop more specific guides on this subject. Meanwhile, in order to give recognition to this concept the population center distances to very large cities may have to be greater than those suggested by these guides.

Notice is hereby given that adoption of the following guides is contemplated. All interested persons who desire to submit written comments and suggestions for consideration in connection with the proposed guides should send them to the Secretary, United States Atomic Energy Commission, Washington 25, D.C., Attention: Director, Division of Licensing and Regulation, within 120 days after publication of this notice in the FEDERAL REGISTER.

GENERAL PROVISIONS

Sec.
100.1 Purpose.
100.2 Scope.
100.3 Definitions.

SITE EVALUATION FACTORS

100.10 Factors to be considered when evaluating sites.
100.11 Determination of exclusion area, low population zone, and population center distance.

General Provisions

§ 100.1 Purpose.

It is the purpose of this part to describe the criteria which guide the Commission in its evaluation of the suitability of proposed sites for power and testing reactors subject to Part 50 of this chapter. Because it is not possible to define such criteria with sufficient definiteness to eliminate the exercise of agency judgment in the evaluation of these sites, this part is intended primarily to identify a number of factors considered by the Commission and the general criteria which are utilized as guides in approving or disapproving proposed sites.

§ 100.2 Scope.

(a) This part applies to applications filed under Part 50 of this chapter for construction permits and operating licenses for power and testing reactors.

(b) The site criteria contained in this part apply primarily to reactors of a general type and design on which experience has been developed, but can also be applied with additional conservatism to other reactors. For reactors which are novel in design, unproven as prototypes, and do not have adequate theoretical and experimental or pilot plant experience, these criteria will need to be applied more conservatively. This conservatism will result in more isolated sites—the degree of isolation required depending upon the lack of certainty as to the safe behavior of the reactor. It is essential, of course, that the reactor be carefully and competently designed, constructed, operated, and inspected.

§ 100.3 Definitions.

As used in this part:

(a) "Exclusion area" means the area surrounding the reactor, access to which is under the full control of the reactor licensee. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations, and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

(b) "Low population zone" means the area immediately surrounding the exclusion area which contains residents the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

(c) "Population center distance" means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

(d) "Power reactor" means a nuclear reactor of a type described in §§ 50.21(b) or 50.22 of this chapter designed to produce electrical or heat energy.

(e) "Testing reactor" means a "testing facility" as defined in § 50.2 of this chapter.

Site Evaluation Factors

§ 100.10 Factors to be considered when evaluating sites.

In determining the acceptability of a site for a power or testing reactor, the Commission will take the following factors into consideration:

(a) Population density and use characteristics of the site environs, including, among other things, the exclusion area, low population zone, and population center distance.

(b) Physical characteristics of the site, including, among other things, seismology, meteorology, geology and hydrology. For example:

(1) The design for the facility should conform to accepted building codes or standards for areas having equivalent earthquake histories. No facility should be located closer than ¼ to ½ mile from the surface location of a known active earthquake fault.

(2) Meteorological conditions at the site and in the surrounding area should be considered.

(3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Unless special precautions are taken, reactors should not be located at sites where radioactive liquid effluents might flow readily into nearby streams or rivers or might find ready access to underground water tables.

Where some unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

(c) Characteristics of the proposed reactor, including proposed maximum power level, use of the facility, the extent to which the design of the facility incorporates well proven engineering standards, and the extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental releases of radioactive material.

§ 100.11 Determination of exclusion area, low population zone, and population center distance.

(a) As an aid in evaluating a proposed site, an applicant should assume a fission product release from the core as illustrated in Appendix "A" of this part, the expected demonstrable leak rate from the containment, and meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and a population center distance. For the purpose of this analysis, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least 1⅓ times the distance from the reactor to the outer boundary of the low population zone. In applying this guide due consideration should be given to the population distribution within the population center. Where very large cities are involved, a greater distance may be ncessasry because of total integrated population dose considerations.

The whole body dose of 25 rem referred to above corresponds to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations, may be disregarded in the determination of their radiation exposure status. (See Addendum dated April 15, 1958 to NBS Handbook 59.) The NCRP has not published a similar statement with respect to portions of the body, including doses to the thyroid from iodine exposure. For the purpose of establishing areas and distances under the conditions assumed in these guides, the whole body dose of 25 rem and the 300 rem dose to the thyroid from iodine are believed to be conservative values.

(b)(1) Appendix "A" of this part contains an example of a calculation for hypothetical reactors which can be used as an initial estimate of the exclusion area, the low population zone, and the population center distance.

(2) The calculations described in Appendix "A" of this part are a means of obtaining preliminary guidance. They may be used as a point of departure for consideration of particular site requirements which may result from evaluations of the particular characteristics of the reactor, its purpose, method of operation, and site involved. The numerical values stated for the variables listed in Appendix "A" of this part represent approximations that presently appear reasonable, but these numbers may need to be revised as further experience and technical information develops.

Dated at Germantown, Maryland, this 9th day of February 1961.

For the Atomic Energy Commission.

Woodford B. McCool,
Secretary.

APPENDIX "A"

Example of a calculation of reactor siting distances:

1. The calculations of this Appendix are based upon the following assumptions:

a. The fission product release to the atmosphere of the reactor building is 100 percent of the noble gases, 50 percent of the halogens and 1 percent of the solids in the fission product inventory. This release is equal to 15.8 percent of the total radioactivity of the fission product inventory. Of the 50 percent of the halogens released, one-half is assumed to adsorb onto internal surfaces of the reactor building or adhere to internal components.

b. The release of radioactivity from the reactor building to the environment occurs at a leak rate of 0.1 percent per day of the atmosphere within the building and the leakage rate persists throughout the effective course of the accident which, for practical purposes, is until the iodine activity has decayed away.

c. In calculating the doses which determine the distances, fission product decay in the usual pattern has been assumed to occur during the time fission products are contained within the reactor building. No decay was assumed during the transit time after release from the reactor building.

d. No ground-deposition of the radioactive materials that leak from the reactor building was assumed.

e. The atmospheric dispersion of material leaking from the reactor building was assumed to occur according to the following relationship:

$$X=\frac{Q}{\pi u\sigma_y\sigma_z}$$

where Q is rate of release of radioactivity from the containment vessel, the ("source term,"):

X is the atmospheric concentration of radioactivity at distance d from the reactor

u is the wind velocity

σ_y and σ_z are horizontal and vertical diffusion parameters resp.

f. Meteorological conditions of atmospheric dispersion were assumed to be those which are characteristic of the average "worst" (least favorable) weather conditions for average meteorological regimes over the country. For the purposes of these calculations, the parameters used in the equation in section e. above were assigned the following values:

$u=1$m/sec;
$\sigma_y=[\tfrac{1}{2}C_y^2d^{2-n}]^{1/2}$;
$\sigma_z=[\tfrac{1}{2}C_z^2d^{2-n}]^{1/2}$;
$C_y=0.40$;
$C_z=0.07$;
$n=0.5$

g. The isotopes of iodine were assumed to be controlling for the low population zone distance and population center distance. The low population zone distance results from integrating the effects of iodine 131 through 135. The population center distance equals the low population zone distance increased by a factor of one-third.

h. The source strength of each iodine isotope was calculated to be as follows:

Isotope	*Exclusion Q (curies/megawatt)*	*Low population Q (curies/megawatt)*
I^{131}	0.55	76.4
I^{132}	.68	1.40
I^{133}	1.19	18.5
I^{134}	.72	.91
I^{135}	1.04	5.4

These source terms combine the effects of fission yield under equilibrium conditions, radioactive decay in the reactor building, and the release rate from the reactor building, all integrated throughout the exposure time considered.

i. For the exclusion distance, doses from both direct gamma radiation and from iodine in the cloud escaping from the reactor building were calculated, and the distance established on the basis of the effect requiring the greater isolation.

j. In calculating the thyroid doses which result from exposure of an individual to an atmosphere containing concentrations of radioactive iodine, the following conversion factors were used to determine the dose received from breathing a concentration of one curie per cubic meter for one second:

Isotope	*Dose (rem)*
I^{131}	329
I^{132}	12.4
I^{133}	92.8
I^{134}	5.66
I^{135}	25.3

k. The whole body doses at the exclusion and low population zone distances due to direct gamma radiation from the fission products released into the reactor building were derived from the following relationships:

$$D=483\frac{Be^{-\mu r}}{4\pi r^2}\int^{t} t^{-1.2}dt$$

where D is the exposure dose in roentgens per megawatt of reactor power

r is the distance in meters

B, the scattering factor, is equal to $\left(1+\mu r+\frac{\mu^2r^2}{3}\right)$

μ is the air attenuation factor (0.01 for this calculation)

t is the exposure time in seconds.

In this formulation it was assumed that the shielding and building structures provided an attenuation factor of 10.

2. On the basis of calculation methods and values of parameters described above, initial estimates of distances for reactors of various power levels have been developed and are listed below.

Power level (thermal megawatt)	Exclusion distance (miles)	Low population zone distance (miles)	Population center distance (miles)
1500	0.70	13.3	17.7
1200	.60	11.5	15.3
1000	.53	10	13.3
900	.50	9.4	12.5
800	.46	8.6	11.5
700	.42	8.	10.7
600	.38	7.3	9.6
500	.33	6.3	8.4
400	.29	5.4	7.2
300	.24	4.5	6
200	.21	3.4	4.5
100	.18	2.2	2.9
50	.15	1.4	1.9
10	.08	.5	.7

[F.R. Doc. 61-1233; Filed, Feb. 10, 1961; 8:50 a.m.]

feasible. However, the rather specific criterion of the ACRS letters of October and December, 1960 (namely, that of no lethal doses at the population center) is by no means apparent (and we shall see it was not used in succeeding years).

It is of interest to note that the minutes of the 39th meeting, February 8-10, 1962, record dissatisfaction by several members concerning the 1962 version of 100 CRF Part 100. Member Silverman believed that the rewriting had eliminated some of the earlier significant ACRS ideas which had been previously incorporated. Member Ergen was concerned over ambiguity in the large city distance criterion. However, Part 100 was issued by the AEC as described above.

The 1962 version of Part 100 specifically allows for sites having multiple reactor facilities. It asks that consideration be given to the possibility that an accident in one reactor might initiate an accident in the other (5)*, and that the simultaneous operation of multiple reactors at a site not result in routine releases beyond the applicable regulations.

What is, in a sense, remarkble is that there has been very little change in this part of the Reactor Site Criteria since their adoption as an AEC Regulation in 1962. This could mean any of many things, including the following:

1) That Part 100, as originally written, was so well formulated that it has passed the test of time and can continue to be used for giving rather direct and appropriate guidance in the choice of sites;

2) that it was formulated in a sufficiently general fashion, or included enough permissible alternatives, that it permitted a wide range of interpretation, enough to cover all situations arising since 1962; or

3) that it is not too meaningful in terms of our present degree of knowledge, but that it is difficult to find a new set of reactor site criteria which can be considered a defensible improvement.

And there are, undoubtedly, other possible explanations for the absence of important changes in Part 100.

*In practice, this has normally been reviewed in the context of a Class 9 accident, that is an accident whose consequences exceed Part 100.

PART 100—REACTOR SITE CRITERIA

Pursuant to the Administrative Procedures Act and the Atomic Energy Act of 1954, as amended, the following guide is published as a document subject to codification, to be effective 30 days after publication in the FEDERAL REGISTER.

Statement of considerations. On February 11, 1961, the Atomic Energy Commission published in the FEDERAL REGISTER a notice of proposed rule making that set forth general criteria in the form of guides and factors to be considered in the evaluation of proposed sites for power and testing reactors. The Commission has received many comments from individuals and organizations, including several from foreign countries, reflecting the widespread sensitivity and importance of the subject of site selection for reactors. Formal communications have been received on the published guides, including a proposed comprehensive revision of the guides into an alternate form.

In these communications, there was almost unanimous support of the Commission's proposal to issue guidance in some form on site selections, and acceptance of the basic factors included in the proposed guides, particularly in the proposal to issue exposure dose values which could be used for reference in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence.

On the other hand, many features of the proposed guides were singled out for criticism by a large proportion of the correspondents. This was particularly the case for the appendix section of the proposed guides, in which was included an example calculation of environmental distance characteristics for a hypothetical reactor. In this appendix, specific numerical values were employed in the calculations. The choice of these numerical values, in some cases involving simplifying assumptions of highly complex phenomena, represent types of considerations presently applied in site calculations and result in environmental distance parameters in general accord with present siting practice. Nevertheles, these particular numerical values and the use of a single example calculation were widely objected to, basically on the grounds that they presented an aspect of inflexibility to the guides which otherwise appeared to possess considerable flexibility and tended to emphasize unduly the concept of environmental isolation for reactors with minimum possibility being extended for eventual substitution thereof of engineered safeguard.

In consequence of these many comments, criticisms and recommendations, the proposed guides have been rewritten, with incorporation of a number of suggestions for clarification and simplification, and elimination of the numerical values and example calculation formerly constituting the appendix to the guides. In lieu of the appendix, some guidance has been incorporated in the text itself to indicate the considerations that led to establishing the exposure values set forth. However, in recognition of the advantage of example calculations in providing preliminary guidance to application of the principles set forth, the AEC will publish separately in the form of a technical information document a discussion of these calculations.

These guides and the technical information document are intended to reflect past practice and current policy of the Commission of keeping stationary power and test reactors away from densely populated centers. It should be equally understood, however, that applicants are free and indeed encouraged to demonstrate to the Commission the applicability and significance of considerations other than those set forth in the guides.

One basic objective of the criteria is to assure that the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low in comparison with what might be considered reasonable for total population dose. Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible. Neither of these objectives were readily achievable by a single criterion. Hence, the population center distance was added as a site requirement when it was found for several projects evaluated that the specification of such a distance requirement would approximately fulfill the desired objectives and reflect a more accurate guide to current siting practices. In an effort to develop more specific guidance on the total man-dose concept, the Commission intends to give further study to the subject. Meanwhile, in some cases where very large cities are involved, the population center distance may have to be greater than those suggested by these guides.

A number of comments received pointed out that AEC siting factors included considerations of population distributions and land use surrounding proposed sites but did not indicate how future population growth might affect sites initially approved. To the extent possible, AEC review of the land use surrounding a proposed site includes considerations of potential residential growth. The guides tend toward requiring sufficient isolation to preclude any immediate problem. In the meantime, operating experience that will be acquired from plants already licensed to operate should provide a more definitive basis for weighing the effectiveness of engineered safeguards versus plant isolation as a public safeguard.

These criteria are based upon a weighing of factors characteristic of conditions in the United States and may not represent the most appropriate procedure nor optimum emphasis on the various interdependent factors involved in selection of sites for reactors in other countries where national needs, resources, policies and other factors may be greatly different.

AUTHORITY: §§ 100.1 to 100.11 issued under sec. 103, 68 Stat. 936, sec. 104, 68 Stat. 937, sec. 161, 68 Stat. 948, sec. 182, 68 Stat. 953; 42 U.S.C. 2133, 2134, 2201, 2232

SOURCE: §§ 100.1 to 100.11 appear at 27 F.R. 3509, Apr. 12, 1962.

§ 100.1 *Purpose.* (a) It is the purpose of this part to describe criteria which guide the Commission in its evaluation of the suitability of proposed sites for stationary power and testing reactors subject to Part 50 of this chapter.

(b) Insufficient experience has been accumulated to permit the writing of detailed standards that would provide a quantitative correlation of all factors significant to the question of acceptability of reactor sites. This part is intended as an interim guide to identify a number of factors considered by the Commission in the evaluation of reactor sites and the general criteria used at this time as guides in approving or disapproving proposed sites. Any applicant who believes that factors other than those set forth in the guide should be considered by the Commission will be expected to demonstrate the applicability and significance of such factors.

§ 100.2 *Scope.* (a) This part applies to applications filed under Part 50 and 115 of this chapter for stationary power and testing reactors.

(b) The site criteria contained in this part apply primarily to reactors of a general type and design on which experience has been developed, but can also be applied to other reactor types. In particular, for reactors that are novel in design and unproven as prototypes or pilot plants, it is expected that these basic criteria will be applied in a manner that takes into account the lack of experience. In the application of these criteria which are deliberately flexible, the safeguards provided—either site isolation or engineered features—should reflect the lack of certainty that only experience can provide.

§ 100.3 *Definitions.* As used in this part:

(a) "Exclusion area" means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

(b) "Low population zone" means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

(c) "Population center distance" means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

(d) "Power reactor" means a nuclear reactor of a type described in § 50.21(b) or 50.22 of this chapter designed to produce electrical or heat energy.

(e) "Testing reactor" means a "testing facility" as defined in § 50.2 of this chapter.

Site Evaluation Factors

§ 100.10 *Factors to be considered when evaluating sites.* Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

(a) Characteristics of reactor design and proposed operation including:

(1) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(2) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur.

(b) Population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.

(c) Physical characteristics of the site, including seismology, meteorology, geology and hydrology.

(1) The design for the facility should conform to accepted building codes or standards for areas having equivalent earthquake histories. No facility should be located closer than one-fourth mile from the surface location of a known active earthquake fault.

(2) Meteorological conditions at the site and in the surrounding area should be considered.

(3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow into nearby streams or rivers or might find ready access to underground water tables.

(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

§ 100.11 *Determination of exclusion area, low population zone, and population center distance.* (a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release [1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem [2] or a total radiation dose in excess of 300 rem [2] to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, due consideration should be given to the population distribution within the population center.

Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

[2] The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the AEC the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

NOTE: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.

Copies of Technical Information Document 14844 may be obtained from the Commission's Public Document Room, 1717 H Street NW., Washington, D.C., or by writing the Director, Division of Licensing and Regulation, U.S. Atomic Energy Commission, Washington 25, D.C.

2.4 1961-1965 THE SITING OF LARGE LWRs

After publication of the proposed Part 100 Reactor Site Criteria in February, 1961, a continuing series of decisions were made concerning proposals for constructing increasingly large LWRs at sites which not only did not meet the 1950 "rule of thumb", but sites which did not meet Part 100 without obtaining "credit" in calculating off-site doses, either for a reduction in the leak rate (by use of double containment, for example), or by reduction of the postulated fission product source available to leak out of containment (e.g., by the use of containment atmosphere cleanup systems, such as containment sprays or closed-loop filter systems). And there was also considerable pressure from the nuclear industry for the construction of large LWR's at sites far more populated than Indian Point, including New York City itself.

We shall see that several high power reactors were approved for construction, using the MCA approach of the site criteria, with the kind of relaxation on exclusion area and low population zone distance mentioned above. These included:

San Onofre 1	1347 MWt	(1963)
Connecticut Yankee	1824 MWt	(1964)
Oyster Creek	1930 MWt	(1964-65)
Nine Mile Point	1850 MWt	(1965)
Desden 2	2527 MWt	(1965)

Although each passing year saw the evolution of new safety requirements dealing with a wide range of technical issues, the principal focus for construction permit review during this period appears to have been on the efficacy of the engineered safeguards (containment plus filters and/or sprays) needed to meet the dose guidelines of Part 100. Accidents exceeding the MCA were not considered as part of the siting or construction permit review.

We shall next examine major aspects of this time period in more detail.

The ACRS has previously reported favorably on a proposed California site in the middle of Camp Pendleton. At the 35th meeting, July 6-8, 1961, the ACRS was asked about the possibility of relocating this site northward by seven miles, placing it considerably closer to San Clemente, California. The ACRS advised the AEC that "Re-siting to a location seven miles north of the proposed site does not rule out its being acceptable. It might, however, require additional safeguards in order to properly protect the public."

At its 36th meeting, September 7-9, 1961, the ACRS reviewed the request of the City of Los Angeles for approval of 8 proposed reactor sites. The sites were in three general areas of northwestern Los Angeles couty, the San Francisquito Canyon area, the Green Valley area, and the Fairmont area.

The first two of these areas are in the western Angeles National Forest and the third is in the southwestern Antelope Valley. The population distribution at the time of review was favorable. The Committee gave attention to projected population growth and looked at the meteorology for these sites. For at least the San Franciquito Canyon site, it was stated that the meteorology could be unfavorable and could give relatively small dilution of releases. The Committee reported that this area might require more engineering safeguards than reactors at the other sites. However, the ACRS concluded that it would be possible to locate reactors at any of these sites without undue risk to the health and safety of the public. It was noted that the proposed concept included the eventual operation of four 300 MWe reactors at whatever site was chosen.

At the 38th meeting, December 7-9, 1961, the ACRS again considered reactor sites for the City of Los Angeles. This time two new sites were proposed. One of them, the Haynes Point site, was very near Long Beach. The other was the Corral Beach site, west of Los Angeles near the ocean. The Regultory Staff were of the opinion that the Haynes Point site was unacceptable because of the very large nearby population. The Regulatory Staff wished to discourage use of both the sites, but they did not believe it impossible to locate a safe reactor at the Corral Beach location. The ACRS did not comment on the two new proposed sites at this meeting.

At the 40th meeting, March 29-31, 1962, there was a meeting between the Committee and Commissioner Loren K. Olson. The minutes note that Commissioner Olson hoped that the project for reactors in the Los Angeles area could proceed. He asked for a positive approach by the safety review groups toward the recently proposed sites, and he asked if either an underground location or one in a hillside would be acceptable. At the same meeting, member of the ACRS Staff reviewed the Congressional hearings related to the development of atomic energy. He reported that a representative of a Boston consulting engineering firm, Mr. Harold Vann, had impressed the Joint Committee on Atomic Energy. He advised that Vann had testified that the site criteria would deter the construction of reactors, and that Vann had proposed more development toward iodine removal to help alleviate the situation.

At its 40th meeting, the ACRS continued its review of the proposal by Los Angeles Department of Water and Power for consideration of the Haynes Point and the Corral Beach sites. Neither site met the new site criteria, and Mr. Newell of the Regulatory Staff reported that there had to be dependence on engineered safeguards because of the lack of isolation. He outlined the required distances for the low population radius and the population center distance for a given release, and for this release reduced by arbitrary factors of 50%, 90% and 99% with engineered safeguards as yet unproved. The low population distances obtained in this fashion were 11 miles, 7 miles, 2 miles and 0.5 miles, respectively. The population center distances (1-1/3 as large) thus obtained, were 15 miles, 9 miles, 3 miles and 0.7 miles, respectively. It appears from the minutes that Corral Beach and Haynes Point were estimated to have (either) population center or low population center distances of the order of 10 miles and 1/4 mile, respectively.

ACRS Member Gifford was quoted as saying that bad meteorological conditions occur in Southern California more often than in the rest of the U.S., but there are worse conditions in the U.S. at times. And Committee member Osborn noted that engineering safeguards might do away with any meteorological considerations.

C. Rogers McCullough, who had been an ACRS member and was now acting as a consultant, is quoted as corroborating fears expressed by Mr. DiNunno of the Regulatory Staff that the heat from the decay of fission products might breach the large amount of concrete in the General Electric reactor that was being proposed.

A Westinghouse group presented information on their proposed reactor for the Haynes Point site. The special features which were provided to make it acceptable included, in particular, a double containment system with twin liners surrounding an annular space to be held at negative pressure. So, we see here the proposal that the provision of very low leakage for the MCA is what is needed for urban (nearly metropolitan) siting, without consideration of the maintenance of containment integrity for severe accidents.

General Electric proposed a 400 MWe direct cycle boiling water reactor, using a pressure suppression containment such as was being used at Humboldt Bay. Its safety objectives included low probabilities of an uncontained accident, but no details are given.

The ACRS also heard a presentation by Dr. George Housner, a consultant to the Department of Water and Power, in which he expressed confidence that a reactor could be satisfactorily designed for seismic conditions at either site.

The ACRS wrote a report on this meeting dated April 4, 1962, and made the following comments with regard to the two new sites:

> In its most recent proposal, the City of Los Angeles presented two coastal sites which its representative stated present appreciable economic advantages over the presently accepted sites. These two sites are a southen site now owned by the City, and a western site which could be obtained.

> In regard to the two new sites proposed for reactors of the general concepts presented, the Committee has the following comments: Neither of the locations can meet the site criteria guidelines proposed in 10 CFR-100 for the power level requested. Both sites are within areas of high and increasing population. In this connection, it should be noted that power reactors of the size proposed have not yet been built and proved. Such reactors would contain larger fission product inventories than any licensed power reactor now operating or under construction.
>
> If the sites proposed are to be considered acceptable, then reliance must be placed on proved engineering safeguards as a means of preventing exposure of significant numbers of people to possible radiation injury. The Committee believes that it is possible with present engineering technology to overcome the potential danger from serious consequences of major earthquake.
>
> The Committee has the following comment concerning the two reactor concepts proposed, and their respective containments: neither proposal provides proved assurance of satisfactory containment of an accident, such as a serious nuclear excursion, which releases radioactivity simultaneously with the release of pressure. The possibility of such an accident cannot be excluded on the basis of present knowledge.
>
> Of the two coastal sites, the western site is in an area of lower population density and is further removed from large centers of population. Neither site is suitable for either of the proposed reactor facilities. The proposed plant designs might more readily be modified to a form suitable for the western site.

There is no indication how the ACRS dealt with the point by Mr. McCullough concerning the possibility that fission product decay heat could lead to a failure of the concrete containment in the G.E. design. The question of why the G.E. design was the one discussed is not clear in the minutes.

At the 41st meeting, May 10 and 11, 1962, the ACRS came to grips with the problem of developing porting criteria for the nuclear ship Savannah. There was a very considerable amount of discussion on this topic at this meeting and at many succeeding meetings. Since it was desired to bring the Savannah into port in or near the heart of large cities, a question of how to make such visits compatible with the Part 100 siting criteria faced the Commission. Whether mobility of the ship could be used as a safety feature complicated the situation. Questions relating to porting of the Savannah used much of the Committee time for the next several meetings.

At the 43rd meeting, August 23-25, 1962, the Los Angeles Department of Water and Power was back to speak with the ACRS concerning the possibility of a newly revised design of the proposed boiling water reactor for consideration at the Corral Beach site. The 1400 MWt reactor design now included a suppression system and a confinement building around the suppression system, which also enclosed the refueling operations for the reactor. A fundamental problem facing the Committee was whether it was acceptable to have part of the primary system leave the containment, as was done in a direct cycle boiling water reactor system. Some members felt that perhaps the turbine should be inside the containment. Other members pointed out that moving machinery is a likely place of failure which could generate large missiles in the containment. The applicant-designer proposed to put two isolation valves in the steam lines running to the turbines, so that if the turbines should fail, these valves could close and avoid a loss of coolant from the core. At the 43rd meeting, the Los Angeles Department of Water and Power also described a revised concept for the PWR, employing a double containment vessel which completely enclosed the primary system and which included a feature involving back-pumping and monitored leakage of a porous "popcorn" concrete-filled space between the containment walls. The PWR proposal also included holdup of routine radioactive gas release.

The Regulatory Staff said that they had concluded that the proposed reactors could be built and operated safely at the Corral Beach site. It is not clear from any of the meeting minutes whether consideration was given at this time to the general recommendation of the October, 1960 ACRS site criteria letter concerning the limitation that, even in the event of the worst possible accident, there should not be a catastrophe. The Committee did not take action on the proposed design at the 43rd meeting.

It is clear from the minutes of the 44th meeting, October 4-6, and 12, 1962, and from other meetings in that time period that, during ACRS review of the Corral Beach site (or possibly the Haynes Point site), strong differences of opinion developed within the Committee (there is even a discussion in the minutes of the existence of proposed majority and minority letters). However, detailed differences, as they might have appeared in majority or minority letters, are not available in the minutes.

The ACRS issued a report on the reactor proposals for the Corral Beach site following its 44th meeting. In view of its importance to the development of future siting, the letter is reproduced on the following pages.

The ACRS report expressed the opinion that the PWR containment system was adequate, but had some reservations concerning the proposed boiling water reactor, particularly its dependence upon the rapid closure of isolation valves in the event of an accident involving the rupture of the steam line outside the containment. However, the Committee concluded favorably toward either reactor-type, if it were provided with adequate containment. Thus,

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

October 12, 1962

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON CITY OF LOS ANGELES WATER AND POWER DEPARTMENT BOILING WATER AND PRESSURIZED WATER REACTOR PROPOSALS

Dear Dr. Seaborg:

At its forty-third meeting August 23-25, 1962 at Idaho Falls, Idaho, and at its forty-fourth meeting in Washington, D. C., October 4-6, and 12, 1962, the Advisory Committee on Reactor Safeguards reviewed the proposed, approximately 1600 MW(t), boiling water and pressurized water reactors one of which may be constructed and operated by the Department of Water and Power of the City of Los Angeles at a site designated as the "western site". The Committee had the benefit of several subcommittee meetings, the references listed below, and discussions with representatives of the Department of Water and Power of the City of Los Angeles, the General Electric Company, Westinghouse Electric Corporation, Stone and Webster Corporation, and the AEC staff.

The Committee in its reviews has focused its attention on the adequacy of engineered safeguards for the containment of any significant potential releases that might affect the health and safety of the public.

The large pressurized water reactor has, as a proposed engineered safeguard concept, a double containment vessel which completely encloses the primary system. Back pumping and monitored leakage of a porous "popcorn" concrete filled space between the containment walls and of all penetrations are provided. The system depends to some extent on keeping the space between the membranes at negative pressure. Redundancy in the pumping equipment is used to insure against failure. The containment membranes are independent as to leakage, but depend on the porous concrete for strength. The reinforced concrete on the outside augments containment vessel strength and provides shielding. The proposal includes holdup of routine radioactive gaseous release. In the opinion of the Advisory Committee on Reactor Safeguards this containment system is adequate.

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The proposed large boiling water reactor has a pressure suppression system surrounded by an additional containment of the dry well and suppression pool. The primary steam line extends beyond this double containment to the turbine building. Containment of fission product release from an accident thus depends upon rapid closure of isolation valves. In view of the stringent requirements imposed by the site, it is the Committee's opinion that the containment as proposed is not adequate in some respects for this reactor at this site. The Committee also believes that holdup of routine gaseous releases will be necessary during unfavorable meteorological conditions.

The Advisory Committee on Reactor Safeguards believes that either reactor if provided with adequate containment of the primary system can be located at the western site with reasonable assurance that such reactor can be operated without undue risk to the health and safety of the public. It is believed also that this site may be adequate for multiple reactors assuming that suitable containment and confinement are provided.

Sincerely yours,

/s/
F. A. Gifford, Jr.
Chairman

References Attached (1 page)

the ACRS (and the Regulatory Staff) was approving rather large reactors (1600 MWt) for a site not very distant from a large population center (say 10 miles). The letter says that, "The Committee in its reviews has focused its attention on the adequacy of engineered safeguards for the containment of any significant potential releases that might affect the health and safety of the public." What appears to have been emphasized in the review was the containment system in terms of the MCA, and not engineered safety features which would keep the core from melting, or measures such as primary system quality control to prevent serious accidents from occurring. In a sense, this seems to have established a trend for the focus of regulatory review from this time until the middle of 1966, when a major change in the requirements for accident prevention and mitigation occurred. We will come to that point later.

There was no further mention in the minutes of these ACRS meetings of any possible adverse effects on containment reliability to be associated with core melt.

At a special meeting, November 9 and 10, 1962, the Los Angeles Department of Water and Power was back once again to discuss additional safety features in the proposed General Electric design for possible use at the Corral Beach site. The ACRS wrote a report concluding that a boiling water reactor of the type proposed, with adequate engineered safety features, could be located at the Corral Beach site. It also noted that its review of both the pressurized and boiling water reactors proposed had been preliminary, and that due to the high power level and proximity to densely populated areas, either the pressurized water or boiling water reactor might require improvement in safety design beyond those features incorporated in existing reactors. However, the emphasis appears to have been that of reducing the dose at the site boundary or the low population zone boundary, assuming the generalized accident and thence providing cleanup systems inside the containment, as well as reduced containment leakage rates.

The minutes of the 45th meeting, December 13-15, 1962, note that ACRS Chairman Gifford read both the AEC and Consolidated Edison announcements of plans to build a large nuclear power plant on the East River at Ravenswood in New York City by 1970.

At the 46th meeting, January 31 - February 2, 1963, the ACRS reviewed the suitability of proposed sites for the Connecticut Yankee 1473 MWt pressurized water reactor. The proposed site did not meet the existing site criteria without additional engineered safety features. The ACRS concluded that the site was suitable for the proposed Connecticut Yankee reactors if adequate containment and one or more of the engineered safeguards was provided.

At the 48th meeting, July 11-13, 1963, the ACRS wrote a report reviewing the reactor safety research program. Of particular interest is that this program devoted much of its attention to the nature and magnitude of fission product releases and to mechanisms for removing fission products from containment. Mention is made of the LOFT program, which was intended to be an experiment in which a small PWR core would be deliberately melted, and the actual course of fisison products from core melt through the containment building and into the environment would be measured. In other words, it was to be a re-enactment of the generalized accident which was being used for siting purposes. During the 48th meeting, the ACRS also began its review of the construction permit application for the Southern

California Edison reactor proposed for the northern Camp Pendleton site. The Regulatory Staff noted this site could not tolerate 100% meltdown of the fuel and full release of the fission products to the containment. Credit had to be given for an emergency core cooling system, so that only 6% of the core was assumed to melt with reduced release of fission products to the containment. The ACRS completed its review of the San Onofre reactor at the 49th meeting, September 5 and 6, 1963. The ACRS accepted the approach discussed above without making a detailed review of the actual effectiveness of the core cooling system. Excerpts from the ACRS report of September 12, 1963 on San Onofre Unit No. 1 follow:

> The applicants propose to contain the reactor in a spherical steel structure designed for a maximum leakage rate of 0.1% per day at pressure and with critical penetrations designed to permit frequent leak testing. Additional engineered safeguards are required for this site. Such safeguards proposed include a multiple, borated-water injection system to prevent extensive core meltdown in the unlikely event of a major break in the primary water system, a contanment spray system, and an internal air cleanup system.
>
> A meteorological factor favorable to the proposed reactor location is the fact that air movement from the site toward San Clemente occurs, at most, only a few percent of the time.
>
> The ACRS has emphasized that the engineered safeguards must be designed and reviewed with great care for both adequacy and reliability. Special attention should be directed to the safety injection system which must perform as proposed to validate the applicants' assumption of low release of radioactivity to the containment under accident conditions. A halogen removal system may be required. Design details of the holdup system for reactor off-gases resulting from routine operation will also require careful attention. The ACRS has recommended study of the consequences of rainout following an accident; the results of this study should be taken into account in the final design of the engineered safeguards.

In view of the favorable prevailing wind direction, conservative seismic design approach, and with engineered safeguards of the type proposed, it is the Committee's opinion that a pressurized water reactor of the type and power level proposed can be designed, constructed and operated at the site without undue hazard to the health and safety of the public.

2.5 SOME PUBLICATIONS AND SPEECHES

At this point, we shall interrupt the recounting of ACRS ation on specific site and contruction permit applications to review briefly a few publications/speeches made during the time period 1962-64.

At the 1962 IAEA (Vienna) Symposium on Reactor Safety and Hazards Evaluation Techniques, W. E. Johnson of Westinghouse Electric Corporation gave a paper entitled, "Principles and Practices in Consequences Limiting Safeguards in Facility Design". In this paper he reviewed the evolution of reactor containment and discussed several of the new containment concepts as of 1962. In particular, he described an absolute containment or <u>no-leakage</u> concept developed by Stone and Webster and proposed in connection with the LADWP Corral Beach reactor, we quote from Johnson as follows:

> "A second design concept is being developed by the Stone and Webster Engineering Corporation. This absolute containment or <u>no leakage</u> concept is the outgrowth and refinement of a containment system designed by Stone and Webster for the Carolinas-Virginia Tube Reactor now under construction at Parr, South Carolina. Here the vapour container is a steel lined, reinforced concrete structure, cylindrical in shape, with a steel-and-concrete hemispherical dome and flat concrete base. The quarter-inch thick steel liner is shaped in the interior outine of the concrete outer container and is welded to steel inserts imbedded in the concrete. This arrangment ensures leaktightness of the system.

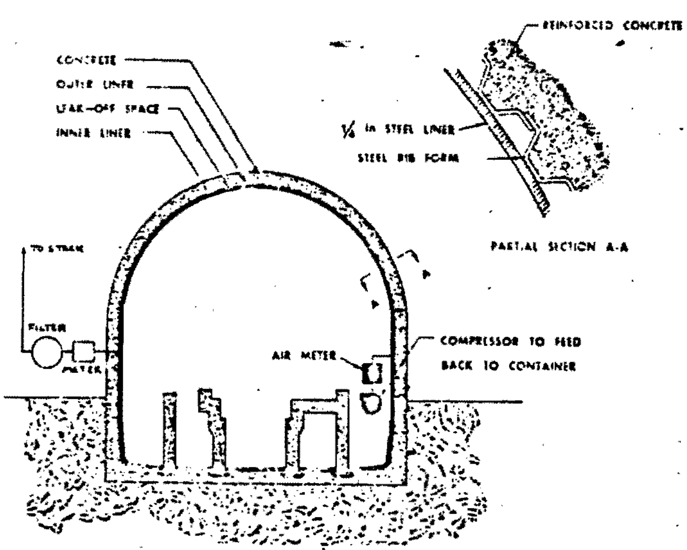

Fig. 2

Double shell containment concept

The new no leakage Stone and Webster design concept utilizes two layers of steel inside a reinforced concrete vessel. This vessel is designed to withstand the maximum internal pressure generated after rupture of the primary system. It also provides adequate biological shielding for plant personnel.

The outer steel shell serves as the form for the reinforced concrete shell. An air gap separates the inner and outer shells, but the inner shell is supported against pressure forces by the outer shell and the reinforced concrete. The inner shell is constructed to assure the highest possible degree of leaktightness. Reasonable care is observed in constructing the outer shell, although it need not be leaktight. Air in the plenum formed between the two steel shells can be maintained slightly below atmospheric pressure by means of intermittent operation of a pump which exhausts air from the plenum and discharges it inside the inner shell.

All container penetrations employ double containment seals, with the space between the seals connected to the plenum chamber. All personnel access openings are of the double door type. During plant operation, the space between these doors is connected to the plenum system. These arrangements ensure that any vapour container leakage will be into the container and that there will be absolutely no uncontrolled release of radioactive material to the atmosphere.

By monitoring vapour container pressure and pump air flow, leakage through each of the shells may be ascertained. This feature, together with the possiblity of utilizing the plenum to pinpoint any existing leaks by conducting periodic freon gas leak tests, assures ability to carry out the necessary maintenance procedures, and to preserve the initial integrity of the containment over the entire lifetime of the plant. The entire structure is designed to withstand the effects of the worst conceivable nuclear incident (generally referred to as the maximum hypothetical accident) for the type of plant the structure is to contain. Such a hypothetical accident might include core meltdown, sudden release of energy stored in the coolant, or sudden release of energy due to chemical reaction within the reactor.

A spray system is incorporated to reduce excessive pressure in the containment following a hypothetical accident. Since the air pump continuously evacuates the plenum, any leakage inward through the concrete vapour container and the outer steel lining would result in a gradual rise in pressure within the vapour container. Based upon leakage rates expected through a typical concrete shell without an outer steel lining, it would require from three to four months to elevate the container pressure to a point where it became necessary to begin releasing gases to the atmosphere.

With the addition of the steel liner, leak rates less than 0.1% volume per day may be expected. Assuming the container is designed for 3 atm integrity, it would require about 3000 d at 0.1% leakage per day to reach the design pressure. Immediately following the accident, however, a gas circulation system could be placed in operation to filter out much of the radioactive material not already precipitated out on cold surfaces. After reduction of airborne activity to safe levels, it could be passed through a series of filters and adsorbers which would remove essentially all remaining activity before discharge of gas to the stack. The contents of the vapour container can be stored for considerable time, therefore stack discharges would be restricted to favourable weather conditions, or, if necessary, the waste could be bottled for off-site disposal.

This <u>no leakage</u> concept is probably the most elaborate consequence limiting system conceived to date. If effectively developed, the concept may bring about the location of nuclear plants near or even within large populated areas. The concept has the additional advantage of being flexible enough to permit either above ground or underground placement of the containment."

Of particualr interest is the use of the term "zero-leakage", and the statement that the entire structure is designed to withstand the worst conceivable nuclear incident including core meltdown.

At the same symposium, a paper by Kellerman of Germany (Kellerman, 1962) expresses the same concept, that an LWR containment is designed to withstand total melting of the core fuel. An a paper by Blasser and Wirtz of Germany (1962) deals with the determination of reactor location and requirements for the pressure shell on the basis of the MCA.

At the IAEA (Bombay) Symposium on Siting of Reactors and Nuclear Research Centers in 1963, W. K. Ergen, a member of the ACRS, gave a paper entitled "Site Criteria for Reactors with Multiple Containment" (Ergen, 1963). In this paper he discussed the U.S. practice of power-reactor-site selection in terms of Part 100 and described how multiple containment could drastically reduce the release of iodine for the postulate MCA. He mentioned the proposed use of this concept for the Corral Beach site and a site in the heart of New York City (Ravenswood).

On September 25, 1963, Dr. Clifford Beck, then Deputy Director of the Division of Licensing and Regulation, USAEC, summarized the basic elements of U.S. siting criteria and described trends in the Commission's siting practices. The basic elements were said to be:

1. The guides serve to identify the factors which should be considered in selection of reactor sites.

2. The guides are to be used as points of departure for detailed evaluation of any individual case, with full recognition that quantitative and unique answers to required distance cannot be obtained.

3. The guides establish the concept of the maximum credible accident as the beginning point -- a concept which is opposed by many people, or even denied as being something that doesn't exist.

   It derives as an inescapable outcome of two basic premises: (1) site selection safety does not depend on the routine effluents and other hazards of normal operations; these can be controlled by any extent necessary, by extra expenditures, so that the reactors can be placed at any place desired. (Hence, for normal operations the choice of sites depends primarily on economics, not on safety); and (2) on the other hand, considering the essential question of accidents, there are virtually no sites acceptable, safetywise, if the worst conceivable accident must be assumed, i.e., the release of all the fission product inventory to the environment. The question which follows as the only recourse, is, "What is the maximum credible accident, the upper limit of hazard, which appears sufficiently possible that it must be taken into account for comparison with the protective characteristics and capabilities of the site?"

4. The site guides established the concept of, and specify numerical values for, potential radiation exposure doses to be employed as measuring indices in evaluating reactor site characteristics. These are not permissible emergency doses; they have no relationship to protective actions to be taken after an accident. They are comparative measuring indices for use in evaluating the adequacy of reactor safeguards and site characteristics, i.e., from the estimated upper limit of hazard from a reactor plant having specified characteristics, and also possessing described safeguards in design and site. If resulting exposure doses at various distances beyond the site boundary would be no greater than those values, then this plant, with these characteristics and safeguards, in this location, would be considered acceptable.

5. The guides make explicit, and define, the concepts of population zones around a reactor which had been observed in practice all along; the exclusion zone, where people are highly mobile and under the direction of the reactor operator in emergencies; the low population zone, where evasive or protective measures could be taken in case of hazardous releases from the plant, and the city distance.

6. The guides, in defining how the magnitude of the zones are determined, establish the principle that safety, in case of accident, depends not only on distances but also on protective featues of the facility: the engineered so called "consequences-limiting" safeguards of the facility itself. (Illustrative examples which were published with the criteria demonstrated the considerable extent to which average current practice, at the time the guides were written, placed dependence on containment and other safeguards in defining sites which were acceptable.)

7. Finally, the guides explicitly articulate two important principles: (a) the existence of the facility of unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials will be considered by the Commission in determining acceptability of a site, and (b) where unfavorable site characteristics exist, the proposed site may nevertheless be found acceptable if design of the facility includes appropriate and adequate compensating engineered safeguards.

   The line of practice in this country with respect to location of reactors and safeguard protection on reactors began to change shortly after the site guides were published. This change in practice was not all due to the site guides. For example, about this time, competitive prices for electricity from nuclear reactors began to come within "smelling range," and efforts to eliminate unnecessary costs, e.g., long transmission lines and large reactor sites increased."

However, the concept of the maximum credible accident, was not accepted around the world. At the 1964 U.N. "Atoms for Peace" Conference in Geneva, F.R. Farmer of the United Kingdom gave a paper entitled, "Reactor Safety Analysis as Related to Reactor Siting" (Farmer, 1964) in which he pointedly attached the concept of the maximum credible accident, particularly in any comparison of different reactor types. In particular, he emphasized the arbitrary selection of MCA which is inevitably involved. Farmer went on to emphasize the importance in the future of a comprehensive safety assessment and not merely a study of the consequences of a few selected major faults.

2.6 1961-1965 THE SITING OF LARGE LWR's - CONTINUED

We continue our examination of the reviews of the LWR construction permit applications in the time period 1961-65. As a brief aside, to give some flavor of the multiplicity of regulatory reviews on-going during this time period, we shall list the principal agenda items for two ACRS meetings.

At the 49th meeting, in which action was completed on the San Onofre Unit 1 construction permit application, the ACRS agenda also included the following:

(1) Review and prepare a report on a proposed power increase for the Yankee Nuclear Power Station;

(2) Review and prepare a report on the proposed power increase for Humboldt Bay Power Plant Unit No. 3;

(3) Review and prepare a report on conversion of the provisional operating license for the Saxton reactor to a full term license;

(4) Review the proposed operation of the boiling, nuclear superheater BONUS reactor; and

(5) Hear a presentation on operation of the Hallam reactor.

Thus, the ACRS (and the AEC Regulatory Staff) were busy with a large number of matters. In this historical review, we are singling out items particularly relevant to the evolution of the siting of large LWR's.

It is perhaps worth noting that there appears to have been little evaluation or emphasis placed in these siting reviews on the potential effects on public health and safety of radioactive contamination of water supplies due to an uncontained reactor accident.

At the 54th meeting, April 2-4, 1964, the ACRS agenda included the following:

(1) Review and prepare a report on proposed fuel irradiation test experiments in the Plum Brook reactor;

(2) Review and prepare a report on the application by the City of Piqua, Ohio, to assume operating responsibility for the Piqua reactor.

(3) Review and prepare a report on the proposed sea trials of the N. S. Savannah;

(4) Review a report on containment testing and send comments to the AEC;

(5) Review and write a report concerning a special mode of operation of one of the AEC's large production reactors at the Savannah River Plant, South Carolina, and;

(6) Discuss a large number of other topics with the Regulatory Staff.

At its 50th meeting, October 10-11, 1963, the ACRS had a considerable discussion concerning the proposed Ravenswood reactor; this will be discussed in some detail later in this chapter.

At its 53rd meeting, February 13-15, 1964, the ACRS completed its review of the proposal for construction of the 1473 MWt Connecticut Yankee PWR. The ACRS report of February 19, 1964, is on the following page.

The Connecticut Yankee letter continues the emphasis on containment and on engineered safety features to reduce the fission product concentration in the containment following the postulated release of the MCA.

A safety injection system (ECCS) is mentioned; however, little evaluation was made of its design basis or efficacy during the review. The allusion to a potentially large containment design pressure, if Zircaloy cladding is used instead of stainless steel, arises from the concept that a large fraction of the Zircaloy would undergo metal-water reaction in a core melt, releasing heat and hydrogen (which could burn, adding more heat). As studies showed some years later, this represented a really incomplete evaluation of metal-water reaction problems.

This letter is the first to call out the requirement for study of the control rod ejection accident, a requirement which led to design changes in large LWR's, either to limit the reactivity worth of control rods (and hence keep the resulting power rise tolerable) or to add an additional mechanical restraint to control rod ejection, thereby making the probability acceptably low (the approach taken in BWR's).

The next ACRS review bearing directly on siting criteria and the increasing substitution of engineered safety features for distance came at the 56th meeting, July 9-11, 1964, when the Committee reported favorably on the proposal of the Los Angeles Department of Water and Power to construct a large PWR at the Malibu site (formerly called the Corral Canyon site). The ACRS letter of July 15, 1964 follows.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

February 19, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON CONNECTICUT-YANKEE ATOMIC POWER COMPANY

Dear Dr. Seaborg:

At its fifty-third meeting, February 13-15, 1964, the Advisory Committee on Reactor Safeguards reviewed the general design of the proposed 1473 MW(t) pressurized water reactor to be constructed at the Haddam, Connecticut site. The Committee had the benefit of a subcommittee meeting on December 13, 1963 with the applicant and its contractors. In addition, the documents referenced below were provided the Committee. The Committee had discussions with the applicant and representatives of Westinghouse Electric Corporation, Stone and Webster Corporation, and with the Regulatory Staff and its consultant from the U. S. Geological Survey.

In its previous report to the Commission on February 6, 1963, the ACRS pointed out that the Haddam Site did not meet the present site distance criteria, and hence reliance must be placed upon engineered safeguards to reduce off-site exposures in the unlikely event of a serious accident. Because of otherwise favorable site location, low population density and meteorological characteristics, a reduction factor of about 6 in addition to that provided by containment is needed to bring the potential dose from a maximum hypothetical accident to guideline limits.

The proposed design has the reinforced concrete containment described below. The design includes the following additional engineered safeguards: an internal recirculation containment spray system; a continuously operated air recirculation system with cooling, involving four independent units; and a filtration-adsorption unit on each of the recirculating systems which can remove halogens and other fission products. The plant is also to be provided with a safety injection system having three independent pumps and a large supply of borated water.

A reinforced concrete containment vessel with a steel inner liner is proposed. Containment leakage is specified to be not more than 0.1% per day and penetration leakage rates will be monitored. The proposed containment is designed for the use of stainless steel clad fuel elements in the reactor. If, for instance, Zircaloy cladding is used, it may be necessary to increase the design pressure or volume of the containment.

The reactor is to be a pressurized water system of proven operating characteristics with cluster type control rods. The use of four separate steam generator loops decreases the significance of a major primary coolant line rupture. Details of the reactor physics behavior will be resolved during the design phase. The Committee believes the possibility and effects of control rod ejection deserve further evaluation and documentation.

The Committee considers that the proposed engineered safeguards provide the necessary redundancy and reliance to assure reduction of releases to below guideline values in the unlikely event of a reactor accident. The filter-adsorber systems, while not finally selected as to performance characteristics, should be protected against steam and water releases, and may require capability for various forms of halogens. These factors should be reliably established before the facility operates.

It is the opinion of the ACRS that the proposed engineered safeguards, including the containment as proposed, will provide the necessary protection in the unlikely event of an accident. On this basis, the ACRS believes that there is reasonable assurance that the general type of reactor proposed for the Connecticut Yankee Atomic Power Company, including engineered safeguards, can be constructed at the Haddam Site with reasonable assurance that it can be operated without undue hazard to the health and safety of the public.

Dr. T. J. Thompson did not participate in this review.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References Attached.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 15, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON CITY OF LOS ANGELES - MALIBU NUCLEAR PLANT - UNIT NO. 1

Dear Dr. Seaborg:

At its fifty-sixth meeting at Brookhaven National Laboratory on July 9-11, 1964, the Advisory Committee on Reactor Safeguards reviewed the proposal of the City of Los Angeles to construct and operate a 1473 MW(t) pressurized water reactor, Malibu Nuclear Plant - Unit No. 1, at Corral Canyon, twenty-nine miles west of Los Angeles. The Committee had the benefit of discussions with representatives of the Department of Water & Power of the City of Los Angeles, Westinghouse Electric Corporation, Stone & Webster Engineering Corporation, the AEC staff, their consultants, and of a Subcommittee meeting on June 18, 1964. The Committee also had the benefit of the documents listed below.

The proximity of large population centers and the probable growth of population in the vicinity of the proposed reactor site require dependence on engineered safeguards to limit the consequences in the unlikely event of a major credible accident. For this reason, safeguard provisions more extensive than those normally employed in nuclear power reactor plants must be provided in lieu of the distance factor to protect the public.

The applicant has proposed as engineered safeguards a novel containment structure intended to prevent any leakage to the environment, and additional features consisting of:

1. A reinforced concrete containment structure.

2. A containment volume spray system, and

3. An emergency borated-water injection system.

The total containment feature of the building is to be achieved by providing two complete steel liners separated by a layer of porous concrete. The space between the liners will be maintained at a sub-atmospheric pressure by continuously pumping back air to the containment volume. An air recirculating and cooling system is required to remove any heat that is generated within the containment volume. Power and water to assure operation of these systems under all conditions must be provided.

Detailed design of the reactor core has not been established yet, but the general features will be similar to those of other nuclear plants proposed for construction by the same nuclear contractor, and expected to be tested in operation prior to completion of the Malibu plant. Nuclear reactivity coefficients are expected to be negative in this reactor. The probability and effects of control rod ejection require further evaluation. The applicant has suggested several possible means of limiting the consequences of such an accident, and the Committee believes that this question can be resolved satisfactorily during the design stage.

Although stainless steel cladding is planned for the first core, it is anticipated that zirconium alloys may be used in future cores. Complete information on the effect of a possible zirconium-water reaction on the course of accidents is not available. Hence, further review will be needed prior to use of zirconium alloy clad cores.

The Committee was informed that the geology of the site was suitable for the proposed construction. It was reported that no active geological faults are present at the site. Grading of the canyon slopes is proposed to ensure that potential landslide motion does not present a hazard to the plant. It is proposed that critical structures be designed for a suitable response spectrum associated with an earthquake which has a maximum acceleration of 0.3 g. occurring when the containment is under the pressure associated with an accident. The resulting stresses will not exceed 80% of the minimum yield value. Components within the building will be designed to withstand 0.3 g. acceleration acting simultaneously in horizontal and vertical plants.

The ability of the plant to withstand the effects of a tsunami following a major earthquake has been discussed with the applicant. There has not been agreement among consultants about the height of water to be expected should a tsunami occur in this area. The Committee is not prepared to resolve the conflicting opinions, and suggests that intensive efforts be made to establish rational and consistent parameters for this phenomenon. The applicant has stated that the containment structure will not be impaired by inundation to a height of fifty feet above mean sea level. The integrity of emergency in-house power supplies should also be assured by location at a suitable height and by using water-proof techniques for the vital power system. The emergency power system should be sized to allow simultaneous operation of the containment building spray system and the recirculation and cooling system. Ability to remove shutdown core heat under conditions of total loss of normal electrical supply should be assured. If these provisions are made, the Committee believes that the plant will be adequately protected.

The applicant has proposed to deny entrance to the containment while the reactor is operating. This mode of operation does not permit frequent surveillance of equipment and prompt detection of incipient defects. Operating experience at other power plants has demonstrated the value of accessibility for inspection. The Committee suggests that the applicant reconsider this question and explore design modifications which will allow entrance without violating the containment integrity.

As the Committee has commented in its earlier letters, the hold-up of routine gaseous and liquid releases may be necessary during unfavorable conditions. In this connection, it will be necessary to conduct additional pre-operational meteorological and oceanographic survey programs.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be suitably dealt with during construction, and that the proposed Malibu Nuclear Plant can be constructed with reasonable assurance that it can be operated at the site without undue risk to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References Attached.

The Malibu site probably had a higher projected population density than any reactor of that size previously approved. And the applicant had proposed use of the "zero leakage" Stone and Webster double containment concept described above in the IAEA Symposium paper by Johnson (1962).

The ACRS report again places considerable emphasis on limiting the consequences of the MCA. The report also calls attention to several other matters including the following:

(1) Need for analysis of the control rod ejection accident and the implication on containment pressure of the use of zirconium clad fuel, as in the Connecticut Yankee letter.

(2) Seismic design considerations, including tsunamis.

(3) The need for emergency, in-house power supplies (at this stage in the evolution of safety requirements, to enable containment cooling and shutdown heat removal, although not for the loss-of-coolant accident).

At the 57th meeting, August 24-26, 1964, the ACRS concluded its review of the construction permit application for a 1600 MWt BWR at Oyster Creek, New Jersey, a site having intermediate population density characteristics. The ACRS report of August 28, 1964 follows on a separate page.

The letter makes no mention of the emergency core cooling system (ECCS) proposed for this reactor. However, the minutes of the meeting indicate that there was discussion of the fact that the reactor included a duplicate core spray arrangement to limit melting of the core in the event of a loss-of-coolant accident. Also, according to the minutes, Mr. Case of the Regulatory Staff indicated that a possible zirconium-water reaction and hydrogen explosion was the chief remaining problem:

> The probability is believed low because a maximum credible accident and failure of the safety injection system must occur first. Also, it is questionable if the concentration of hydrogen would give an explosive mixture. An inert atmosphere, that is nitrogen in the confinement system, might tend to prevent this.

ACRS member Silverman is shown as having noted that, if the meltdown prevention equipment operates satisfactorily, the reactor could be located almost anywhere; however, he reminded the Regulatory Staff that the spray system in the SL-1 reactor was made inoperative by the accident.

It is difficult to tell from the minutes how important the availability of the ECCS core spray was to the ACRS decision-making process. The minutes say that "because of the great dependence being placed on the core spray to limit fission product release, the Committee cautioned that these sprays must be quite reliable."

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

August 28, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON OYSTER CREEK NUCLEAR POWER PLANT OF JERSEY CENTRAL POWER AND LIGHT COMPANY

Dear Dr. Seaborg:

At its fifty-seventh meeting, on August 24-26, 1964, the Advisory Committee on Reactor Safeguards considered the proposal of the Jersey Central Power and Light Company to construct and operate a nuclear power plant on Oyster Creek in New Jersey. This will be a 1600 MW(t) boiling-water type reactor with pressure absorption containment.

The Committee had the benefit of an oral presentation by representatives of the applicant and consultants and contractors, advice by the AEC Staff, and the reports cited. A Subcommittee meeting was held at the site on May 1, 1964, and a further Subcommittee meeting was held in Washington, D. C. on August 7, 1964.

Many details of the proposed design have not yet been completed. The applicant is continuing to study the limitation of maximum reactivity of individual control rods and the design of the reactor protection system. The following additional points should be given examination and consideration:

(1) Under some credible accident conditions, the dry well and absorption pool may require provisions for additional heat removal.

(2) In the unlikely event of a melt-down accident, a zirconium-water reaction may produce hydrogen. Provision should be made to prevent any hydrogen-oxygen reaction that would disrupt the containment.

(3) The adequacy of the reactor protection system when operating at partial recirculation flow rates should be established.

Estimates made by the applicant on halogen retention by absorption in water and by plate-out are based on limited data, and the consequences of the unlikely accident may be more severe than estimated. However, the Committee believes that more conservative assumptions would not make the proposal unacceptable.

With due regard to the above comments, the ACRS believes that the proposed reactor can be constructed at the proposed location with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References:
1. Part B, Preliminary Safeguards Summary Report, Application to the United States Atomic Energy Commission for Construction Permit and Operating License, Oyster Creek Nuclear Power Plant Unit No. 1, Jersey Central Power and Light Company, undated, received April 2, 1964.
2. Amendment No. 2, Application Reactor Construction Permit and Operating License, Oyster Creek Nuclear Power Plant Unit No. 1, Jersey Central Power and Light Company, dated June 26, 1964, with enclosures.

There did not exist the experimental data and analytical methodology at that time to really analyze the efficacy of the core spray system. And, we will find that later in the same year, when the ACRS wrote a letter on the use of engineered safety features to balance against a reduction in distance to populated centers, the Committee was reluctant to give "credit" for emergency core cooling systems.

What is perhaps interesting is that there had by now been a succession of reviews of rather large LWR's, but none of these were operating, none were fully designed, and none had received detailed regulatory analysis and evaluation. So that the actual amount of information available for the decision-making process in 1964 was quite limited, compared to what would be available in the 1970's.

At the 58th meeting, October 7-10, 1964, the ACRS completed its review and wrote a report favorable to the construction permit application for the 1538 MWt BWR at Nine Mile Point. This reactor was at a relatively less populated site and was rather similar to the Oyster Creek reactor earlier approved.

Several months earlier, the Atomic Energy Commission had asked the ACRS to put into a report the manner in which it was permitting engineered safety features to be substituted for distance in meeting Part 100 regulations. At the 59th meeting, November 12-14, 1964, the ACRS prepared a letter entitled "Report on Engineered Safeguards" (issued November 18, 1964). The things which received approval were: containment and certain confinement systems; the pressure suppression method; containment building sprays to reduce containment pressure following a LOCA; heat-exchange methods of limiting containment pressure; containment air cleaning systems, following a LOCA.

It was stated that "core spray and safety injection systems cannot be relied upon as the sole engineered safeguards, that they might not function for several reasons such as severed lines and low water supplies. Nevertheless, prevention of core melting after an unlikely loss of primary coolant would greatly reduce the exposure of the public. Thus, the inclusion of a reactor core fission product heat removal system as an engineered safeguard is usually essential."

The body of the report notes the need for adequate emergency power sources for an ECCS.

What seems to be lacking from the report is any identification of engineered safety systems for residual (fission product decay) heat removal from a core in which the power has been shutdown, although attention is given to methods of removing heat from the containment. Thus, the period from 1960 to the end of 1964 represented a time during which quantitative siting criteria evolved and then were relaxed as the acceptance of engineered safety features instead of distance became part of the regulatory process. It was also a period in which there was a strong beginning of looking at other things in addition to the MCA. However, there was not a comprehensive, systematic look at all (or most) accidents which might, in fact, represent a threat to containment and have consequences far exceeding Part 100. The approach developed in the period up to 1964 in large part represents the siting approach used by the Regulatory Staff to this day. It is necessary to postulate the release of the bulk of the noble gases and iodine into the containment, and to have a combination of containment features and site characteristics so as to meet Part 100. This is quite apart from the fact that for reactors above about 300 MWt, it now appears likely that containment failure will occur in the event of large core melt, leading to release of very many fission products.

Before going on to the review of the Dresden 2 BWR in late 1965, it is of interest to look in some detail at the preliminary application for consideration of a large LWR at the Ravenswood site, which was essentially in the heart of New York. We will also discuss some other aspects of the metropolitan siting question which arose during this general time period.

2.7 THE RAVENSWOOD REACTOR AND METROPOLITAN SITING

On December 10, 1962, the Consolidated Edison Company submitted an application for a construction permit for two reactors to be located at the Ravenswood site in New York City. These two pressurized water reactors each were to have thermal power output of 2,030 megawatts; the estimated completion date was between October 1, 1969 and October 1, 1970. In the construction permit application the summary states that two containment vessels prevent the release of any radioactive material to the surrounding area in the event of an accident. There is to be double containment for each of the reactor systems and for the spent fuel storage facility. These containment structures consist of two steel shells completely surrounded by 5 1/2 feet and 2 1/2 feet of concrete, respectively, with previous concrete occupying the annulus between the inner and outer steel membranes. This is like the containment described in the paper by Johnson at the IAEA Symposium (1962). Double containment is used so there will be no leakage of radioactivity even under the worst conceivable accident conditions. A research and development program is outlined, and it is stated that this program will be completed prior to January 1, 1967. The containment also includes a spray system to reduce the pressure within the containment vessel in the event of an accident, and a pump-back system by which the pressure between the two steel membranes is maintained slightly sub-atmospheric by pumping air from this region into the interior of the reactor containment. A safety injection system is included, which supplies borated water to the reactor core following a LOCA. The design basis is stated to be a rupture of the largest connecting pipe to the main pipes in the primary system. The worst conceivable accident is defined to be caused by the instantaneous release of the entire contents of the primary cooling system into the containment with 100% of the core melted. From this event the hazards to the environment are stated to be less than that of 10 CFR Part 20; the gaseous activity discharge will not exceed non-occupational air tolerance levels of 10 CFR Part 20 at the top of the stack. Relatively less information was stated by the Regulatory Staff to be available concerning the design of this reactor than for other reactors recently reviewed. The site, which was in the Burrough of Queens, is 8.7 acres in size, bounded on the west by the East River, on the north by 36th Avenue, and on the east by Vernon Blvd. The minimum distance from the reactor containment to the fence around the site boundary is approximately 90 feet. The population within a circle of radius one-half mile was estimated to be 19,000 at night, 28,000 during the day; within a circle of 5 miles radius, it was estimated to be 3 million people at night and 5 1/2 million during the day. Although proposals had been made previously for the siting of relatively small reactors within small cities, for example, the Jamestown reactor and the Piqua reactor, this, by far, was the most difficult reactor site proposed by the Atomic Energy Commission.

The Regulatory Staff began to analyze the reactor and site with the information that it had. In view of the fact that the site clearly would not meet the normal conditions of the site criteria with regard to exclusion distance and low population zone, or even the distance to a population center, the Staff decided to see whether the site could be made acceptable on a so-called engineered safeguards basis, this prior to looking at accident analysis, etc. On August 9, 1963, the Regulatory Staff sent out a set of questions, 13, in total, in which it requested additional information from the Consolidated Edison Company. The bulk of these questions related to the design of the containment, the ability to measure leakage rates in the containment, the way in which penetrations through the containment could be monitored, and things like filter systems which could remove radioactivity from the containment. The Consolidated Edison Company responded to these questions in a letter dated November 14, 1963. It was proposed to use redundant systems for the safeguards features employed in the Ravenswood plant; and the use of a single-failure-criterion was planned in order to provide adequate reliabiity. Also, mention was made of the need in the layout of the plant to give careful consideration to insure that an initiating accident would not impair operability of safety components; for example, proper separation of piping and cabling would be used, and it was stated that the components themselves would be designed to operate under the temperatures and steam-air conditions which would exist in the plant containment following a loss-of-coolant accident. It was not planned to protect against gross failure of the reactor pressure vessel, pressurizer, or steam generator. While there was frequent reference in the applicant's submittal to the containment design, and to the engineered safety features which would keep the dose to the public below Part 20, given the fission products corresponding to full scale core melt, nowhere was mentioned the possibility that full-scale core melt might automatically lead to failure of the containment.

On September 25, 1963, the Regulatory Staff sent a report to the ACRS concerning the Ravenswood Reactor in which it outlined the features of the plant as of that time. The report described the ways in which the site departed greatly from what was permitted by the AEC site criteria, and discussed an approach in which they (the Staff) would try to see whether it was possible that a containment scheme could be devised which would permit the site to yield acceptable doses for the MCA. The Regulatory Staff announced their intention first to complete an evaluation of proposed engineered safeguards systems, and then pursue further analysis of accident consequences; and the results of this phase of the evaluation were to be presented for review by the ACRS in time for the December, 1963 meeting of the Committee. An ACRS Subcommittee meeting was held with the Consolidated Edison Company on September 11, 1963, concerning the Ravenswood reactors. It is not possible from the meeting minutes to tell what was in the mind of the members. ACRS member Gifford did make a comment to the applicant that "since the site is lousy, questions would center on containment and engineered safeguards." At the meeting, the Consolidated Edison Company stated that they had specified that radiation from the proposed plants under any and all conditions must not exceed 10 CFR Part 20. This statement was not changed when ACRS member Osborn indicated that reactor vessel rupture was a situation for which there was no protection, or when ACRS member Rogers asked about the adequacy of missile shielding.

The sources of electrical power for the plant was stated to be 4 independent outside sources. When Dr. Gifford asked whether failure of the steam generator was included in setting the containment design pressure in connectin with a LOCA, Dr. Weisemann of Westinghouse stated it was cheaper to support the pipes so that primary system rupture would not lead to a secondary system failure.

The bulk of the discussion during the meeting related to the possibility of providing containment with the very low leakage rates that Consolidated Edison was seeking.

Another ACRS Subcommittee meeting was held on October 21, 1963. The preliminary discussion by the Subcommittee members prior to their meeting with the Regulatory Staff indicated that the primary emphasis was on what degree of assurance could be credited to the proposed zero-leak-rate design of the plant. It was noted that many paths exist by which a double containment scheme might be bypassed, and that even a small release might be intolerable at this site. At least from the minutes, there does not seem to have been much discussion about accidents in which both core melt and containment rupture occur. (In 1977, Dr. H. Kouts, an ACRS member during the Ravenswood review period, recalled that the principal problem with the Ravenswood site was its vulnerability to "small" accidents.) At this meeting, Consolidated Edison noted the design basis was that no single component maloperation will cause damage to the reactor core, and no single component gross failure will impair the ability to contain fission products within the plant. Presumably the reactor pressure vessel and certain other components were not included in that list.

ACRS member Ergen summarized his concern over location of the reactor at this site by stating that it is unrealitic to design and operate a plant with assurance that required very low leak rates can be met in the event of an accident. He had calculated that the required leak rate would have to be of the order of 10^{-3}% per day of the contained volume, in order not to produce more than 300 rem to the thyroid to 4,000 people around the plant. It would be easy to have any number of "Achilles heels", like a minor steam generator tube leak.

The minutes of the 50th meeting, October 10-11, 1963, show there was discussion of the Ravenswood reactor and record some interesting points of view*. ACRS member Osborn is stated as considering this plant to have had a more comprehensive study at this stage than is usual; to him the general ideas of design, construction, and safety are good. Member Thompson feels that the ACRS could approve such a plant, provided it has all the safeguards now proposed. On the other hand, member Rogers would require the applicant to guarantee that no accident will affect the public, before ACRS approval is granted. He has doubts about the effectiveness of the secondary containment.

In the discussion between the ACRS and the Regulatory Staff, Mr. Lowenstein, the Director of Licensing, indicated that computer studies showed that even if all the engineered safeguards operate, leakage must still be limited to the order of 10^{-4} cu. ft./min. The Regulatory Staff sees this as impossible with the design proposed. Mr. Lowenstein indicated the Regulatory Staff would now reject the application on the basis of the proposal entailing too much of an advancement in reactor technology for this location.

At the 51st meeting, November 7-8, 1963, the ACRS had some discussion of Ravenswood in an executive session. The bulk of the opinion seemed to be that the Regulatory Staff was going to reject the site. The concensus of the Committee seemed to be that more time should be given for review of the matter, but this was not unanimous.

As an interesting aside, at the 51st meeting, the ACRS wrote a letter report to Mr. Leudecke, the General Manager of the Atomic Energy Commission, reviewing the reactor safety research program. In this report the ACRS says:

> The Committee believes that it is of primary importance to determine to what extent engineered safeguards can be relied on in relaxing reactor site restrictions.
>
> In the light of present knowledge, it seems unlikely that general principles will render incredible the possibility that high power reactors can have large power excursions or that they can have substantial core meltdown. Therefore, it must be expected that the safety analysis for locating and designing nuclear reactors will continue to assume such accidents to be possible, even if only remotely so.

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*It must be recognized that the minutes do not represent a comprehensive (or necessarily even an accurate) summary. There were long delays in their preparation at that time and they were considered to be less than satisfactory by the ACRS itself.

The bulk of the rest of the letter relates to questions of how much of the fission product inventory would escape from the fuel to the reactor vessel to the containment, and how one could decide whether the fission products would plate out in the reactor containment, etc. There is also mention of the need for research on the probability of gross rupture of primary pressue vessels and other pressurized components, and that information is needed on methods to protect containment from possible missiles.

Apparently, the ACRS felt that large scale core meltdown could not be ruled out at this time; however, as of the November meeting, they had not, as a matter of policy, recommended against the Ravenswood reactor. It is not clear from the minutes how the possiblity of releases very much larger than Part 100 in the heart of a large population center was affecting the review. It is clear from the discussion that reactor vessel failure and other modes of containment failure were considered. It is not clear that the ACRS envisaged that core meltdown would automatically lead to containment failure for a reactor of the size of Ravenswood.

Between the 51st and 52nd meetings, Consolidated Edison withdrew its application for consideration of the Ravenswood reactor.* It is not possible to reconstruct the overall position of Consolidated Edison with regard to the safety of the reactor. The minutes indicate that they considered that, for anything that was "credible", they would be protected. They did not indicate any concern that building those reactors at that time in what was essentially the heart of New York posed an undue risk. It is also not clear what the basis for their judgment was.

Although it was withdrawn, the Ravenswood application forced the regulatory groups to consider the question of metropolitan siting. And in a sense, the application was one form of pressure on the regulatory groups to see in what way, if any, metropolitan siting of reactors could be approved. As we shall see, the industry continued to propose sites involving large surrounding population, not quite like that of Ravenswood but still considerably beyond what had been accepted before.

The minutes of the 52nd meeting, January 9-10, 1964, show that the Committee once again discussed reactors in populated areas even though the application to the Ravenswood reactor had been withdrawn only a few days before. Chairman Kouts asked if the ACRS wished to prepare a letter, for example a letter drafted by member Thompson, regarding the location of

*The ACRS minutes have no record of any negative (oral) opinion concerning the Ravenswood site having been forwarded to Consolidated Edison. However, it is the author's understanding from recent discussion with a senior member of the Regulatory Staff that Consolidated Edison was told by the AEC that their application would receive an unfavorable response, if it were not withdrawn.

reactors in populated areas. Briefly, this letter considered such sites acceptable provided there are (adequate) engineering safeguards and a reactor of the same type and power level has operated safely elsewhere. However, some members felt the Committee should wait, giving advice if and when a particular case is before the Committee. The official reason given for withdrawal of the application of the Ravenswood reactor by Con Edison was the availability of cheaper power from Labrador, 1100 miles away.

A very interesting sequence of events concerning reactor siting began in February 1965. At the special ACRS meeting held on February 6, 1965, Mr. Harold Price, who was Director of Regulation, reported to the ACRS that the Boston Edison Company had made preliminary studies of 6 reactor sites; the preferred one was near Quincy and presumably was in a rather highly populated area. Commission advice on siting policy regarding large reactors near or in big cities was desired by the utility.

At the 62nd meeting, March 11-13, 1965, Mr. Osborn reported that he, together with representatives of the Regulatory Staff and the Bechtel Company, had visited four sites in the Los Angeles area which are proposed for dual purpose (desalination and power) reactors. The power level would be in the neighborhood of 3000 MWT. The minutes of this meeting also showed an executive session of the ACRS in which there was a discussion of a draft position by the Regulatory Staff, which had been formulated around the proposed Boston Edison reactor and which was again concerned with metropolitan locations for reactors. The concensus of the Committee seemed to be that this was a very important matter and that no hasty conclusions should be drawn. Siting of each reactor was seen as an individual case.

Discussions ensued with Mr. Price in which he related the results of recent meetings regarding the public acceptance of large reactors. He said that the proposed reactor near Boston and the possible reactivation of the Ravenswood case, both reactors with powers of the order of 3000 megawatts, were pressing the Regulatory Staff.

The minutes indicate that a complex group of factors entered into the decision-making process, including the timing and nature of any announcement.

According to Mr. Price,

AEC approval of reactors in large cities could be argued as being in conflict with AEC support for the Price-Anderson arrangement. When the Ravenswood project was withdrawn, the Regulatory Staff was in the midst of a study on siting of large reactors. Although no such large reactors had operated, the design engineers claimed that safe designs can be built. Dr. Beck of the Regulatory Staff believed that the engineering ideas on engineering safeguards were quite good and will be proven in years, but that their reliability now is questionable.

ACRS members asked the Regulatory Staff for an opinion on what degree of redundancy in engineering safeguards would be acceptable, but received no answer. Dr. Doan of the Regulatory Staff* considered isolation of reactors from a city as allowing much more maneuverability in operations, latitude in power levels, and in requirements of engineering safeguards. ACRS member Okrent judged that only a few miles of isolation is of little use following large accidents. Mr. Price believed that assuring 100% operability of engineering safeguards is impossible. But ACRS member Silverman considered certain items, e.g., the containment, the filtering and the air cleaning, to be quite reliable.

The ACRS held a Subcommittee Meeting on Siting of Large Power Reactors in Metropolitan Centers on March 20, 1965. According to the minutes of this meeting, the draft Regulatory Staff paper concludes that "the public interest can best be served by continuing to exclude large cities as permissible locations for nuclear power plants". ACRS member Osborn noted that the Staff position appears to reflect a concern more with the smaller, more probable accidents, in addition to the usual MCA concept (e.g., fuel handling or fuel shipment accidents). Also, the effects of natural disasters appeared to be involved in the thinking of ACRS members.

ACRS member Etherington noted that there is no design which will eliminate the question of "operator error" with possible violation of containment, etc. Member Rogers stated that the location of a reactor (city versus county) make little difference if a very large accident (e.g., breached containment with fission product release) were to occur. He concluded, therefore, that the smaller, more probable accident is the situation of concern.

Member Gifford noted that a "formal" moratorium on metropolitan siting would discourage any new developments or improvements in reactor safety over the designs that were then available.

During the ensuing discussions with the Regulatory Staff, Mr. Price said he is trying to avoid a drawn-out, detailed design review which finally ends in a policy decision to turn down the application. Member Etherington noted that a strict interpretation of the Part 100 siting guide would preclude the siting of reactors in cities without any further action on the

*A former ACRS member.

part of the AEC. Mr. Price agreed but indicated such an interpretation would need an endorsement from the Commission itself.

Subcommittee members expressed a concern that a formal moratorium would stifle any further development in the field of engineered safeguards. Member Rogers noted that routine operation of reactor plants alone would not provide information about performance of engineered safeguards under accident conditions. Dr. Beck pointed out that even routine operation, testing and maintenance of engineered safeguards were not then well developed, and operation for several years would help to identify and correct system deficiencies.

Dr. Beck confirmed that the Regulatory Staff was concerned about smaller accidents than the MCA, since larger accidents would have very serious consequences no matter where the location.

Several Subcommittee members suggested that applicants should be advised of appropriate criteria on the basis of a case-by-case review. Mr. Price maintained, however, that this would leave applicants in a state of confusion and uncertainty.

Dr. Beck suggested that a logical set of criteria should be developed <u>before</u> the door was opened, since industry was eager to move into cities with facilities of the Mualibu/Oyster Creek designs. Member Rogers agreed that utilities were not likely to propose any additional safeguards which were not required by the AEC.

In response to a question from Dr. Beck, member Osborn stated that the review of Ravenswood had not been completed but that there was no indication that it would have been turned down on the basis of a policy decision. It seemed generally agreed that the Malibu/Oyster Creek designs were not yet acceptable for use in densely populated areas, and that more stringent criteria must be developed for city reactors.

The matter of metropolitan siting was discussed at considerable length in the ACRS Special Meeting, March 26-27, 1965. Most of the ideas mentioned above entered into this discussion. Member Etherington recalled that no large reactors in cities were contemplated at the time of development of the Reactor Site Criteria (Part 100). Mr. Etherington observed that where the probability of an accident and its consequences can be obtained, the reactor designer can evaluate alternative approaches to limit the consequences or avoid the event. However, if the accident is of extremely low probability and the consequences very large, no such analytical approach is open (to designer or Regulatory Staff). In summary, Etherington felt the problem lies with the exceedingly low probability - high consequence accident.

According to Mr. Price, the Commissioners were not inclined at this time to take any stand against metropolitan reactors. However, the Regulatory Staff needed a public posture, and had to respond formally or informally to recent inquiries from utilities and reactor vendors.

Following the Special Meeting, March 26-27, 1965, the ACRS released for the information of the Commissioners a draft report on metropolitan siting, which is quoted below.

"REPORT ON SITING POWER REACTORS IN METROPOLITAN CENTERS

The Advisory Committee on Reactor Safeguards has been informed by the Director of Regulation that representatives of the nuclear power industry have, in recent weeks, visited him to explore the possibility of locating large power reactors in metropolitan areas. This letter is in response to your inquiry as to the views of the ACRS on this subject.

This subject was discussed with members of the AEC Regulatory Staff during the 62nd ACRS meeting on March 11-13, and the Special ACRS Meeting on March 26 and 27, 1965. In addition, a subcommittee met with the Regulatory Staff on March 20th. The ACRS offers the following comments on the question of locating large power reactors in metropolitan areas:

1. The engineering of reactor safety has been a process of evolution. Much has been accomplished, more remains to be done. The larger power reactors now under construction or described in current license applications represent a large step in this process of evolution. However, considerable further improvements in safety are required before large power reactors may be located on sites close to population centers. None of the large power reactor facilities now under construction or described in current license applications are considered suitable for location in metropolitan areas.

2. A flexible position with respect to locating reactors close to cities should be maintained. License applicants should be encouraged to use imagination and to employ improved provisions for safety. A suitable channel for the early consideration of new facility concepts should continue to be available.

3. Designers should be encouraged to develop engineered safeguards of extremely high reliability and with provision to assure that such reliability can be demonstrated at all times.

4. The quality of operation, maintenance and adminstrative control, upon which dependability of engineered safeguards relies, must be further improved.

5. Guidance for designers and operators should be developed for locations of reactors in cities.

In connection with the last item the ACRS is considering the following tentative points:

(1) The design goal for reactors being considered for metropolitan use should be the elimination of any possibility of a severe reactor accident.

(2) It would seem prudent to operate in metropolitan areas only reactors of a proven type, which do not represent a large extrapolation in power, involving radical changes in reactor design from reactors already in service. In other words, reactors in metropolitan areas should closely duplicate reactors with demonstrable and favorable operating experience.

(3) It must amount to a practical certainty that under no circumstances will significant amounts of fission products reach the public. Provisions taken should include containment of the refueling operation, spent fuel storage area, and radioactive waste.

(4) In order to assure a reliable containment, it is necessary to establish in some way an upper limit to the energy release in any possible accident. This energy release should include nuclear excursion energy, stored thermal energy and chemical reaction energy.

(5) The containment should be adequately protected from missiles both from within and without, including those arising from the disintegration of equipment.

(6) Reliance should not be placed on valves to effect isolation of normally operating ventilation systems.

(7) A design goal for instrumentation and control systems including all electronic and mechanical devices should be that all safety systems are fail-safe including consideration of effects of fire, steam, and other possible environments.

(8) Improved reliability of emergency power supplies appears required. Or, in lieu of that, the facility should require no emergency electric power.

(9) The possibility of simultaneous independent failures should not be neglected in evaluating engineered safeguards.

(10) Primary reliance for safety should not be placed on procedural control methods."

This draft letter was never issued; however, it was discussed in detail with members of the Regulatory Staff, and with the AEC Commissioners at the 63rd meeting, May 13-15, 1965. At the May meeting, the Commission indicated a desire to avoid any interpretation which might preclude large reactors near large cities. The issuance of guidelines for the siting of such reactors seemed to be favored by the Commissioners.

The ACRS draft stated that only reactors of proven design were suitable for city locations. AEC Chairman Seaborg commented that he hoped the ACRS meant an applicant could anticipate favorable operating experience of a reactor at a remote site prior to operation of a metropolitan reactor; this would allow construction to proceed and save time, but with the applicant accepting a risk.

Some of the thoughts included in this draft letter on metropolitan siting were presented publicly in the testimony of the ACRS to the Joint Committee on Atomic Energy in the hearings on extension of Price-Anderson indemnity legislation, June 22, 1965. What is stated in this ACRS testimony is the following:

> The engineering of reactor safety is in a process of evolution; much has been accomplished, more remains to be done. The larger power reactors now under contruction described in current license applications represent a large step in the process of evolution; however, considerable further improvements in safety are required before large power reactors may be located on sites close to population centers. None of the large power reactors now under construciton or described in current license applications is considered suitable for location in metropolitan areas. To put the matter in a different way, the devices and safeguards that prevent all accidents, large or small, must be made even more reliable than they are more fool-proof. The questions to be settled are complex ones, and resolution would depend on the nature and details of each proposal. It also appears that novel reactor systems, or reactors of considerably higher power level than previous ones, should not be operated in population centers.

Depending on how one reads the draft letter, one might or might not read into it a moratorium or semi-moratorium on metropolitan siting. Nevertheless, the review of the proposed Boston Edison BWR for a relatively populated site continued for several months. One finds in the information submitted by the applicant, statements such as the following:

> The integrated dose 600 ft. from the reactor or any point beyond shall not exceed 2.5 rem to the whole body and 30 rem to the thyroid in the event of the design basis, coolant-loss accident, coincident with 100% core melt.

One cannot ascertain unequivocally from the specific written material whether the applicant thought that the containment would remain intact in the event of core melt and that, in fact, the doses that he was calculating were valid for that event; or that core melt was being treated as a generalized accident in which fission products are postulated to be released to a containment which remains intact. However, in more than one place the statement is made that the containment will be designed on the basis that significant core melting and metal water reactions occur; so a logical deduction by the reader is that the applicant is proposing a containment system that can handle core melt.

The ACRS minutes of a meeting between the Boston Edison Group and the AEC Division of Reactor Licensing, September 20, 1965, give further information on the proposed reactor. It is stated that the plant is basically an Oyster Creek type reactor with double pressure-suppression containment and several other improvements. These improvements include use of internal jet pumps for reactor circulation. This design provides a secondary shell around the core which can be reflood following a LOCA. Both core spray and core flooding systems will be provided. Duplicate spray systems will be provided for containment cooling following a LOCA.

A rod drop velocity limiter will be provided. Rod drives will be supported so that a rod cannot be ejected if a drive nozzle fails. Flow restrictors will be provided in main steam lines to limit steam flow to 200% in the event of a steam line break.

The rod drop velocity limiter and the support of rod drives to prevent ejection were not basically new to this plant, since questions concerning reactivity accidents had been raised earlier in connection with review of Oyster Creek and some other plants. However, this appears to be the first mention of use of both a core spray and the core flooding system.

Dr. Doan, the head of the Regulatory Staff, expressed concern over basic questions that must be faced when the location of a reactor in heavily populated centers is considered (for example, do we really know that a collapsed core can be cooled?) There was no response given to that question, and it is not clear whether Dr. Doan had something more specific in mind.

ACRS member Silverman agreed that several improvements had been incorporated in the Boston Edison design, but noted they were based on a specified series of events which were defined as the maximum credible accident which must be considered. Mr. McEwen, however, maintained that General Electric considers multiple, independent failures highly improbable. He noted that the plant would accommodate two independent failures and suggested that the need for additional containment is questionable on the basis of probability.

The ACRS minutes of a further meeting between the Division of Reactor Licensing and the Boston Edison Company, held January 26, 1966, provide further insight into the thinking of the time. Dr. Doan said that the principal problem would be how to evaluate the reliability of the proposed reactor and its safeguards when there had been no operating experience on reactors of the same power and design. He pointed out that at the time Part 100 was written, a different situation prevailed regading the siting of reactors, namely, reactors were being located at some distance from cities. He stated he did not know how to interpret population center distance for a metropolitan site.

Dr. Doan went on to state that it did not appear that a new (safety) approach had been taken in the design of the proposed reactor facility for the Edgar Station site, and that only relatively minor safety improvements had been made over past designs. He said that the matter of whether the proposed reactor facility is sufficiently safe could not be based on experiments to be performed at some time in the future. Dr. Doan stated that, in his opinion, until more information was available regarding experience with engineered safeguards, there was not any chance that the Edgar Station site might be approved for the proposed reactor. He said he did not see how there could be a breakthrough of knowledge regarding safety of large reactors until some of the proposed large reactors had obtained operating experience. Dr. Doan stated that no one wanted to say that metropolitan sites are unsuitable for reactors. He said, however, that acceptability of metropolitan sites would come sooner if industry recognizes that present designs are not acceptable for metropolitan sites, and puts some effort into the matter. In regard to the proposed reactor design for the Edgar Station, Dr. Doan pointed out that the containment could not withstand rupture of the prssure vessel. (We might note that an ACRS letter regarding pressure vessel failure and the possible need for designing for its failure, especially for more populated sites, had been issued two months earlier.)

Mr. Levine of the Regulatory Staff stated the following preliminary comments and conclusions regarding the propsed reactor:

1. The actual design and the reliability of the control systems for the engineered safeguards will be looked at much more thoroughly than has been done in the past.

2. Potential metal-water reactions during accident conditions may be a problem.

3. The basis for establishing the assumed iodine removal efficiency during accident condition will be important.

4. The matter of isolation of the process systems at the containment boundary will also be important.

5. Tornado protection will be considered. (The Boston area was said to be only a factor of 2 in tornado frequency below the worst tornado belt in the country.) Hurricanes will also have to be considered.

6. The reliability and adequacy of the emergency electrical power supply system to run the required safeguard systems will be thoroughly reviewed.

7. DRL is concerned regarding the core spray system relative to:

    a. There is no test data regarding the cooling of a whole reactor core by a spray system. In particular, DRL is concerned that the flow of the steam generated in the core on actuation of the spray may interfere with further flow of a spray into the core.

    b. The core spray sparger ring might be disarranged during an accident.

    c. The reactor may become critical by reflooding if all the control rods are not in an inserted condition at the time of reflooding.

    d. The matter of how long containment integrity can be maintained, if the containment spray system is not functioning, will be considered.

8. Other questions involve:

    1. Provisions for automatic load sequencing during a loss of power incident.

    2. Protection against missiles.

    3. Means of monitoring and testing containment penetrations.

    4. Reliability of stack filter system.

    5. Control rod drive hydraulic lines and number of rods which might be affected by failures in the hydraulic system.

Dr. Doan concluded by stating he could not base a decision regarding issuance of a construction permit on conceptual design and that "hard" information would be required.

At some point after the January 26th meeting, the Boston Edison Company decided no longer to propose construction of the 600 MWe nuclear plant at the Edgar Station; instead of the Pilgrim reactor was proposed.

2.8 1965-1966: PRESSURE VESSELS, ECCS, AND THE "CHINA SYNDROME"

Because of their importance both to future reactor siting policy and to a markedly changed approach to LWR safety, developments during 1965 and 1966 with regard to questions of pressure vessel integrity and to the impact of core melt on containment integrity will be reviewed in detail, prior to proceeding with the detailed, case-by-case examination of siting policy evolution.

On April 15, 1965, the Commonwealth Edison Company applied for a license to construct and operate a 2255 MWt BWR to be located at the site of Dresden Nuclear Unit 1. The largest thermal power previously approved was that of the 1600 MWt Oyster Creek reactor; hence, Dresden 2 represented a large jump. While the surrounding area was relatively rural, the city of Joliet, Illinois was about 14 miles from the site, and the city of Chicago was about 40 miles from the site. Many members of the ACRS saw the Dresden 2 reactor as a probable prototype for other reactors in metropolitan areas; for this and other reasons, Dresden 2 received extra emphasis during ACRS review, and the potential resolution of certain generic matters became tied to the case.

The continuing pressure from industry (and, in a sense from the reactor development side of the AEC) for metropolitan siting was evident in a variety of ways. For example, the AEC established a Steering Committee on Reactor Safety Research, consisting of members from both the development and regulatory sides of the Commission. The minutes of the 67th ACRS meeting, October 7-9, 1965 report.

> Dr. Beck said that the formation of the (steering) committee stemmed from recommendations of the Regulatory Review Panel. Promoting liaison between the AEC general manager's staff and the regulatory staff, (in regard to) the safety research programs appeared desirable to the Commission. This research program is an outgrowth of the need for larger reactors, longer fuel life, and metropolitan location. ... Although it is doubtful that the steering group can arrive at all the necessary and sufficient safety items to allow reactors in metropolitan areas, some items are likely to be identified.

The minutes of the 68th meeting, November 10-12, and 22, 1965, report the ACRS Chairman, William Manly, as telling the Commonwealth Edison group that:

> This reactor could be considered a prototype for metropolitan reactors; consequently, the problems of the jet pumps (a new design), pressure vessel ductility, piping failures, and missiles are important in a more general way. Setting of a precedent with this reactor is of concern to the ACRS, and resolution of the problems is desired.

During the Dresden 2 review, the ACRS heard presentations on recent experiments concerning the effectiveness of the core spray systems (which, in duplicate, were the ECCS). The ACRS raised questions concerning the possible effect of pipe-whip and missiles on containment integrity, and on possible failure modes whereby both isolation valves in a main steam line might be lost concurrently and thus lose containment in a LOCA . However, the most difficult topic discussed by the ACRS in connection with the Dresden 2 case was that of the possibility of pressure vessel failure.

The matter of the importance of pressure vessel integrity to overall reactor safety was not a new one. The minutes of the 33rd meeting show member Connor raising a question concerning the need for improved inspection for reactor vessels. The minutes of the 36th meeting show member Osborn noting that pressure vessel rupture could lead to failure of the containment. In a report to the AEC dated September 11, 1961, the ACRS recommended the development of adequate codes and standards for the pressure vessel and other parts of the primary system of reactors. In a report dated May 20, 1961, the ACRS discussed matters related to the possible embrittlement of reactor vessels due to irradiation by neutrons over the lifetime of the reactor vessel. However, for the reactors previously reviewed, failure of the reactor pressure vessel was either treated as "incredible", or the reactor happened to be such that it could tolerate the failure of the vessel, perhaps because the pressure of the system was small and the power level was small.

The minutes of the ACRS Dresden 2 Subcommittee meeting held on September 1, 1965, show there was considerable discussion of pressure vessels with the members of the AEC staff and with the applicant. Mr. Muller of the AEC staff is listed as saying that pressure vessel failure was incredible. When asked by an ACRS member what would happen if the pressure vessel did fail, Mr. Bray of General Electric replied, "it would depend on the energy rate." He stated "the containment could withstand a larger break than the maximum credible accident (which was rupture of a large pipe) but not a complete break of the pressure vessel." He did not give any additional comment concerning the possibility of keeping the core cool for a rupture of the pressure vessel which might be larger than the so-called maximum credible accident. The minutes of the Subcommittee meeting show one member of the ACRS stating in executive session that he felt that pressure vessel failure was credible. Another member agreed, but suggested that the matter should be handled in criteria rather than with this particular applicant.

Actually the concern about pressure vessels had been growing during the year 1965. In 1964 there had been a failure at a temperature near the nil ductility temperature of a very large heat exchanger under test by the Foster Wheeler Corporation. On April 23-24, 1965, the ACRS held a

Subcommittee meeting on pressure vessel integrity. A range of questions arising from the very high requirement for pressure vessel integrity were left unanswered at that Subcommittee meeting, including adequacy of fabrication and inspection techniques, ability to ascertain the brittle-ductile transition region, and the behavior of thick-walled sections. There were also published in 1965, reports by Britsh research workers concerning the possible rapid failure of steel reactor vessels at temperatures above the nominal brittle-ductile transition range.

During the November 10-12, 1975 portion of the 68th ACRS meeting, one member (Okrent) took the position that, while it was acceptable for the Dresden reactor to be constructed at the site selected, in view of the current state-of-the-art, improvements were needed in the assurance of pressure vessel integrity and consideration should be given to the desirability of designing LWRs to be protected against the remote possibility of major pressure vessel failure. He proposed that it would be desirable, even prudent, to restrict pressurized and boiling water reactors not designed to cope with this extremely unlikely accident to relatively remote sites such as that proposed for Dresden 2; also, that future large reactors of these types should incorporate appropriate protective design features if intended for sites closer to population centers. During the November 10-12 meeting, the ACRS discussed this matter extensively but did not arrive at a decision as to whether it wished to prepare a letter of approval regarding Dresden 2, with added comments by a member, or whether it wished to deal with the matter in some other way, for example, by writing a general letter concerning pressure vessel safety for future reactors. It was decided to continue the 68th meeting and to hold another Subcommittee meeting on pressure vessels as part of the extended full Committee meeting.

To help the Committee in dealing with its problem, two members, Thompson and Palladino, each prepared rather lengthy letters to all the other Committee members in which they summarized the state of knowledge as they saw it, and tried to pose possible points of view and possible approaches. Because these letters in themselves provide a good example of how difficult it is to deal with a problem such as possible failure of reactor pressure vessel, and also because they provide considerable insight into the ways in which the ACRS tried to develop varying points of view in approaching such a problem, the letters are duplicated on the next pages.

Following the pressure vessel Subcommittee meeting of November 23-24, 1965, the ACRS decided to issue a letter favorable to the construction of the Dresden 2 reactor and at the same meeting to write a general letter to Dr. Seaborg concerning reactor pressure vessels. The ACRS members at the 68th meeting were W. Manly, Chairman; H. Etherington, F. Gifford, S. Hanauer J. McKee, H. Newson, D. Okrent, N. J. Palladino, L. Silverman, T. Thompson, and C. Zabel. The pressure vessel report and the reaction of the industry to this letter are shown on the following pages.

To: Members ACRS
From: T. J. Thompson

The discussions which have developed on the Dresden II Reactor appear to me to be the culmination of a growing concern on the part of a number, if not all, of the Committee Members as to whether or not a pressure vessel on a water-cooled power reactor--either pressurized water or boiling water--can fail in such a way as to breach the containment during an accident. I know of no way of stating conclusively that this problem is more serious with one type of water-cooled power reactors than the other. There may be in specific designs some difference in the probabilities of such an occurrence or differences in the ability of the system as designed and built to withstand at least the more minor incidents of this type. However, I do not believe that these differences are more than a fine structure on the overall problem. It may be that there will be differences in the cost to remedy the situation with the different kinds of reactors, but I have not looked into this and, again, I believe that it is a function of the specific design of the system. Therefore, in the discussion below I will assume that we are dealing here either with a pressurized or a boiling water reactor.

It is clear and I believe it has been from the beginning that the likelihood of such an accident is extremely small. I personally first thought of this problem when I was first introduced into the field of power reactors in 1955. Since that time the general problem has been of concern to me and I have continued to question my more knowledgeable peers in metallurgy and in the pressure vessel business as to the likelihood of such a failure. Before coming on the Committee and since as a member of the Committee, I have received in reply to my questions a steady stream of reassurances that such an occurrence was incredible.

At this point it is perhaps worthwhile to digress sufficiently to point out that the word "incredible" differs from the word "impossible". The dictionary defines incredibility as the state or quality of being unbelievable or hard to believe--an unbelievable thing. In my mind and I believe in most others in the nuclear industry the word "incredibility" has taken on a different connotation than "impossible". Dr. Etherington in questioning Mr. Joslin of Commonwealth Edison in regard to Dresden II defined "incredibility" as being something that would be unbelievable to a reasonable man and differentiated it from something that was physically impossible under the laws of nature. Obviously, a high pressure cannot exist within any confinement in such a way that it will be physically impossible for this pressure to relieve itself into a surrounding low pressure. Potential energy will always tend towards the minimum and, therefore, the fracturing of a pressure vessel is certainly not impossible--even ductily. It is therefore certainly not impossible that such a rupture should occur in a way so as to break the containment existing around the reactor.

At various periods during the past ten years I have spent considerable time questioning various reactor groups concerning their methods of selecting bolts, their methods of tightening bolts including use of bolt heaters and torsion wrenches and impact wrenches of various types, the care with which bolt tightening was carried out, and the methods of selecting and checking the performance of bolts. It has always seemed to me that this is the most vulnerable area and I continue to believe that this is probably true. More knowledgeable people than I in this field have continued to assure me that bolt failures in such systems will normally occur and first be observed by a leaking in one region and that this leaking is of such a nature as to give a forewarning of an impending difficulty. It has become common practice in the reactor industry to provide open microphones in isolated areas to insure that audible means exist for hearing the initiation of such leaks. In at least one reactor, the VBWR, such a microphone was instrumental in detecting a cracked major primary pipe.

Prior to the Dresden II case there had been at least one year during which no major reactors had been considered for construction permits. However, during this ensuing period several events have occurred which, rightly or not, have led to a growing concern on the part of the Membership of the Committee. First, there was the failure at a temperature near NDT of a very large heat exchanger under test by the Foster Wheeler Corporation. It is quite clear that the failure occurred at a highly stressed weld point and, further, that the failure did occur at or near the appropriate NDT temperature. Further, since the NDT temperature in general is in the region from 0 to 100°F the stored energy in the system at that time in the reactor is relatively small and, hence, the brittle fracture of a reactor vessel at temperatures below the boiling point of water at atmospheric pressure would normally not be expected to have sufficient energy to rupture the containment structures of any existing water-cooled reactors to date. However, when the NDT temperatures with appropriate safety margins began to get in the region above 200°F, it is absolutely essential to insure that the vessel is well above unsafe temperatures before high pressures are applied if brittle fracture is to be factored out. The review which the Committee held in March of 1965 in regard to pressure vessel and piping failures and to methods of measuring the nil ductility temperature was, I believe, disconcerting to all members who attended. For my own part, my most serious worries arising from that meeting concerned the apparent inability to determine with any kind of accuracy the existing NDT in thick slabs of steel. Further, I personally am now having some difficulty in knowing what NDT is the limiting NDT in a slab in which the NDT may vary through the thickness of the slab. Is it the surface or quarter thickness NDT that is the correct one in trying to judge the possibility of brittle fracture or is it some other value? These concerns, brought about by the Foster Wheeler failure, by the meeting of the Committee in the spring, by recent experience with pressure vessels and piping in other reactors, the increasing radiation exposure on those reactors now in performance, have all tended to point to the need for a general review in this area.

Initially, the reactor sites chosen for power reactors were quite relatively remote and it could be argued that the likelihood of endangering any human lives or at most very few were involved in the event that the incredible happened and that somehow the pressure vessel were ruptured. Since the initial reactors have been built, however, power levels and inventories have been going up as much as a factor of ten or more, the sites chosen have been in more populous districts and even in old sites already considered populations have been growing rapidly in the areas surrounding these. All of the factors cited above point to the fact that this is a good time for a review of the general situation. It is unfortunate that Dresden

II happened to be the first particular new major power reactor to be considered for a construction permit since this concern has been reopened. It is particularly unfortunate that the final review for this reactor should occur at the same session when another pressure vessel problem had to be considered. I am confident that the existence of one such problem in an operating reactor will not sway the judgments of the Committee in regard to the case of Dresden II.

It seems to me that there are a number of fundamental questions which should be asked before we reach a solution to the problem now facing us. These include:

1. Has the evidence developed in the last few months or year shown that we have under-estimated the likelihood that a failure of a pressure vessel may occur in a catastrophic way such as to rupture the containment?

2. Even though we may not be able to make a better estimate of the likelihood and, hence, cannot make an adequate judgment of one, we must ask whether the present or future site locations, the increasing fission product inventories, and other factors are sufficiently different from the situations considered in the earlier reactors that we should require complete protection from all missiles including the pressure vessel itself of the containment?

3. Should this containment be required of Dresden II?

Concerning question No. 1, I do not have a definite position. There are two facets to this problem, it seems to me. First of all, we may have been under-estimating the likelihood of this type of accident from the very beginning. In any case, it is clear that the likelihood of such an accident is very small indeed. Since there appear to be no reported cases of this type from major pressure vessels in the industry, the likelihood must be small, although clearly not zero. The second facet of the problem concerns the changes in technology which have occurred since the first reactors were reported on. Clearly, the understanding available in the field of metallurgy has increased greatly during the past few years. It is also clear that the techniques and the facilities available for vessel fabrication and for vessel cladding have improved greatly. New codes have been prepared and these codes provide better definitions and require more careful analysis than the older codes. At the same time, the reactor vessels have become larger and the wall thicknesses have become greater. The discussions we heard last spring would indicate that the NDT temperature is a function of the point at which the sample was taken through the thickness of the plate. While Section 3 of the code is indeed more stringent in its requirements on stress analysis, at the same time it has reduced the factor of safety from four to three. Improvements in the understanding of metals and in the methods of fabrication and in specifying and inspecting such vessels have clearly been made. The problem that I have is in determining what fraction of these improvements have gone into increased safety and what fraction have gone into helping the economics, as for instance by reducing the required thickness of pressure vessel walls and hence reducing costs. I have no way of knowing what this split may be, but I am under the general impression that the pressures during the past few years have been primarily those of economics since there have been no serious failures within the pressure vessel industry or reactors to indicate that the codes then in existence or now in existence are not sufficiently conservative. It is for this reason that I made the statement at the last meeting that I would have to judge on the basis of my limited knowledge that a pressure vessel or a reactor to be fabricated in the near future might be considered to be in general as safe as some of those fabricated for earlier power reactors. I was discussing this matter on Wednesday, November 10, with one knowledgeable member of the Committee who stated that he was not sure what a factor of three or four in safety really meant in regard to the safety of the vessel. For instance, as I understand it, the factor of safety in design on many key components in aircraft is two. A factor of safety of three or four might very well mean that no airplane could fly. On the other hand, it might be that a system with a factor of safety of say 1.5 or even less might be operated forever with complete safety. Much depends upon the method of use of the system, the likelihood of over-stressing and many other factors. Thus, when I said that it seemed to me that it was likely that the old vessels might be as safe as the new ones so long as both were operated properly within their known bounds of capability and performance, I was saying that the new vessel designed under Section 3 of the code with its reduced factor of safety might indeed be either more or less safe than an old vessel designed under a less stringent code but with a factor of four for safety and that the likelihood of one being a factor of two safer than the other is probably quite small.

I have stated these things in order to delineate my position as an admitted non-expert in the field. While I am neither a metallurgist nor a mechanical design expert, I have been forced through the years by pressure of circumstances as a reactor designer as well as a Member of this Committee to learn something about these fields. I would be particularly interested to have the comments of Bill Manly, Harold Etherington, and Joe Palladino on these points. All of these gentlemen are much more competent than I in these areas and could give definitive comments. Specifically, I would ask, "Do you feel that we can safely take credit for increased safety of the new vessels which will be fabricated in the near future over the earlier fabricated vessels, assuming that both the new and the old vessels are operated appropriately at the proper temperatures and pressures?" If such a factor of increased safety exists, it should certainly be taken into account in considering Dresden II and I would be most happy if my more knowledgeable colleagues can assure me that indeed it does.

It may also be that the meeting which we will attend on November 23rd and 24th will throw further light on this topic.

If the decision of the Committee is that we have indeed underestimated the likelihood of the accident being considered, it would seem prudent to review all of the power reactors presently existing with the following three objectives in mind:

1. Are there feasible ways in which the possibility of a containment rupture could be made physically impossible?
2. Are the consequences of a rupture of the containment during an accident such that, coupled with the incredibility of such an accident, they do not "constitute an undue hazard to the health and safety of the general public"?
3. Is it possible to assure adequate core cooling in event of such an accident by the existing means or should other additional means be introduced?
4. Is it possible that certain reactors should be shut down?

If it is indeed concluded by the Committee that we have underestimated the likelihood of this accident, then it seems to me that the Committee must at the same time take the position that Dresden II should have protection against this accident.

In considering principal question No. 2, if we conclude that there is no reason for changing our past views on the subject, we must still decide that point at which the likelihood of this accident, even though it be deemed incredible, combined with the population density existing or predicted fission product inventories, and the continued growing population lead us to believe that the situation is intolerable. It may be that on a careful review of this part of the subject we may indeed find that we have already passed the point where we are now comfortable and may find it necessary to go back and place additional requirements on already approved cases. It may also be that we can foresee ahead a developing intolerable situation and that a warning letter of the type written by Dave Okrent will be most appropriate. I believe all of us hope that the latter will be the case. However, at the last meeting I did not believe that there were sufficient facts before us concerning past and future cases in order to see whether or not we would be able to back up the arguments advanced by Dr. Okrent in his letter that the Dresden II site was a "remote site".

I am hopeful that we will find an important difference between the sites of the general type occupied by Dresden II and those being considered for the future, so that we can provide by a letter such as that written by Dr. Okrent an adequate and sufficient warning to all reactor fabricators and owners that such provisions will be necessary at some point. Perhaps, in that letter we can even provide a logical basis for the line of demarcation. Obviously, this will be extremely difficult since we have drifted from rather good sites to rather bad sites by easy stages.

It is my own personal belief that unless the tone of the letter is made very strong potential applicants will continue to let contracts and make agreements with fabricators and designs will continue to be started on the basis very much resembling the present basis before they are brought before the AEC. While we have not had a chance to discuss this within the Committee, Dr. Etherington and I attended an Indian Point meeting at Bethesda on November 2. There, in a rather poor way, I tried to indicate that the Westinghouse Corporation, the company involved with Indian Point, should try to use more imagination in their use of containment in order to provide more surety against very serious accidents and that this could hopefully be done without greatly increased costs and perhaps at a saving over present designs.

With these thoughts in mind I believe that we should: (a) be certain that we do have a demonstrable basis for drawing a line at some definite point. If that point is in the future beyond Dresden, I recommend that a letter even stronger than Dave Okrent's be written to insure that there can be no question about our views concerning the subject. I would like to suggest that Frank Gifford consider carefully the meteorology and population distributions and fission product inventories for those reactors we have reviewed in the recent past or will review in the near future so that they may give us the benefit of their special knowledge in these areas at the next meeting.

In regard to question No. 3, I do not have my mind made up in either direction. I do not know whether Dresden II should be required to provide protection for this incredible accident or not at the present time. The information which I lack is discussed above and I believe that that coupled with the pressure vessel meeting will provide sufficient input so that we can all arrive at a well-considered and judicious opinion.

Additional Note

Ten years ago at the time the first water-cooled commercial power plants were beginning to be designed and constructed, the question was very much could any power plant be designed and built. There was a great deal of enthusiasm after the declassification of existing reactor information and because of the thrust of the Eisenhower Atoms for Peace program. The emphasis on safety was certainly as strong then as it is now and people wanted first to insure that these early plants would be safe. There were worries about the stability which might exist in boiling and the response of various types of plants to load changes; and many other problems which we now consider solved or not too important. People were trying to design the perfect control rod mechanism and they are still trying to design the perfect control rod mechanism. The number of unknowns and the lack of knowledge caused everyone to be very conservative in establishing their performance limits and in carrying out their initial operations. It was recognized that plants were probably over-conservative and as years have gone by and as knowledge has improved some of this over-conservatism has relaxed. The power distribution within the core has been flattened, multiregion cores are being used, fuel elements have much longer lives than were predicted in those days, and burnout safety factors have been continually lowered. All of these factors tend to reduce the margin of conservatism. It is our duty to insure that this margin of conservatism is relaxed appropriately as new knowledge comes in that provides sufficient insurance that there continues to exist an adequate margin of safety. This continual review and updating is a dynamic process and the reactor designer is responsive to the pressures which are applied. During the past few years the economic factors have received the most emphasis. Even five years ago it was not clear whether coal or oil or nuclear power would in the long run be more economic. Now, I believe, it is quite clear that nuclear power can hold its own end of the economic race with more conventional fuels. The economic pressures now are those of an extremely competitive market both between reactor fabricators and between public and private power. These pressures tend always to reduce the margin of safety conservatism.

No reactor fabricator and no reactor owner will purposely design, construct, or operate a reactor in an unsafe manner if they in their own conscience believe this to be the case. On the other hand, all of these organizations receive considerable assurance from the AEC--both the Staff and the Committee--in the fact that these reactors are continuing to be approved and that therefore the Government shares this responsibility. In fact, as the regulatory process has become more detailed and its methods more refined, the utility user, especially the inexperienced ones, tend to believe that the problems of reactors are routine and so long as the reactor is approved by the Atomic Energy Commission it is by definition safe. This view is either consciously or unconsciously aided and abetted by the fabricators anxious to sell reactors. At the same time it should again be indicated that the fabricators of reactors in a conscientious way are doing their best as they see it to insure that reactors are safe. It must be recognized, however, that they are under strong economic pressures to respond to a keenly competitive situation. Therefore, personally, I would have great difficulty in arriving at the conclusion that reactors today are per se safer than they were say five years ago. It is obvious that we have caught a number of weak points in our earlier reactor designs, perhaps a good example is the 17-4pH problem. On the other hand, as we reduce the margins of conservatism other problems will arise and some of these have already been predicted. Some of the new problems may indeed be much more serious than those which we have faced to date. As yet, it has not been necessary to shut down any major power reactors for reasons of safety. It is still conceivable that this situation may arise, even with a completely new plant. If this situation should arise, it is clear that the embarrassment to the industry and the anguished gnashing of teeth from the economic viewpoint will be awesome to behold.

Additional Note

Another point that should be considered is the effect on the general public and on the industry of the public introduction of another concern regarding this incredible, but still possible, accident. If the final Committee decision is that existing reactors or future reactors at some definite level of population or other criterion must have such provisions, care must be given to the phrasing with which such a letter is written. It is necessary that the letter be crystal clear in its intent to the industry.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 16, 1965

MEMORANDUM

To : ACRS Members

From : R. F. Fraley, Executive Secretary
ACRS

Subject: DRAFT STATEMENT BY N. J. PALLADINO REGARDING REACTOR PRESSURE VESSEL FAILURES

Mr. Palladino has prepared the attached draft statement regarding reactor pressure vessel failures for discussion at the ACRS meeting on November 22, 1965.

Attachment:
Draft Statement by N. J. Palladino

CC: D. Duffey

DRAFT - NJP
11/15/65

Dear Bill:

As agreed at our last meeting, I have attempted to summarize herewith my current thoughts on the Dresden-2 reactor vessel.

Since our meeting, I have spent considerable amount of time reading up on pressure vessels. However, I must say that, except for the references quoted below, I was not able to find much new information to help me in clarifying my thoughts. The following paragraphs indicate some of the questions which we ought to discuss.

I agree with the other members of the Committee that the probability of a sudden large-scale failure of a properly built and operated pressure vessel is very low. This appears to be particularly true in the case of the Dresden-2 reactor vessel which, as reported by the General Electric Company and the Commonwealth Edison Company, is to be designed and built with great care and pressurized only when its temperature is above the nil-ductility temperature of the vessel material. However, because of the significant increase in the size of the vessel, in the power level of the reactor, and in the potential inventory of fission products in Dresden-2 over Dresden-1 and other previously approved reactors, it is prudent to reexamine the bases for assuming that catastrophic failures of the pressure vessel are incredible.

The safety record of pressure vessels in the United States has been very good during the past thirty years. C. R. McCullough(1) feels that one can ascribe the good safety record of pressure vessels in the United States to the existence of an excellent pressure vessel code to which practically all such vessels adhere. He goes on however to point out another reason for this excellent record.

"There is another reason for the excellent record of pressure vessels; they have been subjected to periodic inspections. It is not uncommon to find flaws in these vessels after they have been in operation for a certain period of time. It is common practice to repair these flaws and to continue to use the vessels. In most cases, flaws have not resulted in any leakage of the contents of the vessel. Unfortunately, in the case of nuclear vessels, up to now at least, there is no such inspection made. In addition, nuclear vessels are pushing design limits with their thicker walls promoting the possibility of more rapid thermal transients. They also are of unusual designs and frequently are using new materials. . . ."

C. K. Beck[2] points to several instances where incipient equipment failures in nuclear power plants could have led to more serious accidents if complete failure had occurred. Included among these have been the cracking of pressure-vessel-closure stud bolts. Having stated, that the safety record in reactor operation is indeed reassuring, Dr. Beck makes the following observations. (Underscoring is mine)

"On the other hand, as in all systems of complex machinery, mishaps of many kinds have occurred. At every reactor facility a long list of operating abnormalities accumulates over a period of time, and a complete tabulation for all reactors would include thousands of such incidents. Breakdown of machinery, malfunction of instruments, deviations from established procedures and operator's errors, would be among the incidents included in this tabulation. Most of such abnormalities would result in no undesirable effects or physical damages, though in a typical facility, a number of instances would lead to shutdown of the reactor and even in a few cases, some possible damages might be involved. But we are referring here to minor events which would not merit an "accident" label.

"Such experiences, which certainly occur to greater or less extent in all reactor facilities, give rise simultaneously to both reassurance and to uncertainty in the level of safety of reactors. Reassurance arises from realization that the margins of safety surely must be large for this number of mishaps to have resulted in so few significant events and no events of magnitude to have caused damage to the public. On the other hand, totally unexpected abnormal situations do occur, and it is the case that relatively minor events in themselves in combination with other abnormalities can turn an insignificant incident situation into a major accident.

"Further, there have been discovered in reactor systems a few incipient failures which, had complete failure occurred, would have resulted in more serious accidents than any thus far experienced. As examples, in three reactors, <u>two or three of the stud bolts on the head closure of the main pressure vessel or at crucial locations within the pressure vessel were badly cracked or broken</u>. In another reactor two main control rod shafts were found to be cracked from stress corrosion. In two reactors significant cracks were found in the piping of the main primary coolant system. Small, easily visualizable extensions of these situations could have led to serious accidents though by no means necessarily to major public hazards, for additional protective safeguards would still have remained."

It appears, therefore,that the good safety record of pressure vessels is, at least in part, due to the practice of subjecting such vessels to periodic inspection. It would seem that more effort should be expended in the development of means for such periodic inspections. (This could be done on Dresden-2 while it was being built). C. R. McCullough supports the need for such work in reference (2):

"A way must be found to inspect the primary system throughout the life of the reactor. It may be that a scheme can be found which can be used with the reactor operating but, if necessary, the reactor can be shutdown for a short time while this inspection is made. Work is underway on this as part of the safety research program using ultrasonic and perhaps other techniques. Such a continued inspection system would detect small flaws before they become large enough to be catastrophic. In the long run, with such a system, we might be able to relax our concern with major loss of coolant accident which is currently receiving so much attention."

He also states that

"The containment systems must be designed with sufficiently low leakage. At certain sites, double containment or an equally effective system may be required. These systems must be monitored, perhaps on a continual basis. The containment must be protected against possible missiles which would impair its integrity and it must be given pressure tests sufficient to promote confidence that it will function as needed under the accident conditions."

Another point which seems to require study with regard to large reactor vessels is the exploration of the influence of size on the integrity of reactor vessels. Initial scaling-up of other equipment such as ships and planes, I believe, has led to unforeseen stability problems. In a large reactor vessel, for example, might large-amplitude wall vibrations be experienced which could lead to unforeseen failure? Can the unforeseen bell-mouthing of large reactors lead to serious trouble; can bell-mouthing of such vessels lead to unanticipated bending moments on bolts? Do not larger vessels also provide more opportunity for defects? These questions are meant only to be examples of the types of questions with regard to size for which data appear lacking. I believe that more work should be done to determine the effect of size on probability of failure.

An important item of concern is the possibility of failure of closure bolts and blow-off of the head. Failure of even a few bolts on a head could so change the nature of the loading on the remaining bolts and possibly lead to their failure. It was just this sort of situation which led to failure of wheel bolts on a number of tank-transporting trucks during World War II. Load redistribution (due to bell-mouthing) led to failure of a head closure on an autoclave at Bettis.

Perhaps, on some bases such as initiation of studies indicated above, I can bring myself to go along with the other members of the committee on Dresden-2. In view of the fact that we have approved reactors such as the Jersey Central reactor, I do not believe that we would have a tenable position if we did not tie our need for further work to size rather than to location alone. Both the Dresden and the Jersey Central sites would be unsuitable for the proposed reactors if the pressure vessel failures were deemed credible. Therefore, I believe that we will have to shift the emphasis which Dave presented in his suggested draft.

As the Committee is confronted with new and larger reactor power plant designs, we may have to give thought to a different procedure for review. Such a procedure may have to involve detailed review of the reactor vessel and closure by the Pressure Vessel Subcommittee and perhaps by appropriate consultants.

REFERENCES

1. C. R. McCullough, "Some Philosophical Comments on Accident Models and Hazard Evaluation", INTERNATIONAL SYMPOSIUM ON FISSION PRODUCT RELEASE AND TRANSPORT UNDER ACCIDENT CONDITIONS, CONF-650407, Vol. 2, April 5-7, 1965.

2. C. K. Beck, "Power Reactor Accidents in Perspective", INTERNATIONAL SYMPOSIUM ON FISSION PRODUCT RELEASE AND TRANSPORT UNDER ACCIDENT CONDITIONS, CONF-650407, Vol. 2, April 5-7, 1965.

3. W. R. Cottrell and A. W. Savolainen, Editors, U. S. REACTOR CONTAINMENT TECHNOLOGY, ORNL-NSIC-5, August 1965.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 24, 1965

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REACTOR PRESSURE VESSELS

Dear Dr. Seaborg:

The design of pressurized and boiling water nuclear power plants has undergone many improvements with regard to safety, improvements which markedly reduce the risk of significant radiation exposure to the public in the unlikely event of certain accidents or system failures in such reactors.

There is a facet of current pressurized and boiling water reactor design practice which should be recognized, however. Containment design is generally predicated on the basis that a sudden, large-scale rupture of the reactor pressure vessel or its closure is incredible. Reactor designers have supported this view by detailing the extreme care to be taken in design, fabrication, and inspection of a vessel, and by specifying pressurization only at temperatures above the nil ductility transition temperature. They further cite the excellent record for large pressure vessels which comply with the ASME Boiler and Pressure Vessel Code.

The Committee believes, with the industry, that the probability of a sudden major pressure vessel failure leading to breaching the containment is very low. Nevertheless, it seems desirable and possible to make some provisions in future designs against this very unlikely accident.

1. To reduce further the already small probability of pressure vessel failure, the Committee suggests that the industry and the AEC give still further attention to methods and details of stress analysis, to the development and implementation of improved methods of inspection during fabrication and vessel service life, and to the improvement of means for evaluating the factors that may affect the nil ductility transition temperature and the propagation of flaws during vessel life.

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2. The ACRS also recommends that means be developed to ameliorate the consequences of a major pressure vessel rupture. Some possible approaches include:

(a) Design to cope with pressure buildup in the containment and to assure that no internally generated missile can breach the containment.

(b) Provide adequate core cooling or flooding which will function reliably in spite of vessel movement and rupture.

(c) If breaching the containment cannot be precluded, provide other means of preventing uncontrolled release of large quantities of radioactivity to the atmosphere.

In view of the very small probability of pressure vessel rupture, the Committee reconfirms its belief that no undue hazard to the health and safety of the public exists, but suggests that the orderly growth of the industry, with concomitant increase in number, size, power level, and proximity of nuclear power reactors to large population centers will in the future make desirable, even prudent, incorporating in many reactors the design approaches whose development is recommended above.

Sincerely yours,

/s/

W. D. Manly
Chairman

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ACRS Qualms on Possible Vessel Failure Startle Industry

The community of power-reactor designers, suppliers and operators was taken aback last month by a terse six-paragraph letter from the Advisory Committee on Reactor Safeguards to AEC, bearing recommendations quite unexpected at this time, and whose effect on the nuclear industry may take weeks to evaluate. The recommendations, in a nutshell, are 1. that sudden, catastrophic failure of a pressure vessel—since the start of the power reactor program classified as an incredible accident, one that need not be taken into account in reactor safety analyses, be reclassified as a possible accident; and 2. that future nuclear power station plans design against the possible consequences of such an accident (*see box*).

Industry reaction was sharp and dismayed. It ranged from resignation to protestations that the desired levels of quality control, stress analysis etc., were already being met, to comments that it's "almost impossible to design against complete separation of the vessel," as ACRS asks, nor is it necessary, and that "this kind of thing, done this way, borders on the irresponsible."

One of the aspects of the matter that troubled industry was that ACRS apparently acted without prior consultation with the technical safety experts on AEC's regulatory staff, and indeed gave AEC only the most cursory informal advance notice. If AEC felt at all uncomfortable about the ACRS letter, it did not dispel such an impression when it took the unprecedented step of attaching a covering statement to the ACRS report. In this, AEC called attention to the positive ACRS comments on safety of water reactors; pointed out that ACRS was recommending "that additional work be done," and recalled that AEC had already launched an "augmented and reoriented" safety program which would include work on the ACRS suggestion.

The blow was all the sharper because only two weeks earlier AEC had issued a set of design criteria for power reactors as guides to applicants for construction permits (NU Wk, 25 Nov '65, 1; text in AEC press release H-252). An immediate favorable response from industry met issuance of the criteria, hailed as a good step, a step in the right direction (NU Wk, 2 Dec '65, 1). Two weeks later, publication of the ACRS report brought wry comments from industry that every step forward on the licensing front seems to be accompanied by two steps backward.

Commented one industry official: "If you assume ductile metals can fail catastrophically from brittle failure, you must also assume the properties and behavior of ductile metals are not what we have assumed over the years, and that you cannot design anything: why do you assume a bridge will stay up? . . . The same thing applies to the possibility of a guillotine break in a pipeline: there has never been one in history; ductile materials don't fail in that way. Should we assume concrete will no longer support a building, or that glass will no longer insulate?"

Said another: "All the pressure vessels for any water reactor operate in the ductile range and therefore couldn't fail in a brittle manner . . . I think it's wise that we look at everything, that ACRS recognize its responsibility for safety and not permit anyone to build anything that isn't safe. The question is—you can analyze things forever and never get anything built . . . I wonder if it isn't coming to a ridiculous point. You could analyze if the reactor operator should ever step out in the street: something might happen to him and he might not be able to come to work."

Observed another: "You can't argue against beefing up quality control, stress analysis, etc.—we go along with that. But that's quite different from saying you should design for a failure."

At Burns & Roe an official said, "We've studied this and concluded that it's not possible: we could see no conceivable possibility of a full vessel rupture. Now you get to the numbers game—you can't say it's zero, but it's so small that it's not a postulated accident as far as we're concerned."

And at General Electric there was open talk about attempting to get the report's recommendations reversed by AEC, and about the existing system [never used yet by a nuclear industry member] of review by the courts, in case AEC did not act responsively.

Vessel Fabricator's Reaction

At Babcock & Wilcox, one of the only two U. S. firms that fabricate the huge reactor pressure vessels, an official said, "I believe the possibility

Text of ACRS Letter to AEC

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Sincerely yours,
/s/ W. D. Manly

of massive failure of reactor vessels has been reduced to zero—and the more we build, the better they will be. In time we will reduce the possibility of failure to nil to the sixth power . . . As for ACRS' three points: we are already doing this, increasingly and constantly."

Combustion Engineering, in a more formal statement, said: "The materials, the design criteria and methods, the inspection techniques and testing of nuclear components result in quality that vastly exceeds that which was previously obtainable for pressure vessels. The design specifications have and do take into consideration the known effects on the material exposed to radiation environments . . . We are certain that evaluation of present and past practice will show that the pressure-containing components of the primary system are quite adequate for the operating conditions for which the unit was designed."

Industry spokesmen did not fail to point out that the recent report of the Mitchell Panel on streamlining AEC reactor-licensing procedures had urged elimination of public disagreements between AEC licensing bodies by the making of "every effort to reconcile differences" in joint meetings. But it appeared that ACRS not only had not done so, but had in fact given AEC only the barest advance notice of its letter report. One industry official felt that in matters of this nature ACRS should have gone even further in the opposite direction: not only consulted with AEC in advance, but put out a draft version of its report for industry comment—as AEC did with the design criteria—prior to issuance in final form.

Three Principal Issues

There were three principal issues contained in the ACRS report to which industry reacted. One was ACRS' concern over the possibility of vessel cracking and failure due to changes in the nil ductility transition temperature. A second was ACRS' feeling that each future reactor should design against the possibility of gross vessel failure. Thirdly, it is understood, one of the Committee's concerns is the possibility of vessel head bolts shearing off and becoming missiles that might breach the containment.

The first, no new issue, is the effect of radiation on the nil ductility transition temperature, that is, the temperature at which a change to brittle behavior occurs in a ductile metal. This temperature normally is in the range of 10–40 F in steels such as those used in reactor vessels, or safely below operating temperatures. But under irradiation the temperature at which the phenomenon occurs may increase until it can approach—or so ACRS fears—the 300–400 F range of vessel operation.

Said one industry official with long familiarity with vessel fabrication: "Based on all data we have yet seen, as the nil ductility transition temperature shifts, the metal yield strength also increases, so that I'm not sure that we don't end up with a vessel that's safer." Another recalled that considerable investigation had been done on this at Argonne a few years ago, with no procedural changes resulting.

Declared another: "If after 10 years of operation you had a crack, your vessel might conceivably break under hydrostatic test; while this would be embarrassing as hell, it would not be a catastrophe since it would be inside containment, and the safety of the public simply would not be involved."

Improved materials of today were stressed. "Materials [for reactor vessels] are undergoing much better inspection than for a power boiler. We know really that the materials that went into power-boiler steam drums 40 years ago were nothing like the quality we now have, yet we have had no power-boiler drum failures in all that time." And: "We are building vessels of better steel than ever before, doing better stress analysis than ever before, using better inspection methods than ever before and using them more extensively than ever before . . . It is awfully late in the game to be coming up with this kind of judgment." And again: "It's almost preposterous to postulate a gross vessel failure, as opposed to perhaps a nozzle crack."

Secondly, on designing against failure, there were mutually self-contradictory views opposing ACRS from opposite extremes. On one hand, it was said, "We now have to design against instantaneous severance of a major recirculation loop in a BWR or of a major primary coolant loop in a PWR, and complete loss of all fluid [in either case]. It's difficult for me to realize that anything arising out of a small crack or flaw in a pressure vessel would exceed that requirement. We are now designing to handle the complete loss of fluid and fission products from the reactor."

On the other hand were those who said, "This is an intolerable requirement—we can't live with it"; or "I doubt personally that it is credible to completely design, at least economically, for this massive failure of a pressure vessel. You just keep piling things one on another and you get to the point where you can't do it. If the pressure vessel can fail, the containment can fail; if you contain that containment, that can fail, and so forth . . . . Back of this has been a steady increase and pyramiding of the number and severity of accident modes we are supposed to take into consideration. It would seem that these things are changing as the makeup of ACRS changes: I have the impression that membership of ACRS is moving more and more toward an academic, a college-physics-professor, type of person."

Academic Trend?

The feeling that ACRS' requirements are trending toward the academic was fairly widespread. Said another experienced industry man: "This [requirement] clearly says, 'Six months, bud, on top of any completion date you got.' Like Rickover, they say 'prove it.' So you do, and they say, 'prove that'—and you can go on doing arithmetic forever . . . They should have to prove the justification of their question. We always have to prove—they can just think, opine. Maybe they should be asked 'where?' 'when?' 'why?' "

Finally, as to the possibility of missiles breaching containment, one industry man countered: "We have a very good story on this, because we do actually stress each bolt with a bolt tensioner, so we know the actual stress on each bolt as we close it, and the bolt is easy to examine."

As to designing containment against internally-generated missiles, one architect-engineer commented, "The question is how big a missile and how much force? Containments are now designed to take rupture of a primary pipe and the hydraulic head if a pipe breaks. Now they ask, suppose a bolt flies off. Suppose the head flies off? I don't know what the intent is. Supose you say the bottom head comes off? How far can you push these things? If you continue, the design problems will become virtually insurmountable. . . . We don't know how to design a structure against the kinetic energy of a 100-ton vessel trying to blast its way through. . . . Is this just thinking up other things to protect against? If so, where is it going to stop? Why bring it up now? Do they know something we don't know? I question it."

69th Meeting
Jan. 6-8, 1966

MEETING WITH COMMISSIONERS

The recent committee letter on pressure vessels was discussed. Mr. Manly said that the letter was an outgrowth of the desire for large reactors in metropolitan areas. Dr. Newson considered the Malibu Reactor, which was the subject of several site changes some years ago, as being the earliest reactor to bring attention to the problem; the Jersey Central Reactor which was approved, reactivated the metropolitan reactor issue and lately the Dresden II Reactor, because of its large size and marginal site characteristics, more directly led to the pressure vessel letter. Dr. Thompson referred to the document, WASH 740, which pointed to the serious effects from accidents with very large reactors; some of the information on the catastrophic consequences could be very alarming if improperly presented to the public.

It was pointed out to the Commission that the problem is not really new: it has been mentioned in Committee letters of the past. Dr. Okrent noted the extended licenses of reactors, e.g., forty years, which may make more important the slow, but progressive, changes in the nil ductility temperature (NDT) of metals. Dr. Okrent feared that reactors are not satisfactorily protected against catastrophic pressure vessel failures. No vessels made with the new pressure vessel code have operated for any length of time; hence, little experience is available on which to base confidence in the sufficiency of the present fabrication methods.

One intent of the pressure vessel letter, the Commissioners were told, was to encourage improvement in the quality of the workmanship and in the extent of inspection and to promote better surveillance throughout the reactor life. The excellent record of industrial pressure vessels probably rests on the careful inspection during fabrication; nevertheless, only a limited number of vessels have been operated under nuclear reactor conditions. A second issue of the letter was towards design changes to handle the consequences of an unlikely, but catastrophic, pressure vessel failure. Possible implementation measures for the letter by the AEC might be collecting the information on the pressure vessel technology and formulation of a technical information document, review of past and present reactors experience, including the results of surveillance predictions on vessel life; finally an incentive method, perhaps financial, might encourage ingenuity to improve the capability of the vessels as well as reactor designs to limit accident consequences. The strongest incentive would be the possibility of acceptance at a metropolitan site. Mr. Manly identified steps taken by the Committee toward this problem, e.g., attendance at the meeting on pressure vessels at which reactor manufacturers and pressure vessel fabricators were present.

Review of reactor pressure vessels at frequent intervals was suggested by Dr. Thompson. Mr. Manly commented on the quality of pressure vessels and the consequences of an accident. The variations in fabrication might lead to a factor of ten between predictions and experience in pressure vessel behavior; consequently, the Committee specifying measures which would change the reliability by a small factor might be criticized by the industry as being meaningless. Much expense is involved, and the industry will have to be forced to take steps towards precautions against catastrophic failure of vessels.

In response to Chairman Seaborg's questions on pending reactor cases, Mr. Manly recalled that the pressure vessel letter said each reactor should be considered separately, as at present; this reflects the continued discussions of the problem within the Committee. Although the Committee hasn't explicit instructions, efforts towards better pressure vessels are desired and steps to encourage the nuclear industry towards this end seems needed; Mr. Manly felt that such efforts could be made without changing the present reactor schedules appreciably. Some of the Committee believes that improved designs might be easier to arrive at than does the industry.

Chairman Seaborg asked for comments on the Nucleonics article regarding the Committee's vessel letter. The Committee said that some of the comments seemed carefully considered by industrial groups while others, some anonymous, were not well developed, e.g., the statement about going to court to object. The feeling of incredibility of failure of pressure vessels by the industry was noted; Mr. Palladino expected such an industrial reaction.

PRESSURE VESSELS

Executive Session

Reference was made to the recent letter on pressure vessels and implementation was discussed. The lack of a clear path of implementation of the letter was noted by Dr. Silverman. Dr. Hanauer considered the letter aimed primarily at future reactors; nevertheless, reactors under design may be required to take reasonable steps to protect against pressure vessel rupture. The consensus was that reactors for which applications have been filed e.g., Millstone and Indian Point II Reactors, would not be particularly affected but later reactors, e.g., the Brookwood Reactor, would probably receive more attention in accord with the letter. Dr. Thomspon finds it difficult to decide on which reactors,pressure vessel precautions are required.

Dr. Thomspon saw the ACRS as having indicated a need for a steady betterment of pressure vessel reliability and not a step increase in requirements; nevertheless, if the Committee feels strongly about the needs for changes, then past reactors would probably have to be reviewed. Dr. Okrent believes the RS has unofficial limits on reactor locations which permits large reactors at some sites but excludes others. Applicants whose cases are pending are probably intentionally overlooking the problem, e.g., the Millstone Reactor group is believed to have no intention to take any steps. The General Electric Co. (GE) was reported to feel that future pressure vessel restrictions do not apply to the Dresden II Reactor and probably not to the other GE reactors; but the RS has warned the group of a need for future compliance.

Plots of the population against distance for several reactors were reviewed. The definition of a metropolitan site could hinge, according to Dr. Hanauer, on the location at which catastrophic pressure vessel failure would be important. Dr. Thomspon said that sites must be considered individually; to him, the Boston Edison site would clearly be a metropolitan one.

The consequences of a breached containment accident were discussed. Mr. Etherington said that within ten miles of the Brookwood Reactor, about which a decision is needed soon, the consequences of an accident would be less severe than at other reactors, e.g., the Millstone Reactor; but beyond this, the Brookwood situation was probably not as good. His general feeling was that the pressure vessel letter does not apply to the Brookwood plant. Nevertheless, the possibility of many individuals being killed makes implementation of the letter imperative; in addition to fatalities to the surrounding population other damage, e.g., genetic effects, must be considered.

To Mr. Palladino, any accidents with a pressure vessel might be considered to parallel the experience with ordinary boilers; leaks in faulty boilers have indicated failure and allowed time for safety measures. Mr. Palladino sees inspections of pressure vessels as a further step to prevent the failure of the vessels or the surrounding confinement.

Dr. Hanauer believed that although the letter may have avoided identifying a threshold in population damage at which the precautions for pressure vessel failure would be required, some such figure does exist, e.g., the death of ten thousand people might require very positive steps; inclusion in the Part 100 Regulation of a threshold figure was suggested. Nevertheless, Dr. Thompson noted difficulty in identifying a loss of life threshold criteria. Dr. Hanauer said that the safety inherent in pressure vessels probably exceeds that assured by engineering safeguards of a reactor. Although there is now much attention to the uncertainties in the behavior of pressure vessels, Mr. Manly postulated that other features of reactors, e.g., the electronics, might have similar uncertainties, but the consequences of failure are probably not as serious.

Perhaps more than a hundred reactors are involved in any decision regarding pressure vessel failures. Ten to twenty large projects are expected to come to the AEC as applications in a year or so, and many of these will be metropolitan sites; attention to this problem, with probable expenditures by the applicants is, therefore, mandatory. Projects such as the Brookwood reactor might be stopped if the measures forced by the ACRS are very expensive. This might be considered to conflict with the Committee's mission; the basic mission is safety, but it might be considered to be indirectly, promotional because of the relationship to the AEC.

Dr. Thompson prefers imaginative steps at this time of transition to more restrictions on reactor design because of the size of reactors and the proximity to cities. Guaranteed means of heat removal from the containment following the release of a large amount of fission products after a loss of coolant accident, even with no missile damage to the containment, is required. This would preclude hazards from fission products passing the successive barriers. Reactor designs to assure no nuclear transient and hence a sequential release of fission products following a loss of coolant accident may be feasible; in this case, only shell cooling is required. However, the Committee has, in the past, lacked enthusiasm for the safety features of sequential release. Ingenuity is needed soon; the Indian Point II Reactor, according to Dr. Thompson, might be the first application requiring serious attention to new pressure vessel measures. The Commonwealth Edison Group has recently mentioned a Dresden III Reactor and a site for a fourth facility. Mr. Etherington suggested applying the letter in a major way with the next metropolitan reactor.

Because of the competition between fossil fuel and nuclear plants, which results in much efforts to decrease costs, any additional expenses against pressure vessel failure will probably be resisted by the industry. Dr. Palladino sees that ACRS rejection of any industry proposals towards this protection will be followed by a request for advice on satisfactory measures; this would put the Committee in the position of designing reactors. But Mr. Manly saw no need for the Committee to be unduly concerned over this. Dr. Gifford and Mr. Palladino believe exciting the applicants to further action is the principle aim of the letter.

Dr. Thompson said that the Consolidated Edison Co. might connect the containments of the new and the old reactor to provide more volume for fission product retention. Mr. Etherington believes that any accident which would remove a vessel head would probably remove the core and make cooling of the core components very difficult. The general problem of missiles and large exposures to the public were not resolved by the Committee with the Dresden case. Dr. Okrent advised the subcommittee to include this in its consideration.

Sometimes efforts to strengthen systems, e.g., pressure vessels, can create weaknesses. Information on the rejection of steel plates for pressure vessels and the consequences of acceptance of an inferior plate is desired by the Committee. Mr. Manly suggested that the AEC support efforts towards development of these proposed precautions. More information on the effects of these fission products which become more important with very large reactors, e.g., the long lived fission products, is needed. Dr. Okrent hopes for an early assessment of the problems and ways to ameliorate the hazards; when available, vendors can be advised and some action might be possible with the current group of reactors. Dr. Zabel postulated that approving the present reactors and delaying a decision on pressure vessel failure precautions might make the problem more difficult to solve; to him, the dilemma is the serious consequences of an accident as considered against the industrial tendencies towards lower costs because of economic pressure. Mr. Manly predicted that the companies will meet requirements on this pressure vessel problem in order to compete.

Dr. Thompson said that since the letter had been written, the Committee must follow through with advice towards metropolitan reactors. Immediate steps might be inquiries to the AEC regarding expenditures towards pressure vessel integrity, resolution of the nil ductility temperature (NDT) problem and better inspection techniques. However, construction uncertainties, such as droping a vessel during installation and unaccessable damage, as with the Hallam vessel or the San Onofre heat exchanger, complicate the picture.

Mr. Manly saw the issue of pressure vessels as not new but one generated some time ago and commented upon in the Committee letters towards research and development and in the draft on the Committee policy towards metropolitan reactors. To him, urging for AEC efforts towards more basic knowledge of pressure vessel failures and better techniques of inspection is a first step; another is, of course, to design reactors to accomodate a large failure. Dr. Hanauer advised that the Committee make clear what is desired of the RS. Mr. Palladino proposed that the RS assemble the evidence that pressure vessels will not fail, e.g., statistics on past experience and on the very careful inspections; if this evidence is not convincing, then the applicants, on whom the burden of proof rests, should be made to take appropriate precautions. Much time may be required to make such a summary; but pending cases, according to Dr. Bush, will require early attention.

Information leading to the rejection of pressure vessels for the Navy program would be helpful. Mr. Manly recalled occasions where vendors successfully pursued claims against the AEC for manufacturing requirements beyond the state of the art. A company with a large piping contract was put in severe financial straits by being held closely to specifications. Frustration over the Navy's contention that inspection has eliminated any possibility of pressure vessel failure with Naval reactors was expressed by Dr. Thompson; this may encourage the industry to develop specious evidence against possible failure. Although quality improvement is a way to limit the hazards of pressure vessel failure, it may not be a very secure basis; continued improvements probably have a limit, but Dr. Newson suggested that new concepts which increase safety will probably be developed. Dr. Bush, based on research to date, believed that a monitoring system of materials placed inside the pressure vessel might not be a particularly expensive way to provide assurance against vessel failure; a feasible scheme is available, but development, costing perhaps a half million dollars, would be required to make it practical.

Although better inspection is probably not difficult, acceptance of new design ideas by the fabricators of vessels may be less easy because economics tend to color technical judgement. According to Mr. Manly, available tests for the integrity of plate, e.g., ultrasonic methods, are not used by manufacturers, e.g., Lukens Steel Co., because of the expense. Consequently, applicants may have to be coerced to better inspection and to design precautions to resist vessel failure; a document outlining the state of the art may be needed to obtain compliance of reactor vendors. Dr. Silverman sensed a lack of realization of the RS regarding these economic factors. Although Mr. Etherington doubted if improvements in pressure vessel precautions will develop with current projects, the agitation will probably lead to manufacturers including a sizable contingency in their design estimates.

Comments on possible industrial standards for pressure vessels were made. Mr. Manly said that industrial codes reflect the consensus of the manufacturers, the construction contractors, and the users. Dr. Bush recalled an effort towards standards for tubes which required about five years, but the standards were not accepted because of economics.

A policy statement, perhaps oral, regarding large reactors near cities, and perhaps clarification by means of the Brookwood case, was discussed. Although Dr. Okrent favored a statement now, Dr. Hanauer was inclined against a firm position until the present reactors have a more thorough review. Dr. Zabel was inclined to stand by the original letter, since further comments might only confuse the issue. To Dr. Silverman, the Indian Point II Reactor is definitely in the metropolitan category which means the issue must be faced almost immediately. The Malibu and Dresden II Reactors, which the ACRS approved, could be a basis for review of the future metropolitan reactors; five more applications for large reactors are expected by the RS in February and a similar number in March. Dr. Newson saw the ACRS position as one of more leniency towards those large reactors whose applications are already before the RS; discouragement of some future metropolitan reactor applicants may be in order. Mr. Etherington noted that operators and manufacturers must give careful consideration to the pressure vessel recommendations and indicate the steps taken to comply. Underground locations as protection against catastrophic failure of metropolitan reactors was again mentioned by Dr. McKee. Subcommittee attention will be given to the steps being taken on the current group of proposed reactors, e.g., the Brookwood Reactor and the Indian Point II Reactor, towards pressure vessel precautions against failure.

Regulatory Staff (RS)

Dr. Kavanaugh, Mr. Shaw, and Mr. Rosenthal of the Safety Research Steering Committee were present for this session as well as Mr. Price, Drs. Beck and Mann, and many other RS Members.

The history leading to the letters on the Dresden II Reactor and on pressure vessels was related by Dr. Newson. Some years ago a boiling water reactor for the Malibu site might be considered to have been rejected indirectly by the Committee's comments. As a parallel, reactors with the present pressure vessel designs are probably satisfactory for sites such as Dresden but not at more metropolitan locations. Mr. Manly recalled comments stemming from the interest of the Joint Committee on Atomic Energy (JCAE) on improvements to reactors needed for metropolitan sites. Both Dr. Beck and Mr. Price felt that insufficient time was being spent jointly by the Committee and the RS on this important problem; divided opinion exists within the RS as to the reliability of pressure vessels. The Nucleonics comments on the recent pressure vessel letter were interpreted by Mr. Manly to imply a difference of opinion between the RS and the Committee on pressure vessels.

Mr. Price expressed serious concern over any failure of a pressure vessel of a major reactor anywhere, not necessarily one near cities, that an appreciable number of individuals might be injured; although a million individuals might be involved in an accident in the New York area, sizable numbers could be involved at more remote sites. Dr. Beck was much inclined against any implication that possible failures are and important only at certain distances from cities.

Mr. Price said that although the probability of pressure vessel failure or head loss is low, it is still not zero. Mr. Manly seemed not seriously concerned over the possibilities of a catastrophic pressure vessel failure; to him the letter urges steps towards more research and development by the AEC and industry. Mr. Shaw complained that if this was the intent of the letter, prior discussions should have been held with the appropriate AEC groups; he interpreted the letter to apply to only current and future reactors. The question in Mr. Beck's mind is a definition of the work future; orderly improvement is desired, but who exercises the leadership is not clear to him.

Dr. Hanauer sees continued efforts towards safety of pressure vessels reaching a limit where additional expense on the vessel is not justified; then redundancy in equipment may be the answer. The positive nature of the pressure vessel letter was emphasized by Dr. Thompson, e.g., negative statements which might limit location of reactors was avoided; the problem is the large accident with serious consequences but of low probability.

A need for statements from the experts at Oak Ridge, Hanford, and other AEC laboratories on the state of the art with pressure vessels was seen by Mr. Manly as a first step to identify information that is lacking. Finally, measures to protect the public even if there is a severe rupture of a pressure vessel would be considered. Dr. Beck observed that major changes in reactor design, e.g., abandonment of the pressure suppression concept might result.

Dr. Thompson pointed to the uncertainties in pressure vessel design and behavior as related to the prevention of a large scale accident, as mentioned in the report WASH 740. Mr. Shaw commented on excessive precautions against failure of equipment; measures to limit damage from a defective head failure might be ineffective, and his preference would be a better vessel head or more bolts instead of compounding precautions to limit motion of a released head. Dr. Kavanaugh doubted if the AEC could now identify a research and development program to allow reactors in populated areas in two years.

The AEC Safety Research Steering Committee recognizes the problems of quality control and the need for research or improved inspection techniques of pressure vessels; more research and development into mechanisms for failure and the nil ductility temperature (NDI) shift is needed. The program towards understanding embrittlement of metals might be pointed more towards pressure vessels; the propagation of cracks in metal may need more study and or re-orientation. Dr. Beck can now establish some criteria for inspection and design, which might be added to the design criteria being formulated; means are already available to improve the quality of pressure vessels. The AEC promoting improved vessel quality and reactor design approaches to protect against the consequences of vessel failure was recommended by Dr. Okrent, e.g., vendors might be given AEC contracts. Planning of such research and development by the AEC might lead to quicker results because of the economic limitations on industry; Mr. Shaw mentioned a million dollars a year for such efforts, but wondered what one would tell an applicant. The AEC can finance such as the LOFT and the Advanced Fast Reactor (AEC), but commercial reactors are a different problem. Mr. Shaw doubts if the industry will be willing to use higher standards unless required by the AEC.

Mr. Kavanaugh asked when the ACRS desired attention by the industry to the pressure vessel problem; Mr. Manly pointed to the statement in the letter that reviews of cases would continue on an individual basis. The RS considers itself in a delemma; if the requirements to prevent or constrain a pressure vessel failure are not strict, economics would result in little industrial effort to comply with the letters intent, and the same old reactor designs will be submitted.

Dr. Bush reported cases where code standards exist without methods of inspection to assure compliance are lacking. All vessels are specified to the same code. Pressure vessels made to the same code may be more expensive for some purchasers because of stricter inspection and increased rejection rate; costs with the Pressurized Water Reactor may have been fifteen to thirty percent more because of firmer fabrication standards and more extensive inspection. To Mr. Manly, any action which would result in a moratorium on reactors must be avoided; progress is desired.

Multiple standards for reactors at the Idaho test site was indicated by Mr. Shaw, e.g., the standards for the Navy and the commercial type reactors differ. He emphasized that the nuclear industry may find itself in difficulty if care is not taken in defining criteria on pressure vessels. Mr. Etherington commented on the failure of pressure vessels; if the failure is from improper fabrication, quality control can improve the vessels but if the vessel is overstressed, then quality control does nothing to prevent failure. Dr. Palladino said that occasionally problems which are deemed by some as insoluble are found, with continued effort, to have answers and savings have often resulted; therefore, study of the quality of pressure vessels and design efforts as related to consequences of a large accident must be continued.

Dr. Kavanaugh was complimentary of the Committee letter on pressure vessels and saw it as generating needed safety features of reactors; he regretted the concern of the industry. Dr. Kavanaugh, Mr. Shaw, and Mr. Rosenthal then left the meeting.

Dr. Doan asked for the Committee position on the long range problems with the pressure vessel of the Yankee Reactor; Dr. Okrent saw no comment soon. Dr. Okrent asked for information on the surveillance program planned by the various design groups regarding the NDT shift and is the RS satisfied. Mr. DiNunno is collecting data on progress. Mr. Manly reported that recently an industrial man refrained from giving NDT information publicly because of industrial competition. However, Dr. Bush recalled slow, but successful, efforts in assembling information on ductility changes.

Mr. Price asked for guidance to the RS on pressure vessel measures in view of the Committee letter and the discussion with Mr. Shaw's group; Mr. Price understands that there is a need to collect information for improved inspection and to develop novel reactor designs to prevent failure or to handle the consequences. The Committee was reluctant to recommend steps; too much detail might hamper the industry and lead to less effort to understand the problem. Nevertheless Dr. Hanauer advised conclusions soon because of the large size and number of reactors planned for metropolitan areas.

Some interpretation of Mr. Shaw's comments was given by the Committee members. Mr. Manly understood Mr. Shaw to feel that the RS can force reactor designers and builders to take appropriate pressure vessel measures. More inquiry by the RS into the efforts of reactor designers was suggested by Dr. Okrent. Mr. Manly stated that Mr. Shaw would rely on strict rules of fabrication and operation, e.g., as with the Naval Reactors, to prevent failure of pressure vessels.

Dr. Beck was uncertain as to RS action and is against any abrupt change in policy. Improvements in pressure vessel technology are appearing, and new criteria on vessel fabrication and inspection may be desirable. He was in doubt over the need for requiring protection against the consequences of major failure. Mr. Levine reported that Mr. Shaw had estimated a half a million dollars expense for a blast shield for a pressure vessel. Although Congress desires no more AEC expenditures towards water reactors, because of the intense industrial activity, Mr. Price said that safety might be in a different category.

Dr. Zabel told Mr. Price of his recommendation of no alteration in the pressure vessel letter. The Committee's inclination to apply the letter in a gradual way was evidenced. The Boston Edison Reactor may soon provide a situation where full precautions against vessel failure may be required. Later Mr. Price indicated that the Edgar site for the Boston Edison Reactor might not be satisfactory even if all the pressure vessel failure precautions now considered are taken. Dr. Okrent stated that the pressure vessel letter will be applied gradually to reactors with current applications. An increased number of ACRS questions urging continued improvements in pressure vessel quality and precautions against failure should be expected. Nevertheless, each reactor will be given individual attention by the Committee. Dr. Beck interpreted the ACRS statement to mean more probing into the cases filed, particularly as to the inspection and surveillance of pressure vessels; with new cases, more definite steps in design would be required. Dr. Bush told the RS of his belief that existing methods can allow reasonable pressure vessel surveillance for large reactors but not within a matter of months. However, space might be left available in the reactor system to accommodate these measures.

Reactors near cities will be given pressure vessel attention by the ACRS earlier and to a greater extent. According to Mr. Price, if any applicant is forced to protect against pressure vessel failure, all other commercial reactors must comply regardless of location. Mr. Case asked for the Committee feeling of the acceptability of protecting against the pressure of a sudden rupture, but maybe not the missiles. No Committee opinion was offered, except the observation that requirements might vary at different sites. Mr. Case would view any response by industry to further pressure vessel requirements as a commitment which leaves him less inclined to accept compliance now.

Mr. Price presumed that military reactors, and those already built, would be exempted from future requirements of pressure vessel protection. Mr. Price hopes that the studies will show reasonable methods can be developed to protect against the pressure vessel failure; if not, the RS will have difficulty because of the incentive to lower cost reactors and the desire of the Commissioners to promote the nuclear industry.

Mr. Price ended the session by reporting that he had again talked to the Commission and to the Steering group after their sessions with the Committee regarding the pressure vessel letter and possible RS steps to comply.

At its next meeting, the 69th, January 6-8, 1966, the ACRS had to give much attention to the question of how to implement the recent letter on pressure vessels. Was it to be applied to succeeding reactors in terms of improved quality? Was it to be applied after a suitable time interval to all reactors in terms of protection against certain vessel failure modes? Was it to be applied in terms of certain vessel failure modes only for reactors of very high power at relatively highly populated sites? The Committee met with the Regulatory Staff, with members of the Division of Reactor Development, and with the Commissioners. Several sections of the minutes of the 69th meeting are reproduced on the next pages for the insight they give into the though processes involved.

Review of the Brookwood and Millstone Point reactors was facing the ACRS, and it was decided to pursue with these two groups various aspects of the pressure vessel question, including a discussion of the consequences of various failure modes and the probabilities of these types of failures, as well as a discussion of the types of things that could be done to improve the quality of reactor pressure vessels.

It was with some anguish that the utilities discussed on the record and in writing questions concerning pressure vessel failure. It was a topic not dealt with in this manner previously; it was a topic relating to accidents for which the reactor was not protected.

The minutes of the 71st ACRS meeting, March 10-12, 1966, show that Mr. Roger Coe, a representative of the Yankee organization appearing on behalf of the Millstone Point reactor, joined the Committee in executive session and read a statement regarding the trend of written questions in regulatory groups to applicants regarding the seriousness of postulated accidents. Mr. Coe stated that the correspondence becomes a public document and, in one case, for example, the Brookwood reactor (later renamed the Ginna reactor), this correspondence was quickly collected and aired by the press. (See excerpt from Nucleonics Week, February 17, 1966 quoted below.)

> ACRS IS PRESSING ITS CONCERN OVER PRESSURE-VESSEL FAILURE IN REVIEWING BROOKWOOD, the first reactor project to come up for a construction permit since the Advisory Committee for Reactor Safeguards in December first postulated as a serious possibility of a large-scale vessel rupture (NU Wk, 9 Dec '65, 1). The 450-Mwe Brookwood station is to be built for Rochester Gas & Electric by Westinghouse. In a letter to RG&E earlier this month, AEC's reactor-licensing chief, R. L. Doan, submitted two series of questions, one on behalf of the licensing staff, the other on behalf of ACRS. The ACRS questions bearing on the pressure-vessel-rupture issue include the following:

- If one postulates the rapid propagation of a crack circumferedntially, with the contained energy of the system, what would happen to the upper section of the vessel including shearing the primary pipes, etc?

- Can you visualize any problem from the propagation of a crack from top to bottom of the vessel but not through the head?

- Do your calculations confirm that the steam-generator tube sheet will withstand shock loading by abrupt loss of primary coolant, or will the head go instead?

- Are you considering procedures for detecting the propagation of cracks within the pressure vesel wall, i.e., acoustic emission?

- Define the pressure vessel flaw size and type that is accepted in the specifications. What flaws larger in size or of special significance might not be detected, particularly in zones of irregular geometry?

- What flaw size is accepted in the studs of the pressure vessel? What frequencies of study, inspection or replacement is planned? How many studs can fail without threatening the integrity of the closure?

- Please describe requirements concerning the support structure for the pressure vessel, including the degree of levelness over reactor life, which are needed to insure no problems due to local overstressing of the pressure vessel.

- Describe how small leaks in the pressure vessel would be detected, and the action to be taken should such occur. How is adequate response assured in the event of a previous existence of small leaks in other parts of the system?

In addition to these, ACRS asked five questions on nil ductility, including validity of neutron flux dose predictions for the pressure vessel, weld regions and heat-affected zones. The AEC staff questions submitted by Doan were largely answered orally by RG&E and Westinghouse at a meeting Jan. 26-28, but the staff wanted a "written response...to confirm the oral information.

Mr. Coe referred to private discussions in the past with applicants where the presentation of information was on a very candid basis. He predicts that this public method of communication will lead to less frankness and perhaps to intervention by the opponents of nuclear power in order to delay private nuclear development. Mr. Coe was quoted as desiring an arrangement by which such questions could be raised without the public being informed. Verbal requests and informal replies are a possible way to do this.

These comments by Mr. Coe were triggered by the written questions, transmitted by the Regulatory Staff on behalf of the ACRS, to the applicants for the Brookwood and the Millstone Point reactors, concerning possible modes of pressure vessel failure and possible means to deal with pressure vessel failures, as well as the probability of differing types of failure.

Questions of this type had not been asked in writing frequently in the past, although discussion of such questions certainly did occur from time to time during the meetings between the ACRS and the Regulatory Staff, or the ACRS and the various applicants. We shall see that the question of Class 9 accidents (accidents exceeding the consequence limits of Part 100) was again to arise in a few months and would introduce, in a public way, the likelihood that a loss of containment integrity would be associated with accidents involving gross melting of the core.

It is clear from the minutes of the 71st meeting, March 10-12, 1966, as well as the minutes of the previous two meetings of the ACRS, that the Committee had decided in the cases of the Brookwood and Millstone Point reactors, that additional design measures to protect the public against postulated pressure vessel failure would not be recommended. Instead, improved quality control during fabrication of the vessel and improved surveillance methods during operation were going to be pursued.

Indian Point Unit 2 posed a somewhat more complicated problem. This was the highest power PWR to be reviewed to date. There was already a smaller reactor at the site (which was the most populated site approved for a reactor having a power of several hundred megawatts). To some, Indian Point represented a nearly metropolitan site, because New York City was less than 30 miles away and there was a considerable population density between the reactor and New York City. To others, Indian Point represented a better location than the Edgar site recently proposed by Boston Edison; it was very much better than the Ravenswood site in New York City. The application for Indian Point was submitted in December 1965, and it was hoped by the applicant and the AEC to get completion of Regulatory Staff and ACRS action by about June 1966, a rather remarkable pace.

The reactor had a minimum exclusion distance of only 0.3 miles; the nearest boundary of Peakskill, the closest population center, was 0.87 miles. In view of the short distances involved in this case, it was evident to the AEC Regulatory Staff that the specifics of Part 100 were not too meaningful. They elected to evaluate off-site doses for an exclusion distance and low population zone (l.p.z.) of 0.32 miles and 0.67 miles, respectively, to see if the expected leakage rate and the proposed measures to remove radioactivity from the containment (sprays and/or filters) would enable a meeting of the dose guidelines. They were following the traditional approach of Part 100, but with a very small low population zone. If it passed this test, it appeared that they would approve the reactor.

To the members of the ACRS, Indian Point looked like a site much worse than the others they had been considering. A major question facing the ACRS members was, "Should measures to cope with pressure vessel failure be applied in some way for Indian Point 2?" Also considered seriously was whether other, additional safety requirements were desirable or needed for this facility. Opinion among the Committee members on these difficult questions was clearly quite diverse.

A partial illustration of the complexity of the problem and a discussion of some of the general considerations involved can be obtained from the questions on the following page written by member Thompson to the other ACRS members during the Indian Point review.

In May of 1968, Morton Libarkin, an ACRS Staff Engineer, prepared a brief summary of ACRS actions during the construction permit review of Indian Point 2, two years earlier. The following excerpt from this memorandum covers the period March 30, 1966 to June 11, 1966.

> <u>3/30/66 - Indian Point-2 Subcommittee Meeting</u>
>
> The question of the adequacy of a "pipe break" MCA as opposed to consideration of a catastrophic pressure vessel failure was raised due to the high power level and population density. (Project Status Report dated 4/1/66, with meeting summary.)
>
> <u>4/4/66 - 72nd ACRS Meeting</u>
>
> Westinghouse described the pressure-time history in the Indian Point-2 containment, assuming no safety injection following a LOCA. A rise in pressure when the core melted and dropped into the water in the bottom of the vessel and a second increase when the core melted through into the vessel cavity were described.

I. <u>Detailed Considerations:</u>

1. What provisions appear acceptable to prevent loss of coolant accidents?

2. What reactivity anomalies can be envisioned and might be acceptable without causing concern in regard to nuclear transients?

3. Can acceptable provisions be made to ameliorate the consequences of an accident:

   (a) Concerning decay heat removal following an accident to prevent subsequent over-pressurization of the containment?

   (b) Concerning leakage from the containment?

   (c) Concerning pressure vessel rupture?

4. Can steps be taken to add assurance in regard to the acceptability of sites by:

   (a) More monitoring and local disaster alarms?

   (b) Flooding possibilities by lowering the reactor?

   (c) Evacuation of the local population? (Items (a) and (b) have very difficult public relation aspects)

II. <u>Is there reasonable assurance that this reactor can be built without undue hazard to the health and safety of the public</u>.

If the answer to II is "yes", then the Committee has no problem. If the answer to II is "no", then what one or more features of I above (or others not listed) would have to be improved and how much before you would accept it?

Several members expressed the opinion that, for the Indian Point site, the applicant should show that any failure of the pressure vessel can be withstood.

5/3/66 - Indian Point-2 Subcommittee Meeting

The vessel head rise due to stud and circumferential failures and the consequences of longitudinal splits in the vessel were discussed. A suggestion was made that Westinghouse consider putting some of the post-accident heat removal equipment inside the containment.

The applicant was asked to discuss means of ameliorating the consequences of pressure vessel failure with the full Committee.

5/5-7/66 - 73rd ACRS Meeting

The Committee was divided with respect to the requirement for protection against containment failure due to large (pressure) vessel accidents at the Indian Point-2 facility.

Westinghouse described mechanical effects of various types of vessel splits. In was noted that duplicate core cooling equipment within containment was being considered. Analyses indicated that a molten core would penetrate the containment 4500 seconds after a pipe break.*

Con Ed was informed that the Committee was still considering the requirement of protection against vessel failures.

6/8-11/66 - 74th ACRS Meeting

The Committee again raised the question of the acceptability of the Indian Point-2 safeguards equipment. Some members felt that the penetration of the containment by the vessel head was a serious enough problem to justify retention measures. Also, if the vessel failed, a relocated, molten core would probably violate the containment liner, etc.

The Committee agreed, by a soft vote, to require protection against the effects of a longitudinal vessel split.

*Westinghouse may have actually said that the pressure vessel would be penetrated 4500 seconds after a pipe break.

The position taken at the June, 1966 meeting remained the position of the Committee for Indian Point 2, and, in fact, represented the continuing position of the Committee for the Indian Point 3 reactor, the Zion 1 and 2 reactors, which were at a site having a surrounding population density similar to that of Indian Point, and also for the Midland reactor, which had a relatively large nearby population.

The integrity of the reactor vessel cavity was to be maintained in the unlikely event of a longitudinal vessel split, but there was no accompanying requirement that the core be kept from melting in connection with this unlikely event. In part, the ACRS recommendation that protection be provided against the forces involved with longitudinal vessel split appears to be related partly to the higher probability of this type of failure, compared to failure of all the studs, or a circumferential vessel failure. Partly, it appeared to be more practical to design against this particular set of forces. And perhaps, partly, it was a way of initiating what might later be more comprehensive protection against vessel failure for still more highly populated sites.

A more detailed look at the minutes of the 72nd meeting in April, 1966, shows that there was considerable discussion concerning pressure vessel quality and failure modes with the representatives of the applicant for Indian Point 2. It was stated by Westinghouse that the reactor design could probably withstand a longitudinal failure of the pressure vessel, but was not clear as to whether a circumferential break or head loss could be also handled.

In Executive Session, ACRS member Palladino was inclined to require the Indian Point group to show that any pressure vessel failure could be withstood. Member Etherington saw the biggest question as "What reactor design the Committee is willing to accept for a metropolitan site; the measures for the much higher hazards of this site are not clear." Member Bush sensed that the engineered safeguards were either insufficient for this site or overdesigned for some other sites.

Similar discussions ensued at the May and June, 1966 ACRS meetings, and at the Subcommittee meetings on Indian Point. In addition, a much more intensive examination was given to the adequacy and reliability of engineered safeguards for Indian Point 2 by the ACRS than had been the case for previous reactors.

Concurrent with the review of Indian Point 2, the ACRS held further Subcommittee meetings (May 4, 1966 and June 3, 1966) on the matter of metropolitan siting. The minutes of the Subcommittee meetings are duplicated on the next pages, essentially in their entirety, for their very considerable insight into the thought process and the problems involved. Discussion papers prepared by Subcommittee Chairman Etherington for the May 4th meeting are also duplicated, for the insight they provide and to help make the meeting minutes more readily understood.

These minutes report discussions held with Dr. Beck and members of the staff of Brookhaven National Laboratory on their redo of WASH-740 (1957). Following the May 4, 1966 Subcommittee meeting, the full Committee heard a similar discussion. The minutes of this discussion, which are also duplicated on the following pages, indicate a reluctance of Dr. Beck to disseminate any quantitative results from the study. The Subcommittee meeting held on June 3, 1966 was the result of considerable pressure by the ACRS on Dr. Beck to make available some representative results to the ACRS.

One specific item of interest in the BNL work was the expectation that the intestinal dose, rather than the whole body or iodine dose, would have the most important health effects for uncontrolled release of the bulk of the radioactivity.

An item of special interest in the minutes of the June 3, 1966 Subcommittee is that Dr. Wensch and Dr. Beck reported that core melt in a 3200 MWt reactor would not only lead to melt-through of the reactor vessel but that calculations indicated the core would melt through the concrete of the containment floor into the earth* until enough material was involved to dissipate its heat.

*The term "China Syndrome" was quickly coined to characterize a core melting its way into the earth (on its way to China from the U.S.).

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 29, 1966

Discussion Outline for
Subcommittee Meeting on Reactor Design and Operating Criteria
May 4, 1966

1. Review of previous DRL and ACRS positions. In general DRL has been concerned over small reasonably probable accidents and has been inclined to declare a moratorium on large city locations as a matter of policy. ACRS has favored an open attitude and a case-by-case review; the Committee has generally expressed a belief that engineered safeguards should permit siting of reactors closer to cities, but believes that improved safeguards are required, including protection against pressure vessel failure.

2. Are major accidents in city locations worse than in some locations already approved?

3. Can criteria for city locations be established? Discussion to be based on H. Etherington memorandum (distributed by R. F. Fraley memo dtd. April 14, 1966) or on some other basis.

4. Should a position be taken on the Boston Edison Weymouth plant location:

   (a) on the basis of interim criteria, or
   (b) as an interim policy decision.

5. Should the staff be invited to discuss the Committee position or to implement its recommended actions?

6. Should the Committee make a determined effort to develop at least an interim position?

H. Etherington by RSS

H. Etherington
Chairman, Reactor Design and
Operating Criteria Subcommittee

EVALUATION OF REACTOR SAFETY FOR CITY LOCATIONS

The purpose of the May 4 subcommittee meeting is to develop a suggested framework for discussion at the Full Committee May meeting. This memorandum is suggested to the subcommittee as a possible starting point for its own discussions.

I. The Problem

Evaluation of "reasonable" hazard requires:

1. Appraisal of the maximum and probable consequences of various accidents.

2. Appraisal of the probability of various accidents.

3. Selection of a formula for combining consequences with probabilities to express acceptable hazard.

Consequences. The immediate consequences of a particular accident can be estimated on a reasonable set of assumptions, both for the worst environmental and meteorological conditions, and for some defined mean of all possible conditions.

Probability. There is no available means of evaluating the probability of accidents that have happened only occasionally or not at all.

Combination of Consequences with Probability. There is no agreed method of combining consequences with probabilities. For example, it is not established whether a 10-fold increase in the severity of consequences is properly compensated for by reducing the probability by a factor of 10.

II. Consequences of Accident

Evaluation of the consequences of an accident to a particular reactor depends on:

1. The nature of the assumed accident.

2. The criteria for accident evaluation.

3. The surrounding population.

4. The assumed environmental and meteorological conditions.

5. The exposure duration.

Assumed Accident. The accidents that appear most pertinent to present considerations are:

(a) The MCA with full functioning of engineered safeguards.

(b) The MCA with partial functioning of engineered safeguards.

(c) The MCA with complete failure of all engineered safeguards except the containment.

(d) The "Extreme Accident", assuming 100 percent core meltdown and complete loss of containment.

Case (a) is presumed to have no consequences beyond Part 100. It is suggested that initially only the Extreme Accident be considered.

Criteria for Accident Evaluation. Consequences could be evaluated in many ways, for example:

1. Number of fatalities.
2. Number of persons receiving over 250 r whole-body dose.
3. Number of persons receiving over 25 r whole-body dose.
4. Number of persons receiving over 300 rem iodine dose.
5. Total man-rem dose to the population.
6. Number of people receiving over 600 rem, 600-500, 500-400, etc.

Some weighted average of such criteria should be used, but for initial discussion it is suggested that the number of fatalities be used as the criterion.

The Surrounding Population. The existing population density and distribution are known for a particular site. Future population is also important, but it is suggested that initial study be limited to the existing population.

Assumed Environmental and Meteorological Conditions. The severity of consequences and the probability of a particular weather regime are both important. Possible assumptions include:

(a) The consequences from the worst possible combination of population distribution and weather regime.

(b) The TID 14844 assumptions applied in a direction that affects the greatest number of people.

(c) Average consequences around the compass based on TID 14844 assumptions and population distribution.

(d) Average consequences around the compass based on population distribution, and on weather-regime probability as determined by meteorological records for the site.

These alternative assumptions are in order of decreasing severity. Probably some weighted combination of assumptions (a) and (d) is most meaningful, i.e., a combination of worst possible and the average conditions. However, for the present purpose it is suggested that assumptions (b) and (c) be used (TID 14844 for worst direction, and for average of all directions).

Exposure Duration. The anlysis could be based on planned evacuation or on continued exposure. Two-hour evacuation may be impractical but a general exodus may be expected after a major accident. The probable exposure duration should be studied, but for the present it is suggested that calculations be based on no exclusion area and continuing exposure.

Large Reactors under Review. To focus the discussion, the criteria should be applied to large reactors currently under review. The criteria are summarized as 100 per cent meltdown, complete loss of containment, TID 14844 fission-product release, long time exposure, TID 14844 meteorology (both for worst direction and average for all directions), and number of fatalities. The following table should be developed correctly and extended to other assumed criteria of consequences.

Fatalities in Extreme Accident*

	TID 14844, worst direction	TID 14844, average direction
Brookwood	5,000	1,500
Millstone Point	5,000	2,500
Indian Point	13,000	5,000
Boston Edison	70,000	15,000

* These numbers are not to be taken seriously. They are guessed from a cursory inspection of a tabulation "Potential Consequences of Breached Containment" received December 28, 1965 from the DRL staff. The numbers should be reduced by some factor for shielding by buildings.

III. Probability of Extreme Accident

Only the Extreme Accident (complete meltdown and complete loss of containment) is considered in this exploratory discussion. To appraise the probability of the Extreme Accident and to review methods of reducing the probability, it is necessary to consider:

1. Initiating accidents or chains of accidents.
2. The relative probability of each link in the accident chain.
3. Measures that can reasonably be taken to reduce the likelihood of the accident.

Initiating Accidents. Some possible initiating accidents are:

(a) The MCA with failure of engineered safeguards (including containment).

(b) Reactor vessel rupture and containment penetration.

(c) Sustained power failure coupled with emergency power failure and failure to find a last-ditch remedy.

(d) Earthquake or other natural phenomenon.

Absolute Probability of Initiating Accidents. If it can be reasonably assumed that the probability of one initiating cause is much greater than all the others, then only that cause need be considered. If, however, two or more causes have probabilities of the same order of magnitude, there is little justification for concentrating on one cause while ignoring the other.

It is almost useless at present to suggest absolute probabilities, but in order to sense the feeling of the Committee on relative probabilities, the following clay pigeons are offered:

(a) MCA 10^{-4} over the reactor life. Containment failure following MCA (by leakage, valve failures, rupture from overpressure or local overstress, improper maintenance, etc.) 10^{-2}. Over-all probability 10^{-6}.

(b) Vessel rupture during its life (material, design, and fabricating inadequacy; overpressure from reactivity insertion; or failure to keep safely above NDT) 10^{-6}. Missile penetration of containment with no holddown 1. Over-all probability 10^{-6}.

(c) Sustained power failure during life (once in 400 years) 10^{-1}. Failure of diesel to start up (if essential) 10^{-2}. Failure of operator to take extraordinary action (e.g., make a field connection to some other source of coolant) 10^{-2}. Over-all probability 10^{-5}. A second diesel of adequate capacity would reduce this to 10^{-7}.

(d) The probability of a destructive earthquake is high in some areas. This probability must be multiplied by the probabilities that the earthquake would cause a particular series of events (e.g., loss of power or coolant).

IV. Combining Consequences with Probabilities

A determination of how consequences and probabilities should be combined would do much to clarify the problem of appraising reasonable hazard as a function of reactor location.

Simple Product. The most obvious treatment is to require that a product of consequence (fatalities) multiplied by probability have a prescribed constant maximum value, i.e. Fp = C. Suppose one fatality in the 40-year life of the plant were considered a tolerable industrial hazard and suppose the average weather conditions produces one tenth of the TID 14844 number for average wind direction. If the extreme accident for average wind direction would cause 1,000 fatalities, the probability of the accident in 40 years must not exceed 10^{-2}; for 10,000 fatalities, 10^{-3}; and 100,000 fatalities 10^{-4}. For the Boston Edison reactor we would then have to decide whether the probability of vessel rupture (or of other cause of extreme accident) was less than 1.8×10^{-3} or 1 in 444. The probability must, however, be reduced (say by a factor of 10) to allow for non-fatal injury, giving 1.8×10^{-4}.

Weighted Product. The simple product appears to be a valid measure of hazard if operations are extended over an infinite time, with statistically even spacing of events.

The simple product does not take into account the difference in public reaction to numerous small incidents and to rare catastrophic events. Certainly, frequent small incidents would provoke a needling type of criticism; but a major catastrophy early in the industry could cause every reactor in the country to be shut down. The problem here is not the frequency of occurrence, but the uncertainty as to when the first accident of this category might occur — if such an accident occurred for the first time hundreds of years from now, it might be accepted in proper perspective. Some higher-power weighting in recognition of the magnitude of the extreme accident therefore appears necessary.

Suggested Procedure. It is suggested that some kind of tentative formula be developed to compare reactors and sites. Some possible approaches are discussed.

(a) Accept the Brookwood plant and location as a reference model. Decide which is the most probable way by which the Extreme Accident could occur. For proposed reactors involving a more severe Extreme Accident, require that the probability be reduced by:

$$(\text{Consequences at proposed site/Brookwood consequences})^2$$

For Boston Edison this would require (based on fatalities for worst wind direction) some engineered safeguard that would reduce the probability by a factor of 196, or, in orders of magnitude, 100. If vessel failure is the most probable cause and is considered sufficiently probable, it seems unlikely that added inspection or surveillance would give so large a factor, and on this basis some degree of hold-down would be indicated. If, however, failure of an engineered safeguard were the most probable cause, a backup system might supply the additional safety required.

(b) Use the Brookwood plant and location as a reference, but first decide whether the probability of the Extreme Accident is much less than 1.8×10^{-4} or 1 in 4,000. Make an adjustment and proceed as before.

(c) Use a less drastic formula than the square of the consequences.

V. Pressure Vessel Requirements

The vessel failure risk that may reasonably be taken should be reviewed and its status reappraised on the basis of conclusions that may be drawn from the foregoing considerations.

Westinghouse and GE requirements in excess of Sec. III of the ASME Code should be tabulated and reviewed.

H. Etherington
April 9, 1966

MEETING OF

REACTOR DESIGN & OPERATING CRITERIA SUBCOMMITTEE

WASHINGTON, D.C.

MAY 4, 1966

This was a half-day executive session to discuss the siting of reactors near large cities.

Attendees:

H. Etherington
S. H. Hanauer
D. Okrent
S. H. Bush
J. E. McKee
N. J. Palladino
R. F. Fraley, Staff
R. H. Wilcox, Staff

Mr. Etherington stated the purpose of the meeting and proposed an agenda to be followed. Additional topics were suggested by the ACRS members present.

1. Review of previous DRL and ACRS positions. In general DRL has been concerned over small reasonably probable accidents and has been inclined to declare a moratorium on large city locations as a matter of policy. ACRS has favored an open attitude and a case-by-case review; the Committee has generally expressed a belief that engineered safeguards should permit siting of reactors closer to cities, but believes that improved safeguards are required, including protection against pressure vessel failure.

   A recent reporting of a speech by Dr. Beck in "Nuclear Industry" appeared to imply that the Regulatory Staff opposed construction of reactors in cities for the present. A "Nucleonics Week" article, on the other hand, quoted a DRL official as saying that DRL had an open mind, would review anything proposed, and encouraged imaginative approaches.

   Dr. Hanauer noted that the ACRS 1964 letter on engineered safeguards was related to the metropolitan siting question. The Regulatory Staff's proposed "moratorium" on siting of reactors in large cities was effectively blocked by the Commission based upon advice from the ACRS. The General Manager's side of the AEC reportedly agreed with the ACRS in this case. Out of this grew the "augmented safety research program" and the present "Steering Committee" for safety research.

It was agreed to ask the Regulatory Staff what its present feeling was at the full Committee meeting the next day.

2. Are major accidents in city locations worse than in some locations already approved?

   It was believed that Dr. Gifford might have some thoughts on this and also that the meeting the next day with Brookhaven representatives would be helpful.

   Dr. Hanauer was concerned not only with the uncontained accident, but with the MCA. He was not sure how many people should be given 25 r (whole body) and 300 rem (thyroid).

   Dr. Okrent noted that a man-rem approach might be used, and that 10 rems to 10^6 people might be unacceptable. Dr. Hanauer pointed out that Con Ed claims only a 50 mr site boundary dose for Indian Point 2, thus it depends on who calculates the man-rems. Applicants usually assume all safeguards work.

   Mr. Fraley noted that applicants are now claiming they can meet Part 20 in an MCA. Dr. Hanauer feels that it is a criteria question, namely what are they allowed to take credit for. Mr. Palladino did not think it wrong to have several "levels of safety", e.g., Part 20 if everything works, Part 100 for other conditions, etc. Mr. Fraley noted some feeling within AEC that Part 20 was not good enough. It was pointed out that many fossil-fuel plants put out more radioactivity than nuclear plants, and some may exceed Part 20. The question of whether Part 20 needed improvement for cities was referred to the Environmental Subcommittee by Dr. Okrent via Dr. McKee.

   Dr. Bush was concerned that the usual meteorological assumptions were not always as pessimistic as they could be. He recalled instances where particulates released from a stack all came to the ground at one small spot. Mr. Palladino had observed similar situations. Dr. Okrent felt that rainout in a city could present a problem. Mr. Etherington noted that there were worse conditions than used in TID-14844, but these were of low probability.

   There was some speculation that an accident 30 miles from a city could be worse than one in a city. Dr. Okrent noted that BNL had calculated, for an uncontained accident, that a reactor 5 or 10 miles from a city could, with good meteorology, have a wide enough plume to give the whole city a lethal dose. If the reactor was in the city, only half the populace might be so exposed. The BNL people had figured that the lethality would be due to the exposure of the intestinal tract. BNL has not been allowed to write this up in a manner that is readily intelligible. The techniques, such as the meteorological analysis, are being written up. The high cost of the accident was reportedly a principal deterrent to any public release. It was noted, however, that expensive accidents could result from a chlorine plant or from another Chicago fire, yet neither chlorine usage nor fire are prohibited.

Dr. Hanauer felt the ACRS should ask for the BNL data. Dr. Okrent had previously tried as a member of the Steering Committee to obtain such a document, but without success. He noted that the meltdown of a large reactor would produce temperatures much higher than in LOFT. Burnup, specific power, and size were important factors. Dr. Bush noted that a release fraction above 1% had serious consequences, even for shipping casks.

Mr. Etherington and Dr. Okrent took some comfort in probabilities that the wind will not be blowing toward densely populated areas. A ten mile lethal distance would affect large cities at Brookwood and Malibu and would affect Joliet for Dresden. Since inversions generally occur every night, they clearly must be considered. It was noted that the wind from Brookwood does not normally blow toward Rochester.

Mr. Palladino pointed out that lesser accidents with higher probability are more tolerable outside the city than within it. Dr. Okrent cited as an example an MCA with 1%/day containment leak rate instead of 0.1%/day. Dr. Hanauer also cited the NRU refueling accident.

Dr. Bush noted that present stack releases at Humboldt Bay might be too high. Dr. Hanauer noted that BONUS operators were donning air breathing apparatus within the containment. Allowable releases in a city was suggested for further consideration, as was the acceptability of portions of the primary system outside of containment.

3. Can criteria for city locations be established? Discussion based on H. Etherington memorandum (distributed by R. F. Fraley memo dated April 14, 1966) or on some other basis.

Mr. Etherington explained his evaluation presented in the above memorandum. He noted that minor accidents should be added to the list of "Assumed Accidents" on page 2.

There was some discussion of uncontained accidents. Dr. Okrent noted that a containment with a door open was like no containment at all. He also noted that all containments are purposely painted on the inside with a paint that doesn't pick up fission products.

As a criterion for extreme accidents, Mr. Etherington suggested the number of fatalities. Dr. Hanauer felt that man-rem considerations had to be included. Mr. Etherington felt that the number of exposures above 100 or 200 rem was also important, since it would hit the newspapers more than man-rems. Dr. Hanauer suggested that the ratio of deaths to man-rems might provide a definition of what is a city. Mr. Etherington agreed that more than fatalities should be considered.

On population, Mr. Etherington suggested that existing, rather than future, population be considered.

On meteorology, Mr. Etherington felt that probability and severity were both important. While various assumptions are possible, Mr. Etherington suggests using (a) TID-14844 assumptions in a direction that affects the most people and (b) TID-14844 assumptions averaged around the compass with the existing population distribution.

Mr. Etherington suggested that infinite exposure be considered for the present. His table of fatalities (page 3 of his memo) was only a "guesstimate", but it shows Boston Edison as an order of magnitude worse than Brookwood or Millstone Pt.

There was some discussion of Mr. Etherington's estimates of probabilities of extreme accidents. There was general agreement that the probability of vessel rupture was higher than 10^{-6}. Dr. Hanauer felt that the sustained power failure probability was more like once in 40 years than once in 400 years. Dr. Hanauer did not agree that addition of a second diesel would reduce the accident probability by a factor of 100, since he doubted that two diesels would be really independent.

On combining consequences with probabilities, a simple product was a possibility. Mr. Etherington pointed out, however, that the timing of the first big accident is quite important. If it does not occur for 500 years, this is one thing. If it happens in the next 10 years, however (or even 50), it might shut down all reactors permanently and tightly. Mr. Etherington therefore suggested that the consequences be squared. Thus, if the fatalities were ten times as high, the probability would have to be one-hundredth as much. If Boston Edison is 14 times worse, the probability would have to be 1/196th as much. He felt such a reduction in probability depended on the accident, e.g., another diesel might reduce the power failure probability this much, but, for vessel failure, holddown might be required. (Improved inspection would not be expected to add this much confidence.)

Dr. McKee suggested that curves of log (man-MW) vs. distance (as have been distributed to ACRS members by R. H. Wilcox in the past) be corrected by multiplying by two factors. One factor, E, would be an environmental factor, to be given a number from one (best site) to five (worst site) based upon the judgement of the ACRS as to meteorology, earthquake, tsunami, etc. The other factor, D, would be a design factor, which would give credit for double containment, engineered safeguards, etc., and also be applied as a judgement factor between 1 and 5. Dr. McKee proposed that the ACRS draw a line on the resulting graph defining regions (to the left) which would not be acceptable, and regions (to the right) which would be acceptable.

The approach was discussed briefly, and Dr. McKee agreed that the 1 to 5 numbers might be improved upon.

Dr. Okrent suggested that large reactors be required to protect against accidents differently depending on location. He was not necessarily proposing this, but felt the Committee had been tending in this direction. The following table is illustrative of this approach:

Reactor Location	Protection Against Pressure Vessel Failure Modes	Containment Design	Engineered Safeguards	Chemical Reactions
City	All	Double	Extra diesel	$Zr-H_2O$ up to ZrO_2 melting temp.
Suburban	Longitudinal	Single, but specially conservative & testable	--	--
Rural	High Quality	--	--	--
Remote	--	--	--	--

Dr. Okrent noted that small ship reactors in cities could be provided for in such an approach.

There was some discussion of what is city, suburban, etc. Dr. Okrent regarded Indian Point as suburban. A suburban reactor with a poor environmental factor could be shifted thereby to the city category.

Dr. Okrent felt that the ACRS could recommend such an approach as being "prudent" and so tell the nuclear community. There was general agreement that new ideas should not be precluded by any such criteria.

For city reactors, the plant would have to meet the most stringent criteria in each category. Protection against small accidents would constitute an additional column.

Dr. Hanauer asked about use in cities of safeguards equipment never built before. He felt that the learning must be done by putting these things on reactors not in cities. Dr. Okrent agreed and felt that the novel aspects should be resolved and the workability of the engineered safeguards demonstrated prior to the city reactor construction permit.

Dr. Okrent pointed out one avenue as being AEC support to build a city reactor not in a city. Dr. Hanauer felt that a utility such as Con Ed should support this. Mr. Fraley noted that some utilities think they have been doing this with the engineered safeguards on plants now being built. Since plants rarely fail and give the safeguards a chance to prove themselves, however, the safeguards must be tested in a safety research program. Little of this is now being done. (CSE will do some of it.)

Mr. Fraley suggested that a site be considered rural if no evacuation is required; suburban if evacuation is counted on; and city if it is impossible to evacuate. Dr. Okrent felt that Joliet and La Grange, Illinois had to be considered cities.

There was brief discussion of the dollar-cost of an extreme accident. The $500 million Price-Anderson coverage would cover 10,000 fatalities at $50,000 per death. Dr. Okrent felt, however, that with land clean-up costs, etc., 1000 deaths would exceed $500 million.

4. Should a position be taken on the Boston Edison Weymouth plant location (Edgar Station):

   (a) on the basis of interim criteria, or
   (b) as an interim policy decision.

   A new containment design will probably be proposed, but the reactor vessel head blowing off will still go through the roof. Mr. Etherington wondered if the ACRS couldn't now say that the head had to be held down. Other problems noted were the turbine outside containment and off-gas stack releases. Dr. Hanauer also suggested that there comes a point when there are too many safeguards that have to work (e.g., Indian Point has 3 or 4; clearly 10 is too many). Mr. Palladino was inclined to vote no for the Boston Edison proposal at that location, while not ruling out the site for a design with real good containment, missile protection, etc. External missiles were also suggested as a problem, but Dr. Okrent felt that enough concrete might be provided to protect against an airplane crashing in.

5. Should the Staff be invited to discuss the Committee position or to implement its recommended actions?

   It was agreed to ask the Staff for its present position on metropolitan siting.

6. Should the Committee make a determined effort to develop at least an interim position?

It was pointed out that the ACRS now has no city reactor case before it, but also that Boston Edison was coming in and that the California Department of Water Resources Oxnard site (300,000 pop. in 10 miles) would be in for ACRS comment in June. Dr. Okrent felt that implementation of the ACRS pressure vessel letter was involved. He also thought that the ACRS should develop its own standards on what it will expect of suburban and city reactors. He did not want to wait for Boston Edison to start thinking about this, although he believed it might not be resolved until then.

It was agreed that Mr. Etherington's memorandum should be distributed to the full ACRS. For discussion with the full Committee the next day, both man-rem and number of fatalities approaches were suggested. The approaches suggested by Dr. McKee and Dr. Okrent were to be summarized and discussed with the full Committee.

* * * * *

DRAFT MINUTES
OF
REACTOR DESIGN AND OPERATING CRITERIA
SUBCOMMITTEE MEETING
ON JUNE 3, 1966

Purpose:

This meeting was to discuss the development of guides or criteria for the siting of metropolitan reactors.

Attendees:

ACRS

H. Etherington
D. Okrent
S. H. Bush
S. H. Hanauer
H. O. Monson
W. K. Ergen, Consultant
R. F. Fraley, Staff

Division of Reactor Development and Tech.

G. Wensch

Regulatory Staff

C. K. Beck
P. Morris

Discussion:

Mr. Etherington noted that several documents related to this meeting have been distributed by the ACRS Office.
These are as follows:

(a) WASH-3 (Rev.) Summary Report of Reactor Safeguards Committee, dated March 30, 1960 (distributed to ACRS Members by R. F. Fraley Memo dated May 20, 1966).
(b) Draft notes prepared for site criteria Sub-committee meeting - F. Gifford, February 16, 1959 (distributed to all ACRS Members by R. F. Fraley Memo dated May 31, 1966).

(c) BNL-10137 - Site Selection in Relation to Engineered Safeguards. An American View (H. Kouts paper distributed by R. F. Fraley Memo dated May 19, 1966).

(d) BNL-10138 - Research for Reactor Safety in the United States (H. Kouts paper distributed by R. F. Fraley Memo dated May 19, 1966).

(e) Memo to File of DRL Meeting with General Electric on May 22, 1963, Concerning Accident Evaluation Methods (distributed by R. F. Fraley Memo dated May 24, 1966).

(f) Agenda for June 3, 1966 meeting on Metropolitan Locations for Power Reactors by H. Etherington (distributed by R. F. Fraley Memo dated May 19, 1966).

(g) Memo from J. E. McKee to Harold Etherington, dated May 7, 1966, Subject: Qualification of Criteria for Reactors near Big Cities (distributed by R. F. Fraley Memo dated May 19, 1966).

(h) R. H. Wilcox Site Comparison Curves for major U. S. Power Reactors (distributed by R. H. Wilcox Memo dated May 18, 1966).

(i) Summary of discussions at the 73rd ACRS Meeting on Criteria for Location of Reactors in Cities (distributed by R. H. Wilcox Memo dated May 13, 1966).

The discussion at this meeting followed the outline of Item (f) above.

The attached comments, provided by Dr. W. K. Ergen, were distributed.

I. Normal Effluents and Releases up to the MCA

Mr. Etherington noted that a question has been raised as to whether or not Part 20 limits are restrictive enough when very large numbers of people are exposed to the routine releases involved.

Dr. Ergen explained that the recommendations of the International Commission on Radiation Protection (ICRP) allowed higher doses to "neighbors" of reactors and to the "group most affected" than the population at large. A definition of these groups and/or the limiting number of people involved in these groups is not specified however.

Dr. Okrent suggested that an additional question is whether or not a plant should be permitted to design up to the limits of 10CFR Part 20 for normal operation. It appears that the ICRP intended that lower figures be used as the design basis for normal operation.

Dr. Hanauer also noted that a set of limits might be considered for the class of abnormal situations between normal operations and the MCA (e.g., accident situations which have a much higher probability than the MCA). He noted that it is now only necessary for an applicant to demonstrate that the resulting doses are less than the MCA. Dr. Ergen suggested that these smaller accidents are the financial responsibility of the operators and prevention should, therefore, be left to them. He proposed that the ACRS might concentrate on those accidents with consequences greater than $74 million dollars since these are the financial responsibilities of the Government.

Dr. Okrent noted, however, that in its draft report of March 1965 on metropolitan siting of reactors the Committee expressed its concern regarding the small as well as the large accident for city sites.

Conclusion:

It was agreed that additional clarification should be sought of the number of people to which ICRP standards apply. The Subcommittee concluded that this should be an assignment for the Environmental Subcommittee.

Dr. Okrent proposed that no action be taken to establish dose limits for the intermediate-size, more probable accidents but that the Committee should continue to give attention to this class of accidents on a case-by-case basis and be more restrictive for those reactors located close to population centers. This suggestion was endorsed by the Subcommittee.

II. Examination of the Extreme Accident and Its Consequences

Mr. Etherington noted that very bad reactor accidents (e.g., uncontained accidents of the WASH-740 type) can result in higher doses than 10CFR Part 100 limits. He suggested that the ACRS should establish an acceptable limit for the consequences of this type accident. Ten thousand (10,000) fatalities was discussed as a possible limit.

Dr. Ergen suggested that the reactor industry has come of age and the hazards should be considered in light of the hazards represented by other competing industries. For example, if one becomes too restrictive in safeguards requirements for reactors it will force construction of other types of power producers such as dams. Since dams do fail occasionally with resulting casualties one has accomplished little in overall safety.

Subcommittee members questioned that reactors have come of age, however, and proposed a study to determine what public reaction might be from a major reactor accident.

Dr. Bush noted that the timing of this accident would have a major effect on the public reaction. For example, an accident in the first year would have a much more serious effect than the same accident if it occurred in the 1,000th year. There was also discussion of the effects that frequency vs. consequences might have. It seemed generally agreed that infrequent but serious accidents would have a more adverse effect than frequent but less serious accidents.

There was considerable discussion of the consequences (e.g., no. of fatalities, man-rem) which the ACRS should consider acceptable vs. the probability of such an accident. It was also suggested that additional information is needed concerning the effect of the site location on the consequences of an accident of this type. For example, does a metropolitan site really make a difference for an accident of this magnitude. It has been proposed on several occasions that the site, especially when city distances of 20-30 miles, are involved, makes little difference in the consequences of a severe accident; although the probabilities may be changed somewhat by the wind rose, etc. Dr. Okrent suggested that some thought should be given to the development of a nation-wide evacuation plan for reactors similiar to that, and perhaps associated with the plan, for civil defense.

Dr. Okrent described the British system wherein the engineers assign appropriate probabilities to the various accidents considered and the hazards evaluators then decide if these probabilities and the related consequences are acceptable from a national risk standpoint.

Dr. Ergen suggested, however, that there is a lower limit which can be assigned to a catastrophic accident which would preclude reactors from cities unless one were willing to accept some serious consequences or the possibility of evacuation. This lower limit on probability would be determined by the lack of 100% assurance that engineered safeguards would function as designed when needed. Dr. Okrent maintained that credit must be given for the operating experience of proven reactor designs and for proven engineered safeguards. If this is not done and one postulates enough pessimistic assumptions every major reactor in the in the country would kill more than 10,000 people if a serious accident occurred. Dr. Okrent suggested that the ACRS should insist on improved component quality, better evacuation plans, etc. as reactors move closer and closer to cities.

<u>Conclusion</u>:

The Subcommittee was unable to reach a decision on the number of fatalities which can be accepted from a serious reactor accident and how this might be adjusted for the frequency of accidents vs. the consequences.

It was agreed that this is a very basic question which must be resolved by the Full Committee.

Dr. Ergen was to develop additional information concerning the ability to evacuate heavily populated areas.

II.1. Tentative Model for Estimating Consequences

It was noted that a model is needed for evaluating the consequences of a serious reactor accident. For instance, recent BNL work indicated that it is the intestinal dose that is limiting rather than the thyroid or whole body doses when lethal effects are considered. Dr. Ergen also suggested that evacuation can be used to counteract the effects of poor diffusion conditions if one is willing to take credit for it.

Conclusion:

It was agreed that Dr. Ergen should work up an appropriate model for use by the ACRS.

II.3. Possible Causes of the Extreme Accident

Sabotage, plant deterioration and/or sloppy operation and exposure of components to conditions beyond design limits were suggested as possible causes in addition to those listed in Item (f).

Presentation by Representatives of AEC Staff

G. Wensch and C. Beck described some of the recent BNL work related to re-examination of the consequences of a serious reactor accident.

BNL studies were based on a 3,200 MW(th) reactor of a type similiar to a typical water power reactor. A Loss-of Coolant Accident was assumed to occur and no credit was given for engineered safeguards or phenomena unless there is information available to prove they will function. The plant blows down and all water is boiled off in about 6 hours. No metal-water reaction was considered. The core then melts and fission products are released from the fuel. The core will melt through the pressure vessel and become molten in about 10 hours. Calculations indicate that the core from a 3,200 MW(th) reactor would melt through the pressure vessel, the concrete of the containment floor and the containment vessel into the earth until enough material was involved to dissipate its heat. For a 1,000 MW(th) reactor the core may actually solidify before it melts through the containment if enough foreign material is incorporated to increase its heat transfer characteristics.

In any event, fission products will escape from the molten mass until all of the significant isotopes are released (10-12 hours).

The rare earths and alkaline earths are released slowly but the volatile oxides (ruthenium, molybdenum, and technetium), cesium and the other more volatile isotopes are evolved more rapidly.

All isotopes were considered by BNL except for the very short and long lived isotopes and the very low yield fission products. A table of the isotopes considered is included in the attached handout (Attachment 4). The release rates were based on experimental data from irradiated uranium oxide fuel which was melted in a steam-air atmosphere. A reduction factor might exist for release rates from a large mass of molten material but since no data exists other than for small samples these data were used.

Plating-out of fission products may occur inside the biological shield but since the fission product heat would probably revaporize it, no reduction factor was taken for plate-out. Once the fission gases pass out into the larger cooler air mass of the containment they are transformed to aerosals by cooling and agglomeration so that plate-out would not occur in this area either.

Doses to the intestine, whole body and lung were then calculated based on the assumption that all of the fission products released inside the containment were breathed by an individual with no credit for atmospheric diffusion, decay in transit, fall-out, etc.

Dr. Okrent noted that some work had also been done by BNL to take into account the effects of diffusion, etc. on a recipient some distance from the reactor. Dr. Wensch agreed that some preliminary work had been done by BNL in this area and agreed to check to determine what information is available for use by the Committee. Dr. Beck suggested that if the ACRS desires that a report be prepared on this topic, the Committee should request it formally from the Commission since BNL is not committed to do any additional work on this subject. He noted that the conclusions from the BNL study are contained in letters from Chairman Seaborg to Representative C. Holifield and from Commissioner Palfrey to Mr. David E. Pesonen. Copies were distributed and are attached.

Biological effects are based on the chemical forms which would exist as the fission products would evolve from UO_2 fuel in a steam-air atmosphere. Distribution in the body is based on inhalation with no ingestion except as coughed up from the lung and swallowed. Strontium is not a significant contributor since it is not vaporized from the molten fuel to any degree. Plutonium was included as an isotope which would be produced in a low enrichment uranium fuel.

It was noted that a lethal dose to the intestine was considered to be 1,200 rem by BNL. The lethal dose to the lung is 3,000-4,000 rem.

A copy of the draft BNL report which describes the fission product release model, etc. was distributed and is attached (Attachment 4).

Dr. Wensch noted that additional data might be developed as part of the safety research program to better define the release of fission products from large masses of molten fuel, agglomeration behavior of fission products after release from the fuel.

Executive Session:

Dr. Okrent noted that much of the work outlined at this meeting would take considerable time to develop and suggested that the Committee needs to continue work on a set of guides that can be applied in the near future to the evaluation of reactor proposals. He proposed that continued effort be directed to the development of a table similiar to that discussed at the 73rd meeting which would identify the design and safeguard requirements for reactors at different sites.

Dr. Hanauer noted that the ACRS has yet to decide what is a metropolitan site.

Dr. Okrent also noted that visits of nuclear merchant ships to heavily populated ports is a problem that will have to be evaluated in the near future.

Attachments:

1. ACRS Subcommittee on Reactor Design and Operating Criteria, June 3, 1966 Meeting on Metropolitan Locations for Power Reactors, Comments by W. K. Ergen on the Points on H. Etherington Agenda.

2. Letter from Chairman G. T. Seaborg to Honorable Chet Holifield, dated June 18, 1965.

3. Letter from Commissioner John G. Palfrey to Mr. David E. Pesonen, dated October 8, 1965.

4. Draft report entitled Exposure Potentials and Criteria for Estimating the Cost of Major Reactor Accidents by A. J. Court, F. P. Cowan, K. Downes, J. B. H. Kuper.

* * *

<u>May 5-7, 1966 - 73rd ACRS Meeting - Large Accident Hazards</u>

Mr. Downes and Miss Court, Brookhaven National Laboratory (BNL), joined the Committee to comment on the report entitled "Exposure Potential and Criteria for Estimating the Cost of Major Reactor Accidents". Mr. Price, Dr. Morris and Dr. Beck also joined this session. The study was originally motivated by an interest in economics for insurance purposes. Extensive computer codes were arranged to assess the hazards from the many isotopes (55 fission products were assumed) which might be released from a very large reactor accident. Both volatile and non-volatile fission products were considered. Many parameters, e.g., meteorology, population, biological effects, were factored into the code.

The radiation exposure to the lower large intestine, perhaps a 1000 rem, from ingested fission products was found to be controlling as far as fatalities were concerned. The external dose, e.g., from a cloud, might be only 100 rems and hence less important. Iodine was not considered controlling for such a large accident, because thyroid loss can be compensated for by medicines or surgery. Dr. Bush said that any alpha emitter would have no effect in the intestinal tract unless there were bleeding ulcers.

The reactor accident of the study was somewhat different than what the Committee normally assumes; the containment was assumed to have a large (few square meters) leak. Of course, if the reactor containment holds, there is no hazard to the public. Even with a sizeable fission product release, enough non-volatile materials, such as strontium and the rare earths, would be left to keep the fuel molten; with no core cooling, melting through the bottom of the reactor appears likely if the power level is as high as 100-1000 MW(e). If fuel is beyond a few months in age, the fission product content doesn't change very much, and fuel a few years old was assumed. Fission products from such large power reactors was shown by the model to spread widely in an accident. A Gaussian probability distribution of the fission products in the plume and a uniform population was assumed to vary exponentially radially. With a two mile per hour wind, fatalities might extend to thirty miles.

Dr. Beck considered the BNL accident analysis techniques as needing much more editing before publication is possible. He doesn't want the conclusions written down. A paper on the meteorology aspects of the codes is now ready for a journal; these have already been used for a Savannah River plant analysis. The staff time of the BNL group is limited, and the AEC has not asked for a concluding report. The BNL group does not wish the chore of preparing a final report; the effort appears to be a large one and is not considered as fruitful as other

BNL projects, e.g., building reactors. Only two BNL individuals have had much contact with this accident study.

Dr. Hanauer and Dr. Okrent expressed a desire for conclusions, but Dr. Beck would promise only to have a methods report prepared. Dr. Beck said that this accident study information is being transmitted to the Joint Committee on Atomic Energy (JCAE); the Regulatory Staff will further explore the results of the study.

This appears to be the first unequivocal statement by the Regulatory Staff to the effect that containment failure was inexorably associated with core melt in large LWR's, although possible allusions to this important conclusion are to be found in the minutes of the May 4 Subcommittee meeting.

2.9 THE CHINA SYNDROME - Part 1

It is clear from the records that at its 74th meeting, June 8-11, 1966, the ACRS first arrived at the conclusion that full scale core melting must be correlated with a loss of containment integrity; also, that the reactor as proposed for Dresden 3 was not acceptable, and that the same would apply to Indian Point 2. Just when the fact that containment failure would be a consequence of the full meltdown of the core of a large light water reactor became evident to various individuals or groups is not so clear. In 1963, when large reactors such as the Ravenswood reactor, the San Onofre reactor, the Connecticut Yankee reactor, and the possible reactor in the County of Los Angeles were all being proposed, there was no mention in any of the available review material for these reactors of the possible connection between full scale core meltdown and a loss of containment integrity. This was despite the fact that the generalized accident (or maximum credible accident), which served as the basis for evaluating the acceptability of containment design and of engineered safety features intended to limit the release of fission products, assumed full scale core melt and the release of the bulk of the volatile and gaseous fission products to the containment building.

In 1963, the LOFT experimental program was initiated by the AEC. In this safety research program it was proposed to build a 50 MWt reactor at the National Reactor Testing Station in Idaho, run it at power for a period of time sufficient to build up a sizeable fission product inventory, and then deliberately incur a loss of coolant accident which would lead to full scale melting of the core, in order to provide a large scale experimental basis for describing the course of migration of fission products from the fuel to the containment building, and from the containment building out into the environment.

At its November, 1963 meeting, the ACRS wrote a letter concerning the AEC safety research program in which it said:

> The Committee believes it is of primary importance to determine to what extent engineered safeguards can be relied on in relaxing reactor site restrictions. In the light of present knowledge, it seems unlikely that general principles will render incredible the possibility that high power nuclear reactors can have large power excursions, or that they can have substantial core meltdown. Therefore, it must be expected that the safety analysis for locating and designing nuclear reactors will continue to assume such accidents to be possible, even if only remotely so.

The letter went on to rule out accidents resembling nuclear weapons, and to emphasize that primary attention will have to be given to potential release of fission products to the environment, e.g., from the fuel to the reactor vessel to the containment, and finally release to the environment.

In this letter the ACRS supported a program on fission product release and transport, but gave only lukewarm support to the large scale test in LOFT since it was "not expected to contribute significantly to basic understanding of the phenomena mentioned above."

In any event, this letter supports a supposition that in November, 1963 the ACRS did not correlate large scale core melt directly with a loss in containment integrity for the LWR's then under review.

Similarly, the minutes of the construction permit review for the San Onofre 1 reactor in 1963 discuss the applicant's proposal to assume an assumption that only 6% of the core melts, thereby reducing the assumed fission product release to the containment, and enabling Part 100 to be met at the site boundary with the assumed containment leak rate. The assumption of 6% core melt was not made because large scale melting of the core would, in fact, lead to a loss of containment integrity, nor was the actual course of events associated with 6% core melt examined at all.

With the advent of the use of zirconium cladding, instead of stainless steel cladding, concern arose for possible zirconium-water reactions in light water reactors as a consequence of the postulated core meltdown. At that time it seemed plausible that differences in containment design might be required, because of the probability of a more complete chemical reaction and much more heat being associated with the zirconium-water reaction than with the stainless steel-water reaction.

The AEC Regulatory Staff held a symposium (AEC, 1965) on possible zirconium-water reactions in water reactors, chaired by Clifford Beck, Deputy Director of Regulation, in Germantown, Maryland on April 29, 1965. In his introductory remarks at the symposium, Dr. Beck pointed out that, as part of the licensing process, the Regulatory Staff had analyzed and evaluated the safety aspects of water reactors containing substantial amounts of zirconium in the core. But now the Staff was faced with the possibility of having to review a large number of water reactors whose fuel would be clad with zirconium. This would increase the possibility of a zirconium-water reaction with, as its consequence, a substantial release of energy and hydrogen, which could burn or explode. The Regulatory Staff had to decide if safeguard systems must be designed for the above situation. The first speaker, J. Waage, described the LOFT program in which a core was to be deliberately melted, and discussed how the use of zirconium cladding instead of stainless steel cladding might affect the course of the experiment. The second speaker, L. Baker, discussed various experiments including several intended to measure the rate at which zircaloy would interact with a steam environment at temperatures up to and including the melting temperature. The speaker for the Westinghouse Atomic Power Division, R. Wiesemann, concluded that, if engineering safeguard systems worked properly, there would be essentially no Zircaloy-water chemical reaction. He went on to discuss the effects of large scale zirconium-steam reactions, the resultant hydrogen generation, and the potential effects on pressure in the containment. He arrived at the conclusion that containment integrity could be maintained in the face of substantial zirconium water reactions, without alluding to the difficulties that would be associated with any situation wherein the temperatures of the core were so high that such large scale Zircaloy-water reaction occurred, namely, that it would then be very difficult to assume that the core did not become molten. L. Epstein of General Electric gave a talk in which he discussed the possible course of events if one assumed a loss of coolant accident with no emergency coolant, and concluded that the range of metal-water reaction would be limited to about 15 to 20 percent of the total Zircaloy in the core. This analysis assumed that the Zircaloy is quenched when it falls into cool water below the core melting and subsequent effects including more metal-water reaction and containment failure.

R. L. Doan, then Director of the AEC Division of Reactor Licensing, commented on the problems faced by the Regulatory Staff with regard to metal-water reactions. The magnitude of the reaction depended upon where the break was. There was a great deal of variability in what could happen, and he asked the audience what was a reasonable amount of metal-water reaction to assume for design of containment. He also asked, was there a basis for accepting such a figure, and could the figure be determined by analysis or should it be assumed. Doan expressed the hope that the metal-water reaction experts at the symposium would resolve these problems and questions, and advise exactly what conditions containment should be designed to withstand. Again, implicit in this discussion is the assumption that the core will grossly overheat and yet containment integrity will be maintained.

Dr. Beck asked the representatives of Westinghouse and General Electric if they had considered the situation wherein a loss of coolant accident occurred, the coolant left the core, no metal-water reaction occurred, and then the core spray system functioned late, supplying water to a core which was not very hot from decay heat. Beck wondered if this wasn't a more dangerous situation than if water had not been added by the core spray system at all. The reactor operator had the dilemma of letting the core melt by decay heat, or due to adding the water and getting a metal-water reaction. Levy of General Electric responded that the answer was to design a good core cooling system; the best way to prevent the metal-water reaction was to keep the fuel rods cool. This meant that one had to design an injection system that did the job in all instances. The Wetinghouse representative, Wiesemann, agreed about design to prevent the metal-water reaction; however, he stated that the containment design had to be such that even if safeguards failed, the metal-water reaction could be handled by the containment system.

For several of the reactors that were reviewed in the period of time before, around, or after the April 29, 1965 meeting on metal-water reactions, the Regulatory Staff used a figure of 25% of the clad as being involved in metal-water reactions, and assumed that the containment had to be designed to accept the energy generated from the reaction itself and the combustion of the hydrogen (as it was formed). In a report by the so-called Study Group on Metal-Water Reactors in Nuclear Reactors, (Phase 1 report, 1966), the summary and conclusions include the following: "In the case of large power reactors, i.e., of the order of 1,000 MWt, suffcient fission product decay heat will be available to not only melt a large percentage of the active core but also to melt through a pressure vessel. Such conditions are possible whether a metal-water reaction occurs or not, provided adequate heat removal systems are not available in the case of a major nuclear reactor accident to remove decay heat. (No mention is made of the possible continuing downward motion of the molten fuel through the foundation of the containment and the loss of containment integrity due to this cause).

In another conclusion, the Study Group stated that "in the case of zirconium clad cores, the safeguard system to remove heat from the core may be effective in limiting the metal-water reaction initiated at the start of the accident or within 5 to 10 minutes later." And in another conclusion the Study Group states that "if one assumes the safeguards systems do not function in such a manner as to prevent or limit the possible metal-water reactions, such reactions could encompass most of the metal available in the core and also produce adequate quantities of hydrogen to result in an explosive hydrogen-air mixture in the containment or confinement system." This last conclusion suggests a possible containment failure mode due to large scale overheating of the core; however, it is not clear if the Regulatory Staff ever evaluated this in their licensing review of PWR's at that time. There appeared to be no consideration of a corresponding change in the approach to containment design. At least, for the PWR's reviewed for construction permits in the few months following February 1, 1966, namely the Brookwood reactor and Indian Point 2, the Regulatory Staff did not assume 100% metal-water reaction. Nor did the ACRS.

Another source of insight into the timing of the development of thinking concerning the relationship between full scale melting of the core of a large light water reactor and a correlated loss of containment can be obtained from a review of the records of the work performed by Brookhaven National Laboratories on a possible revision of WASH-740 (1957). This work was initiated by the Atomic Energy Commission at Brookhaven National Laboratory in 1964 and was performed under the guidance of a Steering Committee chaired by Dr. Beck of the AEC, and which included in its members, W. D. Claus, R. L. Doan, A. P. Kenneke, W. J. McCool, J. McLaughlin, U. M. Staebler and F. Western, all of the AEC, and F. Gifford of the Weather Bureau and D. Okrent of Argonne National Laboratory. WE shall not herein try to examine the details of how decisions were taken on what to report concerning the work performed by Brookhaven National Laboratory, nor of the controversy which has arisen concerning whether or not the Atomic Energy Commission should have released more information in 1965 when it chose to publish only a very brief summary of the result of this work. Rather, we shall look at selected pages in some of the minutes and reports prepared in connection with this work for insight into the technical knowledge which was available or at least was portrayed in writing.

On January 22, 1965, Dr. Beck wrote to the members of the Atomic Industrial Forum Safety Committee and sent them a working draft of Chapters 1 and 2 of the proposed re-examination of WASH-740 (Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants). We first quote from Page 8 of draft Chapter 2, "Loss of Coolant with Containment."

> The Emergency Core Cooling System cannot be made foolproof. It must be turned on and must have an adequate water supply in order to operate effectively. Thus, if one of the major coolant pipes fails and the emergency core cooling system also does not perform adequately, then the fuel element temperature would rise, the elements would melt and the fission products would be released from the fuel matrix. An aerosol of fission products could be swept out of the vessel and into the containment shell by convection currents.
>
> If the containment is effective, that is, if the leakage rate is less than or equal to the design leak rate, then the fission product aerosol will deposit within the containment shell at the rate of 50% per day and will leak out of the system at a very low rate. The design criteria for maximum leak rate from the containment system is that the leakage of fission products resulting from an accident of this nature will not subject anyone beyond the reactor site boundary to more than 25 R to the thyroid.* Personnel exposure levels in this region produce essentially no damage. Thus, if the containment system is effective, a loss of coolant accident in which the emergency core cooling system also fails, would result in essentially no damage to the public.

This statement can be interpreted to mean either that the writer expected it was possible that one could melt the core of a large reactor and maintain the containment system effective, or that they were treating this as a hypothetical case. However, there is no suggestion in the report that the writers at this time expected a correlation between melting of a large core and loss of containment integrity. Again on Page 13 of the same draft, Chapter 2, prepared by Brookhaven and forwarded to the Atomic Industrial Forum by Dr. Beck, we have,

> If there were an <u>unimpeded</u> path for convection of the air in and out of the vessel, then most of the fission product aerosol would be dragged out of the vessel. Calculations have shown that the

*This seems o be an error; 300 R was probably intended.

> particle size of the aerosol under such conditions would probably not exceed 1 micron. At the end of 4 hours the molten fuel will have melted its way through the bottom of the pressure vessel and quickly will have gotten to a concrete floor. Fission product afterheat contained within the molten fuel would spall the concrete until such time that a large enough area for conduction of heat to the ground has formed and the fuel solidifies, thus terminating any further fission product release. The fission product aerosol which was dragged out of the pressure vessel by convection currents would enter the containment shell and would begin to deposit in the shell at the rate of 50% per day. Since the majority of surfaces available for deposition are painted surfaces, no one fission product group would be preferentially deposited within the shell.

Here we see mention of attack of the concrete floor of the containment by the molten fuel; however, there is no direct implication that the loss of containment is expected. And on Page 14 of the same draft report we find the following statement.

> <u>Release from Containment</u>
>
> At this time we would normally expect the containment shell to be intact; and the containment spray and/or filter system to be effective and to trap most, if not all, of the fission product aerosol.
>
> The containment shell itself is the last of the present day safeguards. If the containment system does work, that is, its leakage rate is as designated in the hazard summary report, then, as has been shown, very small damage to the public will ensue. There remains, however, the small but finite probability that the containment will be breached by an open door or other mechanism. Thus, we must assume that the containment is not complete and depend only on the natural deposition mechanisms for depletion of fission products from the air. An opening the size of a door will have an exhaust time due to wind action which is short compared to the fission product deposition time. Under these conditions most of the fission products would be released to the atmosphere.

Again there is no direct suggestion that the melting of the core would lead to a breach of containment.

Another example can be derived from a letter from Dr. Beck to members of the Steering Committee, May 14, 1965, which forwarded some possible drafts of a report on the new Brookhaven work. Included for consideration was a memorandum by K. Downes and A. Court of BNL entitled "Theoretical Consequences of Hypothetical Accidents in Large Nuclear Power Plants," dated May 5, 1965. On Page 3 of this memorandum it says:

> However, to achieve the purpose of the present study, which involves identifying the point at which damages and public injury would occur, it is necessary to suppose that all the means of assuring safety have failed to function. For instance, the emergency cooling system is simply supposed not to operate as it should. In addition it must be assumed that some large penetration in the reactor containment building is open at the time of the accident or that the containment building is damaged by a missile from the accident, so that containment is violated. At this point, the hypothetical accident would become a hazard, and its consequences would be severe. In order to identify the point in the spectrum of hypothetical reactor accidents where public injury and financial damage would begin, it has been necessary to assume that a very improbable event is followed by a failure of a complete set of safeguards that are engineered to prevent hazards to the public or to reduce these hazards.

Another bit of evidence on the thinking at the time comes from a packet sent by Dr. Beck to the members of the Steering Committee on April 21, 1965. The packet included a possible draft letter to Representative Holifield, Chairman of the Joint Committee on Atomic Energy, which was to review the results of Brookhaven's re-evaluation of WASH-740. On Page 3 of the draft letter it states:

> The preliminary results of Brookhaven's re-evaluation can be summarized by noting that in the first two cases postulated, i.e., where the emergency cooling system and/or the containment system function as designed and tested, a loss of coolant accident, irrespective of the degree of fuel melting, will not result in substantial injury to the public or damage to off-site property. It is only in the highly improbable instance where these and all other engineered safeguards failed simultaneously that a loss of coolant accident could result in a public hazard.

It seems fair to assume that in the spring of 1965, if there were groups or individuals who made a direct connection between large scale core melting and a loss of containment integrity, it was not a widely held piece of knowledge; it was not a part of the thinking process undergone in the re-review of WASH-740; and it was not a part of the information presented to the Atomic Energy Commissioners or to the Joint Committee on Atomic Energy in 1965.

The minutes of a meeting held November 2, 1965 between the Regulatory Staff and Consolidated Edison concerning Indian Point 2, reaffirm that containment was considered an independent safeguard at that time. The excerpt which follows quotes Dr. Doan, Director of the AEC, Division of Reactor Licensing, and Mr. R.A. Wiesemann of Westinghouse.

> Dr. Doan indicated that DRL and ACRS would look at the reactor system itself to see that meltdown is prevented for all including the biggest rupture. Then, the containment is put around it in order that there be protection even if the others fail, and metal-water reactions occur.
>
> Dr. Doan indicated that it was no more credible that safeguards in the containment work than those in the primary system, but he believed that the containment had to be designed to contain something.
>
> Mr. Wiesemann stated that this had been done; that the containment was satisfactory if the safety injection system does not work at all. A low head pump will be available with high reliability, but this is not being depended on for containment integrity.

In May and June 1966, however, we find statements to the effect that core melt in a large light water reactor would indeed lead to a loss of containment integrity. In the mintues of the 73rd ACRS meeting May 5-7, 1966, the section on Indian Point 2 says, "Melt through of the containment vessel by the molten core might be at 4500 seconds after a loss of coolant accident." There is no further discussion of this point, and the possibility exists that it was the pressure vessel which was really referred to.

Also, in the minutes of the 73rd meeting, Mr. Downes of Brookhaven is quoted as pointing out that, "if engineered safeguards don't work, the containment would rupture." The mintues go on:

> "BNL considered a 1000 MWe plant and followed the decay heat, which remains near 30 MW for a long period of time. The molten UO_2 core would eat its way right through the pressure vessel. This was not so for 100 MW(e) reactor, but there is a crossover near 200 MWe."

The minutes do not indicate any extensive discussion in May, either with the Indian Point group or with the Brookhaven group, concerning the possible failure of containment due to core melt. However, such discussion may have occurred.

The minutes of the June 3, 1966 meeting of the ACRS Subcommittee on Reactor Design and Operating Criteria, which are completely duplicated at the end of the previous section, unequivocally relate core melt to containment failure as part of the description given by Dr. Beck and Dr. Wensch of the recent BNL work related to re-examination of the consequences of a serious reactor accident.

So, sometime between the spring of 1965 and June of 1966, both Brookhaven and Dr. Beck had reached the conclusion that for the large water reactors, full scale core melt would be associated with the loss of containment integrity, at least by melting through the bottom of the containment. Nevertheless, there had been no change in the Regulatory Staff approach to the acceptance of reactors based on this knowledge, nor was it mentioned as part of the safety evaluation issued by the Regulatory Staff for any of the reactors reviewed during that time period, namely Dresden 2, Brookwood, Millstone Point, Indian Point 2, and Dresden 3. However, Mr. Price and Dr. Beck did take a rather negative attitude toward the possibility of reactors being constructed at metropolitan sites during this time period. (Indian Point was not categorized as such a site by the Staff).

The minutes of the 73rd ACRS meeting, May 5-7, 1966, also indicate a considerable reluctance on the part of Dr. Beck to make available to the ACRS detailed results from the Brookhaven work, and it was only under some considerable pressure that the presentation was made at the Subcommittee meeting held on June 3, 1966. It's not clear from the minutes why Dr. Beck was reluctant to have the results of the Brookhaven study made available to the ACRS. The overall consequence of a large release of radioactivity, as obtained by BNL, were not dissimilar from results that the ACRS was receiving from its own studies and from other sources. However, some of the detailed information was different, particularly the estimate that the dose to the gastro-intestinal tract was lethal when the whole body dose was only 100 rem.

The question of core melt and its adverse effect on containment integrity finally came to a head as part of the licensing process at the 74th ACRS meeting, June 8-11, 1966. The mintues show that there was considerable discussion among the ACRS members concerning the possible ultimate fate of this large amount of molten fuel. There were varied opinions ranging from the possibility that there was insufficient information available to portend a serious safety problem to the point of view that fuel melting was an unanalyzed safety probem (which required resolution before proceeding).

According to the summary of this meeting prepared on May 27, 1968 by M. Libarkin,

> Several members felt that an important question had been identified regarding the melt-through accident. Others thought that a general letter might be appropriate, since the problem was applicable to many plants. Several thought it would not be appropriate to raise the question on Dresden 3 in view of the Committee's action in accepting Dresden 2.

The matter was discussed in detail with Commonwealth Edison, who were applying for a construction permit for the Dresden 3 reactor, and they were advised of the ACRS concern. At the request of Commonwealth Edison, the ACRS met with them again on Saturday (which was not usual); at this session General Electric presented a preliminary analysis of what might happen in the core melt situation for a boiling water reactor. The minutes indicate that Commonwealth Edison and General Electric were suggesting that possibly the fuel might be retained in the concrete base of the containment, if there were sufficient water above the molten fuel to remove a large amount of the heat from the fuel by radiation. At the conclusion of the 74th meeting, the ACRS decided that more information was needed on the subject of core melt as part of the Dresden 3 and Indian Point reviews.

When we next review what transpired during the summer of 1966, beginning in June and ending in October, we shall see that during the first two months there was intensive examination by the ACRS, and by Westinghouse and General Electric, of the possibility of providing engineered safeguards which would maintain containment integrity in the presence of large scale core melt for the large reactors being considered. We shall see that, of its own, the ACRS arrived at the conclusion that it was very difficult at that time to provide a solution for coping with core melt for the Indian Point 2 PWR which had a large dry containment; and that it was relatively impossible, or nearly so, for the Dresden 3 reactor design with its smaller, pressure-suppression-type containment. We shall see that the Dresden 3 applicant and its reactor vendor, General Electric, presented information in support of the thesis that maintaining containment integrity in the face of core meltdown was not feasible for their design.

They did not propose to try to design to cope with core melt; they also believed that their existing emergency core cooling system was adequate to prevent core melt in the face of a LOCA. We shall see that Cnsolidated Edison and their reactor vendor, Westinghouse, proposed that they could supply a structure below the reactor vessel which should be able to hold the molten core and keep containment integrity intact. We shall see that the Regulatory Staff took the position that each reactor design, as proposed prior to the June ACRS meeting, was acceptable, although they acknowledged that some further study was warranted on both the question of emergency cooling systems and on problems associated with core melt. We shall see that a wide range of opinions existed among the members of the ACRS, which eventually arrived at a consensus that they could write letters favorable to the construction of the Indian Point 2 reactor and the Dresden 3 reactor on the basis of greatly improved emergency core cooling systems and much greater emphasis on primary system integrity to reduce the probability of a LOCA. We shall see that the pressures on the ACRS were indeed very great, and that in the midst of this extremely complex discussion, evaluation and review, the AEC publicly noticed the beginning of hearings on the Indian Point 2 reactor by the Atomic Safety and Licensing Board, despite the fact that the ACRS had previously requested that the AEC refrain from doing this before the Committee had completed its review.

We shall see that the ACRS decision on these two reactors also included the writing of a general letter in which the ACRS was to make strong recommendations concerning the rapid development and future implementation of further engineered safety features to cope with problems associated with core melt; and that this was the basis by which several members agreed to the issuance of letters favorable to the construction of Indian Point 2 and Dresden 3. Such a letter was prepared by the Committee and submitted for comment to the Regulatory Staff who submitted it to the Commissioners themselves. We shall see that, at the September, 1966 meeting of the ACRS, the Commissioners urged that the Committee, rather than send such a letter, await the report of a Task Force that the Commission would establish to study and quickly report on problems associated with core melt. And we shall see that the majority Committee opinion was to go along with this proposal by the Commissioners regarding the recommendation for safeguards to deal with core melt. And we shall see that the ACRS did write a safety research letter in October, 1966, recommending that the safety research program of the Atomic Energy Commission strongly emphasize problems associated with phenomena related to large scale core melting as well as to improvements in ECCS (but did not recommend that new safeguards be developed for possible implementation in, say, two years).

Under a separate heading we will later discuss the report of the Ergen Task Force (or Task Force on Emergency Core Cooling), ACRS reaction to the report, and the action, or lack thereof, by the AEC and the industry on the general problem of core meltdown in the ensuing months and years.

It must be recognized that during this same period, there were many other reactors being reviewed by the Regulatory Staff and by the ACRS, and that for Indian Point 2 and Dresden 3, there were many other technical questions under review in addition to core melt. For example, the acceptability of a positive moderator coefficient and its possible consequences on the postulated reactivity accident that could result from ejection of a control rod was discussed for Indian Point 2. Questions were raised concerning the adequacy of fire protection at Indian Point 2, and the adequacy of the reliability of various systems like the emergency power suplies, etc.

Looking back a decade later, one may well ask, "Why was the review procedure pushed at so rapid a pace with so serious a question involved?" And "Why was resolution accepted based on partial information and on general criteria?" It's clear that there was considerable pressure from the industry not to impose further delays on beginning construction of these plants, and that in the particular time period, 1965-1966, the AEC Regulatory Staff was very sensitive to the question of delays arising from the regulatory process. Curiously, this was the period during which the time between application for a construction permit and issuance of a construction permit was perhaps the shortest it has ever been.

Also, looking back with the hindsight of another 10 years or so, it is clear that the loss of coolant accident was uppermost in the minds of the ACRS and the Regulatory Staff as the most probable source of core meltdown. They were not ignoring other accident sources, and as time passed, because of the clear relation between core melt and the loss of containment integrity, all possible sources of core melt began to be searched out, to be evaluated, and to be modified, as possible, to reduce the probability that any particular source would be an important contributor. Nevertheless, the emphasis during that period was on the loss of coolant accident, the adequacy of the emergency core cooling system, and on means to reduce very much the probability of LOCA.

Now we return to a relatively detailed history of events following the June, 1966 ACRS meeting. One June 14, 1966 the Regulatory Staff contacted the ACRS office to ascertain if the Committee desired that written information be submitted by the applicants for Dresden 3 and Indian Point 2 regarding the course of a core meltdown accident and the reliability of emergency cooling systems. The Regulatory Staff then advised the applicant for each reactor that neither the ACRS nor the Regulatory Staff was requesting written information in regard to the above items. Commonwealth Edison indicated that no written information would be submitted regarding Dresden 3. Consolidated Edison stated that they might provide a limited amount of information in their third supplement, which was in the process of preparation for submission. An Indian Point 2 Subcommittee meeting was held on June 7, 1966. The mintues of the Subcommittee meeting note that an ACRS member called attention to the last paragraph of the 3rd supplement to the Preliminary Safety Analysis Report which had recently been filed by the applicant. This paragraph states that the cavity below the Indian Point 2 reactor pressure vessel will have the capability of preventing breaching of the containment by the molten core through the use of the water in the cavity.

The minutes of the Subcommittee meeting also show that the question of whether reactivity transients resulting from the postulated ejection of control rods had been adequately treated and whether the positive moderator coefficient proposed for the Indian Point 2 reactor during part of its lifetime would lead to an unacceptable effect for reactivity transients. (With hindsight, one sees that it is fortunate that this positive moderator coefficient was designed away by Westinghouse, for reasons other than the reactivity transient. When the matter of anticipated transients without scram came up some years later, Westinghouse's detailed analysis showed that it needed a negative moderator coefficient in order to calculate tolerable consequences).

The major discussion during the meeting related to consequences of pressure vessel failure and to the question of core melt. Mr. Boyd of the Regulatory Staff stated that the Staff presently believed that the proposed Indian Point 2 plant is an acceptable one. They considered Indian Point 2 "to be a suburban reactor." The Staff indicated they did not believe there was a significant difference between Indian Point 2 and Dresden 3 regarding the consequences of a core meltdown accident and the ultimate fate of the molten fuel. Mr. Boyd said he recognized the inconsistency of the Staff's present position that the core will melt, leading to fission product release and metal-water reaction considerations, but that core melt will not cause a problem from a loss of containment integrity.

Westinghouse made a presentation of the heat transfer calculations they had made concerning the ability of the reactor vessel itself to hold molten fuel; they had concluded that rather large fractions of core (as much as 60%) could be held in the vessel itself. They went on to present the heat transfer analysis of the refractory lined, water-cooled, stainless steel core catcher device which they proposed to place below the vessel, in case the core melted through the vessel. Westinghouse indicated they believed that, if the core melted through the pressure vessel, when the molten fuel hit the water located below, there would be rapid chilling and formation of solidified uranium oxide.

The Indian Point 2 Subcommittee emphasized the need to provide adequate information to the full ACRS regarding the core melt-through accident.

The Westinghouse presentation on their core catcher device did not include considerations of the possible generation of large amounts of hydrogen and its effect on containment integrity, the possibility of a steam explosion, or several other phenomena relevant to the reliability or the effectiveness of such a system.

The Westinghouse presentation on the ability of the containment to withstand several modes of gross pressure vessel failure, including longitudinal splitting or circumferential rupture of the vessel below the flange, was relatively optimistic concerning ability to withstand such failures. Westinghouse also classified any such failure modes as being of extremely low probability.

A Subcommittee meeting on Dresden 3 was held July 7, 1966. Prior to this meeting, on June 25, 1966, ACRS member Etherington provided a memorandum to other ACRS members giving the results of a very quick and crude analysis he had performed on core melt. The insight shown and the long-term validity of the general conclusions are a tribute to his ability. The memo is on the following pages.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

June 25, 1966

To : ACRS Members

R. H. Wilcox for

From : R. F. Fraley, Executive Secretary
Advisory Committee on Reactor Safeguards

Subject: BACKGROUND INFORMATION ON DRESDEN 3
PROVIDED BY H. ETHERINGTON

The attached comments have been provided by Mr. Etherington as background material for consideration in connection with the Dresden 3 emergency core cooling questions raised at the 74th ACRS Meeting.

The applicant will be prepared to discuss the information that has been developed at the ACRS Subcommittee meeting on July 7, 1966.

<u>Attached</u>:

Background information by H. Etherington on Consequences of Meltdown, dated 6/22/66.

<u>Copy</u>:

Dr. Dick Duffey

CONSEQUENCES OF COMPLETE MELTDOWN

The following ball-park numbers and conclusions may serve as background for appraisal of the expected analysis of the meltdown accident by the applicant.

SUMMARY

The molten fuel will penetrate the bottom of the building and will form a liquid pool that will grow for many months by fusion of concrete and earth. The "Simplified Spherical Model" gives liquid-pool diameters of 32 ft., 54 ft., 80 ft. after one day, one week, and one month, respectively. Loss of building containment will probably occur both by collapse of parts of the building into the pool and by escape through the hole melted in the containment liner. The liquid pool may not contribute greatly to the immediate airborne activity, but the mass would remain as a long-term problem.

Table 1. Decay Heat, Fraction of Operating Power (Untermyer-Weills Formula)

Operating Time	Time After Shutdown: 1 sec.	1 min.	1 hr.	1 day	1 week	1 mo.	1 yr.	5 yr.
1 year	0.0582	0.0390	0.0157	0.0066	0.0033	0.0016	0.00018	
2 years	0.0584	0.0393	0.0159	0.0068	0.0035	0.0018	0.00026	0.00003
Infinite	0.0589	0.0397	0.0164	0.0073	0.0040	0.0023	0.00067	0.00035

Table 2. Decay Heat for Two-Year Exposure at 2600 MWt (8.87×10^9 Btu/hr)

Time After Shutdown	1 sec.	1 hr.	1 day	1 week	1 mo.	1 yr.
Heat rate, 10^6 Btu/hr.	519	141	60	31	16	2.3

Table 3. Thermal Properties of Materials

Sensible heat of slag at 3000 F 693(earthy constituents at 2600F) + 180 latent heat + 120 superheat = 993 (say 1000) Btu/lb (heats of decomposition assumed to equal heat of formation of slag)

Mean specific heat of earthy mixtures (70 to 2600 F): c=0.275 Btu/lb
Mean conductivity of compacted earth (70 to 2600 F): k=1.5 Btu/ft.-hr.°F (conservatively high)

Table 3 continued

Density of compacted earth: 96 lb./ft.^{3}
Mean diffusivity: $1.5/(96 \times 0.275) = 0.057$ ft.^{2}/hr.

SIMPLIFIED SPHERICAL MODEL

The Model. A spherical homogeneous molten source at 3,000 F is surrounded by earth. The source grows by fusion of surrounding material and incorporation of the melted material into the source.

Summary of Conclusions. The molten source grows, rapidly at first and then more slowly (Lines 1 and 4 of Table 4) -- a diameter of 54 ft. is reached in one week. Growth continues for many months until conduction into the earth stops further melting and the sphere starts to freeze very slowly as the decay heat diminishes.

Source Growth Assuming No Heat Removal by Conduction. Assume infinite exposure in the reactor, and adapt the Way-Wigner formula:

Decay fraction $= 0.0622\ t^{-0.2}$ (t in seconds)
$= 0.0121\ t^{-0.2}$ (t in hours)
Rate of decay heat generation:

$$= 0.0121 \times 8.87 \times 10^9\ t^{-0.2} = 107 \times 10^6\ t^{-0.2} \text{ Btu/hr.}$$

$$107 \times 10^6\ t^{-0.2} = 4\pi r^2 \frac{dr}{dt} S \rho$$

where r = radius of melted source

S = sensible heat of fused earth at 3,000 F = 1,000 Btu/lb.

ρ = density of packed earth = 96 lb./ft.^{3}

Integrating: $r^3 = 334\ t^{0.8} + C$

The constant C should be determined from analysis of the meltdown and melt-through. This analysis has been evaded by assuming a point source at shutdown, i.e., C = 0. After the first day or so the starting condition probably becomes unimportant.

$$r = 6.94\ t^{0.267}$$

Table 4 gives the radius r and the rate of growth dr/dt as a function of time (Lines 2 and 3).

Heat Conduction Into the Ground, Steady State Condition.

$$8' = -k\ 4\pi p^2 \frac{d\theta}{dp}$$

where: p = radius of any shell in unmelted ground, ft.
= total heat flow at radius p, Btu/hr.

θ = temperature at radius p, ^{o}F
k = conductivity, 1.5 Btu/ft.-hr.-^{o}F

Boundary conditions: At $p = \infty$, $\theta = 0$; at $p = r$, $\theta = 3{,}000$ F

$$q' = 0.0565 \times 10^6 \ r \ \text{Btu/hr.}$$

As shown on Line 4 of Table 4, heat conduction is negligible when the source is small, but eventually becomes equal to the heat source and prevents further melting.

Heat Stored In Ground. Melting is retarded by heat absorbed in raising the temperature of the ground. For a sphere at 3,000 F suddenly buried in cold packed earth, the average heat penetration for the first six hours has been calculated by the Schmidt graphical method to be 7,500 Btu/ft^2 hr. The average heat absorption q" for the source over the first six hours is given in Line 5 of Table 4.

These numbers are not to be compared directly with the heat source strength, because:

(1) new sources are not repeatedly inserted,

(2) the rate of heat penetration decreases asymptotically with time to the steady state conduction rate, and

(3) part of the heat is recovered in melting. The numbers are, however, large enough to suggest that the source radius will grow more slowly than indicated in Line 2.

Table 4. The Spherical Model

Line	Time	12 hr.	1d	2d	1 wk.	1 mo.	3 mo.	1 yr.	
1	Rate of heat generation,	65	53	49	38	29	23	17	x 10^6 Btu/hr.
2	Sphere radius, r	13.5	16	19.5	27	40	56	78	ft.
3	Growth rate, dr/dt	.30	.18	.11	.043	.015	.006	.0024	ft./hr.
4	Rate of heat conduction,	.76	.92	1.1	1.5	2.3	3.2	4.4	x 10^6 Btu/hr
5	Rate of heate storage,	17	25	36	70	153	294	577	x 10^6 Btu/hr

Note: (1) The Way-Wigner formula gives much higher values (Line 1) than the Untermyer-Weills formula for times over one week.

(2) 1 month = 30 days.

RADIATION MODEL

The Model. The fuel melts down into the 20 ft. diameter well under the pressure vessel. The fuel is retained in the well and all of the decay heat is radiated to the water cooled cavity above. For 724 assemblies, each containing 492.7 lb., the pool depth is 1.82 ft.

Summary of Conclusions. From the analysis and considerations under the heading "Probable Behavior", it is concluded that radiation cannot be relied on as a dominant factor in heat removal.

Radiating Capacity of Surface. Table 5 and Line 1 of Table 4 show that, if the surface remained clear, the pool could, with a reasonable combination of emissivity and geometry, radiate all the decay heat at a temperature around 4,000 F. The model is, however, shown under the heading "Probable Condition" to be unrealistic in other respects and should be discarded as a mechanism for complete heat removal.

Table 5. Radiation From Pool Surface

Temperature	3000 F	4000 F	5000 F
Black-body radiation per sq. ft.	0.245	0.66	1.52 x 10^6 Btu/ft^2-hr.
Black-body radiation from 20 ft. diam. pool	77	207	478 x 10^6 Btu/hr.

Condition of Pool. The high rate of heat transfer requires that the surface remain clear and free from solid surface crusts of substantial thickness. For example, at the 12 hr. rate of heat release of 65 x 10^6 Btu/hr. (Table 4), a surface temperature of 4000 F, a source melting point of 4500 F, and a crust conductivity of 10 Btu/ft.-hr.$^\circ$F, the crust thickness is given by

$$\Delta = k \Delta t/(q/A) = 0.025 \text{ ft.}$$

The pool must therefore remain molten, except that a very thin surface skin will form when the melting point is above 4000 F.

The rapid convection that is characteristic of high-temperature liquid pools will probably be effective in transferring heat from the source interior to the surface. On the other hand, layers of slag or iron on the surface will strongly reduce heat flow by resistance to conduction and interface convection.

PROBABLE BEHAVIOR

The spherical model is modified by geometric asymmetry in the vertical axis and the behavior of a molten material in a non-refractory hearth.

Erosion and Melting of Foundations and Earth. Compacted earth has negligible cohesive strength and the melting point is low, e.g., 2500 F. The condition of a higher temperature liquid resting on a hearth of such material is highly unstable. Not only is the hearth melted progressively by contact with the hot material, but it is also disintegrated by intrusion of the melt into fissures, with consequent detachment of large lumps which float to the surface of the pool. If the hearth material contains moisture or combined water, generation of steam will make the process violent. Concrete and clay-soils both contain combined water (kaolin, $Al_2O_3 \cdot 2SiO_2 \cdot 2H_2O$, contains 14 per cent water by weight).

Character of the Melt. It is assumed that the fused uranium oxide will combine with the melted earthy materials to form a slag of progressively lower density and lower melting point -- no attempt has been made to verify this assumption from phase diagrams.

Upper Surface of the Melt. The radiation model shows that, initially, the melt would have to remain at a very high temperature to permit radiation to be an important factor in heat removal. Radiation, therefore, cannot cool the melt sufficiently to inhibit hearth disintegration, and temperatures will adjust until the heat flow is properly distributed between the two heat sinks.

Under probable actual conditions, the role of radiation will be further reduced by unfavorable surface conditions. The surface would contain floating islands of cool material from the bottom and sides of the pool and debris from collapsing superstructures. In the early stages, the heat flow might also be inhibited by a three-layer condition of the pool -- liquid fuel at the bottom, floating iron above, and zirconium oxide and slag on top; the iron would insolate the slag from the heat source until the melt became sufficiently dilute to permit the iron to sink and collect in pockets. At times the pool might be partly bridged over.

Shape of Pool. The shape of the pool will probably be between the extremes of a shallow open pool and a deep pit-like sump. The shallow pool would become an effective heat radiator as the diameter increased, but it would also increase the probability of building collapse.

The deep-pit condition is promoted by gravity effects on soil erosion. The deep-pit condition appears more probable and less damaging -- the condition might be encouraged by providing a deep well under the building.

RELEASE OF FISSION PRODUCTS

The onus is clearly on the applicant, if this accident is considered feasible, to show that containment will be preserved and that there will be no intolerable release of fission products from the melt. Containment may be breached by one or both of two processes -- partial collapse of the building and escape through the foundations or surrounding soil.

Building Collapse. Unless the building can retain its integrity with only the outer walls supported on foundations, settlement and collapse into the pool of interior portions and failure of cooling systems can be expected within a day or two. Release of fission products stored in the primary containment must then be assumed.

Escape Through Foundations. If the pool level were high enough to seal openings created by fusion, it could be argued that containment would be maintained. However, there will surely be destruction of the lining just above the liquid level and the integrity of the containment would, at best, be questionable.

The possibility of sealing by fusion is made more remote by shrinkage of earth as it melts. The assumed density of the compacted earth is 96 lb./ft.^{3}, whereas the density of fused silica and light glasses is 140 lb./ft^3. If the latter density applies to slag, the porosity of the earth will, therefore, cause a 31.5 per cent volume contraction on melting -- more if there is release of combined water or carbon dioxide. A cylindrical pool, having the same volume and radius as the one-week pool in the spherical model, would be 36 ft. deep based on fusion of earth, but (ignoring frothing) the surface would be 11 ft. down from the top of the hole.

Release of Fission Products from the Melt. The hearth will tend to be self-sealing by fusion, but high pressure from gas or steam, or from the primary containment, might open fissures through which material would be forced to the surface. However, airborne fusion products from the melt have presumably already been allowed for in the TID 14844 criteria.

Slow leaching of fission products by ground water, after the mass has frozen, and the feasibility of isolating the mass, remain to be investigated.

Among the other sources of information which helped the ACRS to form opinions concerning the likelihood of being able to deal in a feasible fashion with core meltdown, assuming the current reactor design approach, was a set of brief memoranda prepared by Messrs. Hesson, Ivins, Tevebaugh and Wilson of Argonne National Laboratory. In these memoranda they discussed problems related to steam explosions, hydrogen generation, and the size of a core retention structure that might be needed in order to provide sufficient heat transfer surface that one would have a rather high likelihood of the molten fuel freezing and staying frozen, assuming it fell through a hole in the reactor vessel onto some kind of special retention structure.

At the July 7, 1966 Subcommittee meeting on Dresden 3, General Electric presented a detailed analysis of the course of full scale core-melt; containment failure was the consequence. Various potential design modifications intended to enable the containment to maintain its integrity were discussed. These included a fire brick lining; cooling coils in the concrete; a thick, cooled steel plate; increased containment volume, etc. General Electric felt that none of these could be adequately engineered; they saw the answer in prevention of melting rather than holding the molten material.

In executive session, several ACRS members were pessimistic about the possibility of containing a core melt. Some members apparently felt that the ACRS should proceed with Dresden 3 on the basis that melt-through is not credible. Some felt that improving core cooling was a proper solution, and other members felt that more study might produce workable ideas of containing the molten core. General Electric was asked to give further consideration to ways of containing a melted core even if design changes were involved beyond those being suggested in Dresden 3.

On July 7, 1966, the Regulatory Staff issued Report No. 3 to the ACRS on Indian Point 2. At the beginning of this report the Staff notes that Indian Point 2 has special siting considerations and that the applicant has proposed a containment and engineered safeguard system which the Staff believes to be superior to that provided at facilities in less populated areas. The special features noted by the Staff are as follows:

a) The containment is designed to have negligible leakage under the postulated maximum credible accident conditions.

b) Even though negligible leakage is anticipated, two independent means of iodine removal within the containment have been provided. These are an air filtration system and a containment spray system.

c) The recirculating water system that is required to provide long-term cooling of the core is located inside the containment. This arrangement is provided so that it will not be necessary to pump radioactive liquids outside the containment after the maximum credible accident unless both internal pumps fail.

d) Three diesel generators are proposed for onsite power; two are required for operation of the minimum safeguards required to preclude containment overpressurization and core meltdown after an MCA.

On Page 41 of the Staff report, just before the conclusion, the Staff has a topic called, "Items Requiring Futher Study" which is reproduced below, together with the conclusions.

Items Requiring Further Study

Specifically as a result of our review of this facility and generally because of our continuing review of pressurized water reactors, a number of problem areas have been identified. Many of these have been mentioned in the body of this report. We intend to keep the items listed below in mind as we continue our review of the Indian Point II facility and of pressurized water reactors submitted for subsequent licensing action.

1. Adequacy of diesel generator capacity.
2. Design of seal water system.
3. Consequences and causes of reactor vessel rupture.
4. Melting of core through reactor vessel.
5. Stresses in containment liner.
6. Applicability of RCC ejection point kinetics model.
7. Adequacy of air recirculation system.
    a. Flow rates under accident conditions.
    b. Back flow.
    c. Demister.
    d. Design of filters and charcoal beds.
8. Behavior of hydrogen after an MCA.
    a. Imperfect burning as released.
    b. Subsequent burning potential.
9. Control rod worth limiter.

Conclusion

On the basis of the foregoing safety evaluation of Indian Point Unit No. II, we have concluded that there is reasonable assurance that the facility can be built and operated at the proposed location without undue risk to the health and safety of the public. The Staff believes that the resolution of potential adverse effects of reactivity transients can be deferred to the operating license stage of review. As indicated, the control rod worth and moderator temperature coefficient can be appropiately limited if final design of the first core indicates that such limitations are necessary.

It is not clear how the Regulatory Staff planned to deal with questions such as melting of the core through the reactor vessel, or consequences of reactor vessel rupture, as part of the continuing review of Indian Point 2. Nor is the basis clear in the Staff report for the Staff judgment that a construction permit could be issued for Indian Point 2 as then proposed.

At the 75th meeting, July 13-15, 1966, the ACRS met with both the Dresden 3 and the Indian Point 2 applicants. A brief summary of these two meetings is available from the May 27, 1968 memorandum of Mr. Libarkin.

Indian Point 2

Some members felt that the proposal to retain melted core in a water-cooled refractory-lined structure in the lower part of the pressure vessel cavity was a step in the direction of safety. The Committee, however, did not see this as a guarantee of retention. It was also noted that Westinghouse's view that a molten core could be retained by cooling the vessel was not consistant with G.E.'s view that melt-through was inevitable. The Regulatory Staff was noted as preferring steps to retain the core in the vessel to a means of holding a molten mass underneath. Consolidated Edison proposed three high-head and two low-head pumps outside of the containment shell, and two additional low-head pumps inside. The proposal for a water-cooled refractory basin below the vessel was repeated. Westinghouse pointed out the lack of data in support of this design and added that they were not proposing an experimental program to validate it. Primary reliance was being put on core cooling.

Dresden 3

Several members favored no more action with respect to protection for core melting until consideration of future cases. Action on Dresden 3 should not be delayed because of this problem. The Committee agreed not to ask for provisions similar to the Indian Point 2 core catcher (by a soft vote - 4 in favor of such a request, 5 against, 3 abstentions). Although they were not opposed to the Consolidated Edison proposal, one member suggested that Westinghouse had proposed a core catcher only to get Committee approval. The Regulatory Staff felt that the low probability of core melt-down and a melt-through was so small as to be acceptable. General Electric described the results of studies made and several designs to retain a molten core. They had investigated flooding the containment vessel, installing a steel pan with refractory liner and cooling coils, a water tank with 1 million cubic gas volume, a ceramic pebble bed with the water fluidizing the medium for the molten mass and the air cooling. All of the above would require major structural changes. General Electric had concluded that the then current containment design could not handle the consequences of a melt-down and that all of the concepts proposed would fall short of the engineering needs. The basis of design must therefore be the prevention of core melting by emergency cooling. An ECCS design modification was then proposed by General Electric. Either two core spray systems or a core spray and a high volume flooding system with redundant components would be installed.

Mr. Libarkin noted that the Committee asked for additional oral information on both the cooling and the melt-through accident at the end of the meeting with Dresden 3.

Again to provide insight into the discussion that went on at the July, 1966 75th meeting, long excerpts from the Committee minutes are given on the following pages.*

*It must be recognized that the minutes do not represent a comprehensive (or even an accurate) summary. There were long delays in their preparation at that time and they were considered to be less than satisfactory by the ACRS itself.

75th Meeting
July 13-15, 1966

INDIAN POINT II REACTOR

Executive Session

Dr. Palladino reported for the Subcommittee. Improvements in the isolation valve system are planned. Over pressurization of the primary system might release the pressure vessel head which could impact with the crane; steps to resist a longitudinal failure of the pressure vessel were recommended by the Subcommittee. However, measures needed to resist such failure are uncertain; perhaps locating the vessel so that water would fill the surrounding concrete structure following a longitudinal failure and indirectly cool the core would be useful. Much discussion of the core melting accident was had. The Indian Point group is more relaxed over the consequences of a molten core. Some of the group believed that the core decay heat could be removed by cooling the pressure vessel walls; nevertheless, this view with a concentrated reactor core is inconsistent with the General Electric (GE) assumption that a hole would be melted in the bottom of the pressure vessel of the larger core boiling water system.

Dr. Ergen considered the summary by the ACRS staff of the reactors broken down into city, suburban, rural and remote reactors as indicating a need for protection against longitudinal failure of pressure vessels for metropolitan sites. Dr. Bush recalled that under pneumatic conditions, vessels have occasionaly failed in a brittle fashion even when above the Nil Ductility Temperature (NDT) if cracks are present; both the Naval Research Laboratory (NRL) the British metallurgy groups agree on this. According to Mr. Etherington, the late reactors, which are bigger and nearer to cities, probably will all have accompaning improved safety measures.

The extension of the performance of reactor systems, e.g., higher heat transfer rates, and relaxation on acceptable flaw size in vessels, was predicted by Dr. Okrent because of economic pressure. The proliferation of reactors will probably be accompanied by a standardization in designs, according to Dr. O'Kelly. Dr. Hanauer noted that the next reactor at Indian Point might be a GE design; he recommended a consistent committee position on pressure vessel failure for the General Electric (GE) and Westinghouse reactors. The extreme reliability need for the engineered safeguards was noted by Dr. Hanauer.

Dr. Ergen suggested that the controlled release of the hydrogen from a core melting accident with the accompanying noble gases would result in casualties, but this would prevent a pressure rise which might endanger the containment; nevertheless, the applicant would probably not accept this fatality hazard. Not much has been done towards the emergency plans for this reactor. The position of the Regulatory Staff (RS) regarding a reactivity transients is not clear; the consequences of a fire in the control room have not been studied. For RS was reported satisfied with the planned containment testing at 47 psi, with a claim that it would be satisfactory at 70 psi.

The proposed arrangement to retain the melted core in a water cooled refractory lined structure in the lower part of the pressure vessel cavity was viewed as a step in the direction of safety; but the Committee saw this as no guarantee of retention. Dr. Monson predicted that a melted core would slowly pass through the pressure vessel wall rather than in a large mass, but others saw the possibility of a larger portion falling through into water underneath. A demonstration of such an arrangement was seen as desirable by Dr. Mangelsdorf; nevertheless, any analysis of the results of any demonstration would be lengthy. The retention of the molten core in the proposed refractory basin was considered by Dr. Ergen as not much different than any retention in the soil underneath. The need for studying other large reactors with respect to this melting accident was observed by Dr. Newson, Dr. Hanauer believed that accident studies by Oak Ridge groups indicate that the containments are of less service for accidents with very large reactors.

Regulatory Staff (RS)

Mr. Case said the plant is an improvement over the Brookwood design. Originally the Westinghouse group claimed that adding water to the molten core of this reactor would prevent any melting through the vessel. Although the containment for this reactor can probably withstand more metal water reaction then other designs, the RS considers this probably not good enough; to the RS, the core melting accident needs more study. The reactivity transient analysis by Westinghouse showed that slightly less reactivity additions are acceptable than do the GE analytical models. Control rod worths are to be limited by procedures and by devices; calculations on transients are necessary to establish the control rod requirements. The possibility of reactor pressure vessel rupture must be faced now. The RS is satisfied that the high head safety injection system will limit core melting. The RS has recommended improvements for the low head core flooding system. Mr. Case commented on the reliability of emergency equipment under accident conditions, e.g., in a steam environment.

Many operator actions are required during any emergencies; loss of off site power and fires could lead to such conditions. The circuitry has been examined by the RS with respect to fire hazards; no deficiencies have yet been found. Fires have occurred in reactor control rooms but no case of serious consequence is known. Nevertheless, Dr. Okrent recommended careful examination of the circuitry to see that effects of fires would not be serious.

Mr. Case prefers steps to retain any molten core material in the pressure vessel rather than rely on other measures to handle the released core, e.g., the collection basin for molten fuel underneath the reactor. A portion of the wall or pipe of the primary circuit of this reactor designed to break at a lower pressure than would the pressure vessel; this is to provide a safety release and help assure pressure vessel integrity and, hence more assurance of successful core cooling. However, this release scheme might not protect from a pressure pulse in the vessel itself.

The possibility of pieces from a disintegrated turbine damaging the reactor has not been of much concern to the RS. Mr. Case observed that turbine locations are a matter of economics; most are parallel to a tangent to the circular containment which might favor turbine parts impacting on the containment. Multiple reactors at one site complicates this problem of turbine orientation.

Indian Point II Group

The redundancy in emergency equipment was described. Three high head injection pumps outside the containment have four connections to the reactor system. Two low head pumps are inside the container; these are backed by two other low head pumps for residual heat. These residual pumps are outside the containment and have four connections to the primary system. A sump in the containment allows these residual pumps to circulate water on a long term basis. Automatic activation of the emergency cooling system is by three pressure transducers and three liquid level signals; coincidence of two out of three of the low pressure or the low level systems is required. A primary water charging pump is normally operating, and a second will star on a low pressurizer level signal. In addition to this automatic activation, manual starting of the system and the components can be done from the control room.

For a pipe severance with a four square foot cross section opening, the three high head pumps would provide 1450 gallons per minute with a 10 second delay and the two residual head pumps or the two recirculation pumps could provide 4800 gallons per minute with the same delay. Under one percent zirconium water reaction was assumed to occur for this case. For partial power to the emergency system, i.e. the diesel power, one of the high head pumps at 650 gallons per minute with a 50 second delay and one residual heat pump with 4000 gallons per minute with a 20 second delay would be available.

The sequence for diesel operation is for loss of power at the 480 volt buss; the diesels start automatically and if one should fail its buss, automatically connects to that buss which is energized. The emergency equipment which can operate from the diesels are two residual heat pumps, two recirculation pumps, two container spray pumps, two service water pumps, five fans, and two motor control centers. After the level in the refueling storage tank is withdrawn from 350,000 gallons to about 30,000 gallons, the operator may decide on recirculation and manually terminate the safety injection signal; later he would start the one component cooling pump, and the third service water pump to add service water to the cooling heat exchanger. The addition of the 320,000 gallons of water and collection in and around the vessel would raise the level to about 4.7 feet over the bottom of the core. The top of the sump below the core is about 1 foot above the core structure bottom; this should fill soon from the spray to the containment.

The coolant flow in the core during a loss of coolant accident from a break of the hot coolant line was described. Any vapor in the core can move inward and rise. A consultant from the University of Pittsburgh has been relied upon for heat transfer correlations to analyze such a two phase flow cooling during an accident. The decay energy was considered as that following an infinite irradiation time. Appreciable radiative heat transfer would occur from the fuel rods to the steam. If a hundred pounds per second of steam flows through the core (which is a fifth of the capacity of the low head system) the amount of fuel melting and dropping to the bottom of the vessel is predicted to be not more than 20%. Nevertheless, Mr. Etherington doubted if the fuel pellets would have much lateral supports from the oxide of the fuel cladding; therefore, collapse of more fuel and falling to the vessel bottom seems likely. Some of the steam may go into the formation of oxide from the cladding, which would reduce that available for core cooling.

The consequences of a rupture of the cold coolant line to the reactor was also given. Sixteen seconds for loss of coolant is estimated with addition of emergency water from the high head and the low head systems in about 10 seconds, Most of this initial addition of water would be to the hot coolant line. About 95 seconds is estimated to fill the vessel to the bottom of the core with the steam supplying the water for the zirconium reaction. The vessel would be expected to fill to about 2/3 of the height, at which time a 3 psi back pressure would react somewhat against the high head coolant addition flow. Later the core should cover with no fuel melting and only one percent zirconium water reaction. Recent Harwell data was mentioned to indicate that even with temperatures of 500° F. initially, water would still flow down the rods with the steam flowing up.

The control rods of this reactor are in thimbles. The thimble design has not been examined with respect to possible forces following loss of coolant or blowdown of the system; local voids are believed sufficient to shut down the reactor in perhaps 3 seconds even with the rods removed. The flooding of the core would be with water containing three thousand parts per million of boron; about two thousand are required for shutdown. If the molten core should accumulate in the botton of the vessel, decay heating would tend to impair the integrity of the bottom. A possible eutectic is molten at 2400° F., which is not much different than the melting point of steel. To protect the system against molten fuel falling through the vessel wall, a water cooled refractory basin underneath is proposed. The molten fuel would fall through water and to this basin. The 220 thousand pounds of uranium dioxide in the core material with about fifty thousand poinds of zirconium, and some of the pressure vessel bottom, would provide the molten mass for this basin. The experience in steel mills and at Batelle Northwest Laboratories indicates that dispersal of this molten material in fragments would occur in the water.

The analytical model for the basins thermal behavior is based on a square shaped stainless steel container lined with magnesia fire brick several inches thick; inside of which would be solid uranium dioxide, then a molten mass of liquid uranium dioxide with some zirconium, and steel, and finally a vapor phase. Solid uranium dioxide is expected on the top of this fused material with water and steam then on the outside of this.

An analog computer was used to study the heat transfer through this basin of molten fuel with thickness layer of solid uranium dioxide on the top of the melt as a variable. A plot of the temperature of different sections of this basin against time was shown. Temperatures went to about 6,000° F., the boiling point of uranium dioxide, and time to about 5,000 seconds, mass transfer of the vaporized materials and circulation in the melt was assumed for the heat transfer to the water above.

If the metal of the bottom of the pressure vessel, 30 tons, and all the zirconium, reacted with the water a large amount of hydrogen would be produced; bubbling through the water should cool it, but burning is a possibility. If the site were on limestone and the molten material penetrated into this region, then the carbon dioxide formed might add to the pressure in the container and a downward path might be available for release of fission products to the atmosphere. The Westinghouse group considered this basin a satisfactory device to retain the molten material from the core, but no data is available as support. Limited experiments may be possible, but none is proposed yet. Nevertheless cooling of the core is being relied upon by Westinghouse to retain fission products in the pressure vassel.

Little lateral pressure is seen in the core, and the fuel rods are predicted to stand in place. No problem of buckling of the core support structure is expected during an accident. The mechanical forces on the internals from a coolant line break are believed insufficient to cause significant core damage. Careful supporting of the heat exchangers, the pumps, and the manifolding lead to group to expect no serious pipe whipping effects. Because the designers feel that a jet aircraft would not be able to penetrate the container, it seems unlikely to them that a missile from a turbine is a hazard.

The worth of an ejected control rod might be as much as .6% multiplication constant (k) which would be, perhaps, only .3% at full power. The boron needed for the Selni Reactor was over predicted by 100 parts per million (ppm) when cold and 60 ppm when hot; at power with equilibrium poisons, the predictions was under by 25 ppm. The moderator reactivity temperature coefficient was originally under estimated and is now .1 x 10^{-5} more positive. This amounts to about a 5% inaccuracy in the boron worth.

The fuel lattice for this system is under moderated. Hence loss of the core structure would cause a positive reactivity change; if one fuel assembly were spead over the space normally occupied by two, the increase in k might be as much as .3%. Recent calculations on possible dispersal of the fuel laterally in the water show the k reduced by 10%; with fuel slumping, a 4% increase in k might result.

In conclusion, the Consolidated Edison group were told that probably no letter would be issued at this meeting; the group replied that the time schedule is becoming short.

DRESDEN III REACTOR

Executive Session

Mr. Etherington reported for the subcommittee with the help of the minutes of the meeting of 7 July 1966. Possible melting of the core was given particular attention at the meeting. Many other high power level reactors present this same problem; the only clear solution is cooling to prevent melting, but now cooling is not considered a design criteria for this plant. Melting would probably result in pressure vessel failure, over pressurizing of the containment, and in deterioration of the concrete underneath. Concern in the design of the cooling system is evidenced by the core sprays to be relied upon by the applicant to prevent containment over pressure. The group has reported it incredible that no water would reach the core to limit melting hazards; Nevertheless, Dr. Ergen recalled that the core sprays of the SL-1 reactor were considered inoperable because of the violent vibration when water was added to this spray ring. Partial submergence of the core should result in the steam cooling the upper parts. Recommendations of the subcommittee included: no shared emergency coolant systems, precautions against missiles from a turbine failure penetrating the containment, and appropriate steps to resist seismic forces.

The comments of Congressman Hollifield and Aspinall were referred to by Dr. Okrent to indicate a desire for much conservatism in reactor designs, but, nevertheless, a request for streamlining the safety review procedures was included. Emotional sabotage, e.g., the acts of unstable individuals, was feared by Dr. Bush as a reactor hazard. Dr. Okrent believed that the General Electric Company (GE) sees no way now to guarantee retention of a molten core; however, if no economic restrictions were placed on the design, prevention of melting would probably be possible. The GE group considers any AEC approved reactor as a proven design even though it has not operated. Dr. Newson recalled that although GE assumed cooling of the core was possible, the build-up of hydrogen and possible recombination with the oxygen would be a serious problem; more containment to hold the volume of materials would be a solution. The Canadians were reported planning to vent the possible exhaust of six reactors to one containment. Dr. Palladino stated that Westinghouse proposes cooling the bottom of the containment of the pressure vessel to prevent any vessel failure from core melting.

The uncertainty in the succession of operations which are needed to assure the effectiveness of the engineered safeguards was pointed out by Dr. Hanauer. But Dr. O'Kelly said this reactor is in the design stage where not all features have been identified; he favored the combination of core spray and core flooding to prevent core melting, but the design was seen as incomplete. More reliability in flooding arrangement is desirable. Dr. O'Kelly stated that equipment may deteriorate more when not used because of the lack of maintenance. Dr. Ergen observed that since heat must be removed from the containment vessel following a large accident, an active component is needed to assure this heat exchange. The GE group is depending on the reliability of core sprays.

Dr. McKee suggested that volcanologists might have knowledge bearing on the behavior of the molten material from a core as it penetrated the containment and entered the underlying earth. Dr. Monson postulated that although such experts might predict retention of a melted core by the underlying rocks, other possible hazards, e.g., a steam explosion, to violate the containment, still exists; past reactors have the same problem. Dr. Ergen told the Dresden group that Dr. McCullough, in a American Nuclear Society (ANS) paper, stated that a molten core would penetrate the primary system of the Selni Reactor in a matter of minutes unfortunately, no further speculation was included on the fate of the melting material. Core melting has been considered to be a problem with fast reactors. Originally, the Fermi Reactor included a graphite crucible underneath to catch a molten core; later this was removed when graphite changes were made.

Dr. Okrent listed the ACRS alternatives: approval in the same way as Dresden II; approval with core sprays; approval but with reservations on the design of the core sprays; precautions to handle any molten core; and finally the reactor could be rejected. Dr. Palladino considered that approval of Dresden II implicity included acceptance of unit III. However, Dr. Hanauer stated that the Committee must be receptive to new information which may allow acceptance of risks with past plants but changes may be needed for future reactors. Little effectiveness of the containment sprays is seen by Dr. Okrent should the core sprays fail. Any failure of the pressure vessel would tend to complicate the cooling of the core.

A memorandum of Mr. Fraleys of 11 July 1966 titled "Metropolitan Siting Criteria - Tabular Hardward Approach" was reviewed; reactors were categorized according to city, suburban, rural, and remote locations; e.g., city reactors included the proposed Edgar Station of the Boston Edison Co., the Ravenswood reactor and the Palo Seco plant. Dresden III was a rural reactor, the underlying basis was that city reactors must be protected against all accidents. Suburban reactors must be protected against a large accident. Rural reactors need containment with only one method of preventing core melting. While remote reactors need one method of accident prevention.

Mr. Etherington suggested a plot of the logarithm of the consequences of a reactor accident as the ordinate against the logarithm of the probability of an accident as the abscissa. A lower line for perhaps loss of a thousand lives with a probability of one in a million might be acceptable while a consequence beyond this might not be. The difficulty of defining a acceptable relation was noted. Other serious hazards for big reactors beyond the melting accident can probably be postulated. If a vessel suddenly ruptured or the top portion separated from the bottom, Dr. Bush predicted vertical motion or jumping of the vessel. Dr. Bush said he could find no difference between the inspection and fabrication methods of pressure vessels of the General Electric and the Westinghouse groups.

The early ACRS letters asking for further experiments on the pressure suppression system were recalled by Dr. Okrent; after satisfactory demonstrations, Committee approval was given. To him, the core sprays arrangement is a parallel case needing development and demonstration. Dr. Monson did not favor any conclusions on precautions against the core melt down accident now; consideration with future cases should assist in establishing an ACRS position. As new information becomes available, Dr. Okrent forecast further restrictions in the design of future large reactors; further steps towards precautions against a melt down accident are favored by him now.

Dr. Mangelsdorf said that established systems can be extended somewhat by calculations; however, this meltdown accident is much different. The many uncertainties require much information and he was against setting a time at which a protective system would be available. The effectiveness of the systems to contain the melted material from a core is not clear, and hazards may be worse than proposed. Dr. Mangelsdorf recommended further search for measures against this melting accident.

Leaving some uncertainties in the design of the plant at this construction permit stage seemed necessary to Mr. Etherington, as was done for the Brookwood Reactor containment. Dr. Palladino was comforted by the additional redundancy planned for the Dresden III plant. Dr. Monson predicts no solution to this core melting problem for many months; he desires no negative responce which would mean considerable delay in other reactor cases. Several considered that a reasonable measures might be taken now to limit the hazards from such an accident with research and development to continue.

Dr. McKee noted the common source of water for the emergency systems, namely the suppression pool, which makes the emergency reactor systems not completely independent. To Dr. Palladino accident prevention rather than amelioration, e.g., collection basin for a molten core, is preferred. A separate letter regarding core melting rather than including recommendations in the Dresden II letter, was proposed by Dr. Hanauer. The consensus was to delay a letter on this reactor but it was decided to start formulation of preliminary drafts. A delay in release of any general letter on the core melting problem was seen by Dr. Ergen.

Dr. Mangelsdorf reiterated that any prevention of core melting must be by the primary system; he has little confidence in the Westinghouse proposal for the Indian Point Reactor towards collection of any melted core, and he doubted that the Westinghouse group really considered the collection proposal anything more than a method to obtain ACRS approval. The GE group was seen by Dr. Palladino as proposing no solution to the core melting problem because of the higher costs for such precautions for their boiling water reactors. Dr. Ergen said that allowing some release of radioactive materials at Dresden but with appropriate evacuation procedures might be a way to rationalize any Dresden III reactor approval. Because of this lack of assurance of the reliability of a basin to collect molten material, as is proposed for the Indian Point reactor, the Committee agreed not to ask for such core melting provisions for the Dresden III Reactor; nevertheless, the Committee was not against such efforts for the Indian Point II Reactor. Dr. Newson believed that degassing the molten material from the core and collection of these effluents might leave the residue with little fission product heat from any melting into the earth. There was divided opinion as to the need of documentation for these proposed provisions to retain a molten core from the Indian Point II Reactor. Dr. Okrent plans a special meeting to continue the discussions.

Regulatory Staff (RS)

The Dresden I Reactor is undergoing maintanance. Mr. Case said the safety analysis already presented is still the RS position.

Additional information on vibration damping is being obtained to satisfy Dr. Newmarks questions; the problems appear resolvable. Now two separate core containment spray systems for each reactor are planned instead of the shared system. The RS recalled requests from other reactors to rely on only one emergency pump.

The RS has decided to require complete redundancy for the Dresden III facility with the exception of some items, such as emergency diesels, which have outside power as an alternate.

The reactor structure is designed to resist 300 miles per hour winds; tornados causing an accident is not believed likely since the reactor can be shutdown on receiving a tornado warnings. The upper limit of the speed of objects in a tornado is probably that of sound; Mr. levine said that the Southwest Experimental Fast Oxide Reactor (SEFOR) had also used a figure if 300 miles per hour wind speed for the design. The possibility of large objects, e.g., automobiles, falling on the containment during a tornado was admitted by the Dresden group which appeared to Mr. Case to be a separate missile problem for that of turbines on primary system parts.

With respect to the tornadoes, Mr. Waterfield stated that although no reinforced concrete building had failed in a tornado, windows have broken to release the pressure; no answer was given to the question of Dr. Bush about the experience with grain elevators in tornados. Mr. Case, saw the containment as possibly violated by a tornado.

The refueling accident is not considered serious for this reactor, but the GE group will continue to analyze, since the reactivity affects are not yet clear. The GE conclusion that no steps to retain molten core material are feasible is taken by Mr. Case as evidence of GE conservatism. The two core spray systems planned for the reactor may have some weak points; both must discharge into the core barrel arrangement to be effective. GE has suggested a second system of a different design, e.g., flooding, but with the same cooling objective.

Although Dr. Beck said that the RS considers a meltdown accident as presenting a difficult situation, the low probability of the safeguards failing at the same time leads the RS to accept this melting hazard. Dr. Doan considered any conclusions on core melting for Dresden III as also applying to Dresden II; although important, he sees no reason why this possible accident should delay these reactors. Dr. Doan recalled no consideration of core melting of the Dresden I Reactor. Although coolant loss and core melting was considered by the Fermi Group, Mr. Levine said that coolant would always be adequate to assure no melting. The melting problem is a long range one, and the RS is not prepared to recommend measures now. Dr. Beck reported the RS is now focusing sharply on the effectiveness and reliability of safeguards sytems. The RS believes that the amount of design, research, and development now done leaves it in a better position to assure public safety from reactor accidents than in the past.

Dresden Group

The reactor is inside a surrounding supporting shield twenty feet in diameter; containment sprays and circulation pumps are inside the dry well. Fabrication of the pressure vessel is well along. Mr. Bray commented for the General Electric Group (GE) on the proposed meltdown accident and possible measures. The separate phases of such a postulated accident included; melting of the core, vessel bottom failure, structural heating, accumulation of the core melt in the sump, and finally the effects on the containment of the molten material. The decay heat of the core might be 70 MW initially, which would decrease to about 20 MW in a matter of hours. The fuel temperature would rise rapidly on coolant loss and be perhaps 4,000° F. at 400 seconds; studies have been made of the amount of cladding which would be above such temperatures.

The liquid core would melt through the rod structure underneath this boiling water system and collect in the sump on top of the thick layer of concrete lining the dry well. About 1200 square feet of area is available at the bottom to receive this molten material. The GE group presented a plot showing the temperature through the concrete at the bottom of the dry well, the dehydrated concrete above, the molten metal and solid uranium oxide mixture, other fused material floating on the surface, and the water steam blanket at the top.

The estimated heat fluxes for different surface temperatures were given with the thickness of oxide on the top as a parameter. About half of the heat transfer from the molten mass would be by radiation. The uranium dioxide is not assumed to melt by GE, but others have predicted melting. Probably not much metal water reaction occurs below a temperature of 2500° F. The containment pressure as a function of percent metal water reaction was given for both an inert containment and for the reaction with oxygen; this approached perhaps 200 psi for 100 reaction. A steam explosion in the pressure vessel might yield about 0.6 million foot pounds of energy, which is about 1/3 of that needed to break the pressure vessel.

Studies have been made of flooding the containment region, e.g., cooling the bottom of the pressure vessel with water. The vessel wall would be at about 4,000° F. on the inside of 250° F. on the outside. Perhaps six feet of molten material would be in the vessel, which would probably not retain this material. A steel pan with a refractory liner, e.g., aluminum oxide, with cooling coils underneath to catch a released molten core was proposed. A forty-eight foot diameter pan would be needed. The uranium dioxide as a solid might collect in the pan to cause over heating and failure of the cooling coils. Possibly the center of the pan could be raised to allow a distribution of this fused material. The GE conclusions on this collection crucible approach are; a feasible design may be possible, the hydrogen production would have to be limited, major design changes would be needed to accomodate the system, the vessel supports would probably have to be insulated and cooled, heaping of the molten mass would be a problem. Dr. Newson paralleled the situation of such a molten core with that of open-hearth furnances which undergo continued heating with their integrity maintained.

Another possible approach is to have below the core a water tank into which the melt would fall. A gas collector of about a million cubic feet at 62 psi would be needed; about 300,000 cubic feet of volume is available in the present dry well and suppression system. This water collection approach was considered technically feasible if the dry well walls are protected; hydrogen production requires a larger gas volume and changes in the structure below the pressure vessel would be required. The heat transfer characteristics through any molten mass is uncertain. Whether the molten core would fragment on passing through the water underneath is questionable.

Another GE approach to handle a molten core was a ceramic pebble sieve support in a sleeve; the molten material would be fluidized above the sieve with water, perhaps 20,000 horsepower of pumps would be required. The metal water reaction would be limited. Retention of the fluid bed would be difficult. Cross changes in the structure of the containment would be needed. Air cooling the core is another possibility. To prevent core melting, a large air flow rate would be required, and assuring air flow would be difficult. The feasibility of this air scheme seemed doubtful to the GE group.

In summary, the GE group said that the current containment can not handle the consequences of such a core meltdown and the proposed design methods to ameliorate such an accident fall short of satisfying the engineering need. Much research development would be needed towards any melting accident precautions and this would vary with the approach. The basis for the present reactor design must be prevention of core melting by emergency cooling.

Mr. Levy presented information on the core cooling arrangements. Two core sprays are supplied from the suppression system. Potential improvements in the plant design might be provisions for loss of coolant but with slow depressurization. The core spray system, with redundancy of the active components such as two pumps or a high volume flooding system, again with redundant active components, might be used. Two core sprays each with redundancy are another possibility.

The recirculation system is proposed to add water to the reactor core on coolant loss; this is believed capable of handling the slow depressurization accident and provide core cooling during any loss of power. Steam generated in an accident might be used to operate a turbine to pump water from the suppression chamber. The steam condensate of the turbine would go to the suppression chamber. The pressure of the system was shown against time with or without the recirculation system, the initial pressure of about a thousand psi reduced to about 400 psi at something over a 1000 seconds, and with no recirculation cooling arrangement an increase followed.

The design assumptions for the present core spray systems are to assure core cooling, structural integrity, quality control of the material, and continued surveillance and testing of this core spray. Tests on the spray effect have been made with rods heated as high as 1400° F. at the start with a peak temperature of 1925° F. Tests performed on the quench ring or core sprays has been with the ring cold. The effect of a heated ring was not predicted by the GE group; substantial distortion is believed allowable in the core spray system with no impairment of reliability. Wetting of the fuel is not taken as a heat transfer credit for the effectiveness of the core cooling; although some of the spaces in the core might be blocked on cooling loss, substantial flow stoppage is believed incredible.

The predicted temperature of the fuel for a 30 second delay in operation of the core sprays after an accident was shown, with the reaction of the fuel at the different temperatures as a parameter. All the fuel would be above about 500° F. with none above perhaps 1500° F.; similar curves were shown for a 60 second delay. The pressure load in psi for the design value, the capability, at steam line severance, and for recirculation line severance was given for; the spray sparger and internal pipe, the upper shroud, the lower shroud, and at the shroud support. For the spray sparger and internal pipe, these were 250, 1000, 16, and 8 psi, respectively; the normal pressure for the spray sparger and for the internal pipe regions are 8 and 107 psi, respectively. For the shroud support, the normal pressure is 22 psi; the other pressures would be 100, I00, 54, and 22, respectively.

The quality of the core sprays is believed by the GE group to be equivalent to that of the pressure vessel. Surveillance on core spray reliability will be by pressure measurements and measuring the level in the suppression chamber. The total system will undergo thorough preoperation tests. During a hot standby condition for the reactor, loop tests will be done, and the total system will be visually inspected during refueling. Extensive inspections of the Dresden I Reactor are underway; fine cracks in the vessel metal have been detected, but these have not enlarged. Dr. Bush said that he had lost some confidence in the reliability of type 304 stainless steel, e.g., because of intergranular failure; the GE group was admonished to be careful in the fabrication and use of this material.

On loss of power, one isolation condensor and one oversized core isolation cooling system would be in service. For loss of coolant and slow depressurization, reactor isolation and operator blowdown would occur. On loss of coolant and rapid depressurization, one core spray with redundant active components and one core flooding system would be available.

The core flooding is to take water from the pressure suppression chamber when a level of the pressure vessel reaches 150 psi. The higher rate of the core spray whould allow very rapid addition. Although the core flooding arrangement is not fully designed, a reliable system is seen as feasible. The pressures of the core flooding elements for; normal operation, ASME design value, ASME capability, line service steam, and line service of the recirculation system were given at the core plate region, the guide tubes, and the jet pumps. At the core plate these would 14, 45, 53, 38, and 14, respectively, and 22, 70, 600, 54, and 22, respectively at the jet pumps. The flooding system can operate from emergency power, e.g., diesel engine. Test of the flooding ring would be at 500° F. and 1000 psi. The GE group is confident that the flooding water will enter the system; Dr. Ergen recommended tests to see if transients might give vibrations to violate the system.

The GE group was non-commital about any variation of engineering safeguards requirements for the reactors with sites; the best available safeguards are used regardless of location. The principal aim is to avoid core melting. No practical solution for withstanding the core melting accident is seen by the GE group, and GE plans no development towards this end. Although practical emergency core coolant arrangements can be made for operation at higher pressures, e.g., a 1000 psi, this would be expensive; in addition, the hazards to the reactor from false activation of such a unit, with related reactivity changes from void effects and thermal stress problems, weigh against its use. Mr. Levy said the emergency cooling systems need to operate for a day or so after the accident; after flooding of the core, cleaning devices on the water would be used to allow opening the reactor vessel and fuel removal from the core region. Some heat removal capability would be needed for an indefinite time. More information on the reliability of the core cooling system should be available in September; however, it will be months before significant data on the possible core melting precautions can be developed. Any firm proposal on prevention of a core melting through the pressure vessel is perhaps a year away; all proposals lack information and understanding of the processes and a demonstration is needed. Mr. Joslyn said that the core melting problem involves many reactors beyond the Dresden III design; the effort already expended for the STEP and the LOFT projects indicates the extent of research and development.

The plant is believed to have had a thorough analysis for tornado damage, e.g., external missiles and the critical parts are believed protected by the concrete structures. A tornado with winds from of 300 to 500 miles per hour would probably remove a part of the reactor building, but all the key equipment of the plant, e.g., the diesel generator and the reactor, is below ground level and should not be damaged. Nevertheless, the outside electrical sources might be damaged by such a violent storm, Mr. Ergen said that a tornado after a reactor accident might lead to a more serious release of fission products. Dr. Ergen postulated a million curies of noble gases and a 100,000 curies of halogens released from the core; filters decontaminate by a factor of a 100 and dilution reduces concentration another 50 to 500 factor.

It is rather clear from the minutes of the 75th meeting that the ACRS was fairly well convinced that means of coping with a fully molten core within the framework of the General Electric suppression pool design, and probably within the framework of the Indian Point 2 design, were not readily available and would be difficult to demonstrate on a short time scale. What is less clear is what direction the Committee thought the review of these two construction permit applications should take. There was a considerable divergence of opinion on the Committee.

In the days which followed, members tried to develop possible points of view that either the Committee might use in its continuing evaluation, or that might express the point of view of the individual member at that point in time. On July 16, 1966, member Newson submitted the following memornadum to his fellow Committee members:

REPORT ON DRESDEN III

The applicant and the General Electric Company have informed the Committee that no provision had been made to maintain containment in the unlikely event of a core melt-down following loss-of-coolant.

Under these circumstances, the applicant must design for one of the following criteria to be applied at the time of issuance of an operating license.

1. The licensed operating power of the reactor must be reduced to the point where a molten core may be cooled efficiently enough to prevent melt-through of the bottom of the containment vessel or
2. An emergency cooling system, far more reliable than any which now exists, must be designed and installed with sufficient precautions to convince the AEC Staff and this Committee that after a loss-of-coolant accident at design power, there is as much assurance that core melt-down can be prevented as the reliability of conventional containment for relatively low-power reactors.

At the present time, neither an efficient cooling system for a melted core nor a highly reliable emergency cooling system have been invented so that it is impossible for the Committee to advise that Dresden III reactor may be built and operated at the proposed site and power level without undue risk to the health and safety of the public. If a construction permit is issued before the development of these novel engineered safeguards, the safe operating power can only be determined after their construction and the demonstration of their capabilities.

This was followed by the preparation of first drafts of possible ACRS reports on Dresden 3 and Indian Point 2.

On July 28, 1966, member Hanauer circulated the following memorandum to other ACRS members:

THE UNIVERSITY OF TENNESSEE
KNOXVILLE
DEPARTMENT OF NUCLEAR ENGINEERING

July 28, 1966

TO: ACRS Members

FROM: S. H. Hanauer

SUBJECT: Tentative position on fuel meltdown

The following thoughts have been stimulated by the recent Okrent draft on "Primary System rupture" and by the Wilcox memo of July 25 stating that Dresden Subcommittee members would be asked on the telephone for an "opinion on the acceptability of the emergency core cooling systems now proposed for Dresden 3."

1. The primary system rupture is one of the design-basis accidents for hazards analysis for Dresden 3, Indian Point 2, and lots of other reactors. In discussing its consequences, therefore, or the adequacy of safeguards to cope with this accident, it seems to me that adversion to the improbability of the accident is out of order. We should indeed require steps to be taken to reduce the probability, but that is a separate subject.

2. In view of the potentially serious consequences of the primary-system rupture, it is my present view that each reactor should have two defenses against this design-basis accident. One of these defenses might be a well-engineered, redundant system to put water onto or into the core. However, because of the many uncertainties regarding the functional adequacy of pouring water (emergency core coolant pipe rupture as a result of the accident, fuel melting in spite of success in getting water to core, loss of cladding strength leading to blockage of coolant channels or dropping fuel to the bottom of the vessel, steam explosions, steam blanketing, etc.), a second defense of a different species should be provided. Item 5 of the Okrent draft seems to require this.

3. The "crucible-catcher" approach, currently touted by Westinghouse and downgraded by General Electric, seems to be the only "second defense" seriously proposed thus far. I have discussed some aspects of this problem with colleagues who design nozzles for fluids at 8-9000° F. Their successful approach has been to use thin walls of high-conductivity metal cooled so fast on the back side that only a small gradient can exist in the metals even for heat flux $\sim 6 \times 10^6$ btu/hr - sq. ft., and that the gradient is then forced into the boundary layer of the flowing fluid. Although they have made no calculations, they suggest that pool boiling is almost surely inadequate to cool the metal, but that good forced cooling might be successful in our situation as well as theirs. Of course problems remain such as hydrogen (burning, pressure, explosion), cooling reliability, ultimate heat sink, local burnout, chemistry and possible attack on the metal, and so forth.

4. Alternatively, one might abandon this accident as a design basis. (The ATR safety analysis is based on the incredibility of gross primary-system rupture.) For power reactors, our pressure-vessel discussions over the past year make the acceptability of this approach doubtful.

5. Regarding the forthcoming telephone call, I don't know what emergency core cooling system is now proposed for Dresden 3 (or Indian Point 2, for that matter), and we have no agreed-on basis for any opinions on its acceptability.

What I really think we have to have is a good core cooling complex with an analysis which shows that if it works the vessel won't fail, plus some other complex with an analysis that shows a reasonable probability of averting danger to the health and safety of the public if the core cooling does not provide the expected protection.

cc: R. F. Fraley

On August 16, 1966 the Committee sent out two reports, one on Dresden Nuclear Power Station, Unit 3 and another on Indian Point Nuclear Generating Unit No. 2. These letters are reproduced on the following pages.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

August 16,1966

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON DRESDEN NUCLEAR POWER STATION - UNIT 3

Dear Dr. Seaborg:

At its seventy-fourth meeting, on June 8-11, seventy-fifth meeting, on July 14-16, a special meeting on August 4-5, and its seventy-sixth meeting on August 11-13, 1966, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Commonwealth Edison Company to construct a third nuclear power plant at the Dresden site, near Morris, Illinois. Unit 3 will include a boiling water reactor to be operated at 2255 MW(t) power level with pressure suppression containment. Unit 3 would be similar to Unit 2. The Committee had the benefit of discussions with representatives of the applicant, the General Electric Company, Sargent & Lundy, the Babcock & Wilcox Company, and the AEC Staff, and of the documents listed. A Subcommittee of the ACRS met to review this project at the Dresden site on June 2, 1966, and in Washington on July 7, 1966.

In its report on Dresden Unit 2, dated November 24, 1965, the Committee recommended that the AEC Staff follow development work by GE to resolve particular design problems. The Committee recommends that the Staff continue to follow the development work in connection with both Units 2 and 3, particularly with respect to operation with jet pumps, testing of emergency cooling methods, and studies of reactivity transients to assure no impairment of emergency cooling effectiveness as a consequence thereof.

The Committee also urged that the designers pay particular attention to the design of the pressure vessel, and of the high pressure steam lines with their isolation valves and fittings. The Committee reiterates its opinion on this matter in connection with Unit 3.

The Committee notes that the applicant has made improvements in the requirements for pressure vessel inspection during fabrication and urges that the applicant pursue vigorously the implementation of adequate in-service inspection techniques.

The effectiveness of emergency core cooling systems is a matter of particular importance in the unlikely event of a pipe rupture in the primary system. The applicant proposes the following improved complex of emergency cooling systems:

1. a high pressure coolant injection (HPCI) system,

2. a high-volume flooding system to permit rapid injection of water into the reactor vessel following blowdown to a low pressure,

3. two core spray systems,

4. a system that will make river water available to the feedwater pump for emergency cooling.

The applicant advised the Committee that equivalent changes in the emergency core cooling systems of the Dresden 2 unit would be made. Three diesel-driven generators will be installed to serve Units 2 and 3.

The Committee concurs that the proposed systems should increase the reliability and effectiveness of emergency core cooling. Complete details of the systems are not available, but the Committee believes that these matters can be resolved during construction of this facility. The Committee believes that the Regulatory Staff and the Committee should review details of design, fabrication procedures, plans for in-service inspection and the analyses pertaining to the emergency core cooling systems, as soon as this information is available and prior to irrevocable construction commitments pertaining thereto.

Careful examination of the forces during blowdown on various structural and functional members within the pressure vessel is necessary to assure sufficient conservatism in the design. The Committee recommends that the AEC Staff satisfy itself fully in this respect.

The Committee believes that the combination of emergency cooling systems has a high probability of guarding against core meltdown in the unlikely accident involving rupture of a primary system pipe. In view of the present state of development of such emergency cooling systems, however, and since the cooling systems may be subject to certain low-probability interrelated modes of failure, the Committee believes that the already small probability of primary system rupture should be still further reduced by taking additional measures as noted below. The Committee would like to review the results of studies by the applicant in this connection, and the consequent proposals, as soon as these are available.

1. Design and fabrication techniques for the entire primary system should be reviewed thoroughly to assure adequate conservatism throughout and to make full use of practical, existing inspection techniques which can provide still greater assurance of highest quality.

2. Great attention should be given to design for in-service inspection possibilities and the detection of incipient problems in the entire primary system during reactor operation. Methods of leak detection should be employed which provide a maximum of protection against serious incidents.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactor can be constructed at the Dresden site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

David Okrent
Chairman

References Attached.

References (Dresden 3)

1. Dresden Nuclear Power Station, Unit 3, Plant Design and Analysis Report, Volumes I and II, Commonwealth Edison Company, undated, received February 14, 1966.

2. Letter dated May 3, 1966 from L. F. Lischer, Commonwealth Edison, to Dr. R. L. Doan, AEC, transmitting Amendment No. 1, Answers to AEC Questions, including replacement and additional pages to Plant Design and Analysis Report.

3. Amendment No. 2, Answers to AEC Questions, undated, received May 20, 1966.

4. Letter dated May 26, 1966 from Murray Joslin, Commonwealth Edison, to Dr. R. L. Doan, AEC, transmitting Amendment No. 3, Answers to AEC Questions, including replacement pages to Plant Design and Analysis Report.

5. Supplement to Construction Permit Application, Reactor Vessel Non-Destructive Testing of Plate, dated June 10, 1966.

6. Letter dated July 8, 1966 from Murray Joslin, Commonwealth Edison, to Dr. R. L. Doan, AEC, with attachments.

7. Letter dated July 26, 1966 from F. A. Hollenbach, General Electric, to Mr. E. Case, AEC, with attachment.

8. Letter dated July 29, 1966 from Murray Joslin, Commonwealth Edison, to Mr. Edson Case, AEC.

9. Letter dated August 10, 1966 from W. D. Gilbert, General Electric, to Mr. E. Case, AEC, with attachment.

10. Letter dated August 12, 1966 from M. Joslin, Commonwealth Edison, to Dr. Richard L. Doan, AEC.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

August 16, 1966

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its seventy-fifth meeting, July 14-16, 1966, and its special meeting on August 4-5, 1966, the Advisory Committee on Reactor Safeguards completed its review of the application of Consolidated Edison Company of New York, Inc. for authorization to construct Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the seventy-second and seventy-third meetings of the Committee, and at Subcommittee meetings on March 30, May 3, and June 23, 1966. During its review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants and with representatives of the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed.

The Indian Point 2 plant is to be a pressurized water reactor system utilizing a core fueled with slightly enriched uranium dioxide pellets contained in Zircaloy fuel rods; it is to be controlled by a combination of rod cluster-type control rods and boron dissolved in the primary coolant system. The plant is rated at 2758 MW(t); the gross electrical output is estimated to be 916 MW(e). Although the turbine has an additional calculated gross capacity of about 10%, the applicant has stated that there are no plans for power stretch in this plant.

The Indian Point 2 facility is the largest reactor that has been considered for licensing to date. Furthermore, it will be located in a region of relatively high population density. For these reasons, particular attention has been given to improving and supplementing the protective features previously provided in other plants of this type.

The proposed design has a reinforced concrete containment with an internal steel liner which is provided with facilities for pressurization of weld areas to reduce the possibility of leakage in these areas. The containment design also includes an internal recirculation

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containment spray system and an air recirculation system consisting of five air handling units to provide long-term cooling of the containment without having to pump radioactive liquids outside the containment in the event of an accident. Even though the applicant anticipates negligible leakage from the containment, two independent means of iodine removal within the containment have been provided. These are an air filtration system using activated charcoal filters, and a containment spray system which uses sodium thiosulfate in the spray water as a reagent to aid removal of elemental iodine.

The reactor vessel and various other components of the system are surrounded by concrete shielding which provides protection to the containment against missiles that might be generated if structural failure of such components were to occur during operation at pressure. This includes missile protection against the highly unlikely failure of the reactor vessel by longitudinal splitting or by various modes of circumferential cracking. The Committee favors such protection for large reactors in regions of relatively high population density.

The Indian Point 2 plant is provided with two safety injection systems for flooding the core with borated water in the event of a pipe rupture in the primary system. The emergency core cooling systems are of particular importance, and the ACRS believes that an increase in the flow capacity of these systems is needed; improvements of other characteristics such as pump discharge pressure may be appropriate. The forces imposed on various structural members within the pressure vessel during blowdown in a loss-of-coolant accident should be reviewed to assure adequate design conservatism. The Committee believes that these matters can be resolved during construction of these facilities. However, it believes that the AEC Regulatory Staff and the Committee should review the final design of the emergency core cooling systems and the pertinent structural members within the pressure vessel, prior to irrevocable commitments relative to construction of these items.

The applicant stated that, even if a significant fraction of the core were to melt during a loss-of-coolant accident, the melted portion would not penetrate the bottom of the reactor pressure vessel owing to contact of the vessel with water in the sump beneath it.

The applicant also proposes to install a backup to the emergency core cooling systems, in the form of a water-cooled refractory-lined stainless steel tank beneath the reactor pressure vessel. The Committee would like to be advised of design details and their theoretical and experimental bases when the design is completed.

In order to reduce still further the low probability of primary system rupture, the applicant should take the additional measures noted below. The Committee would like to review the results of studies made by the applicant in this connection, and consequent proposals, as soon as these are available.

1. Design and fabrication techniques for the entire primary system should be reviewed thoroughly to assure adequate conservatism throughout and to make full use of practical, existing inspection techniques which can provide still greater assurance of highest quality.

2. Great attention should be placed in design on in-service inspection possibilities and the detection of incipient trouble in the entire primary system during reactor operation. Methods of leak detection should be employed which provide a maximum of protection against serious incidents.

Attention should also be given to quality control aspects, as well as stress analysis evaluation, of the containment and its liner. The Committee recommends that these items be resolved between the AEC Regulatory Staff and the applicant as adequate information is developed.

The applicant has made studies of reactivity excursions resulting from the improbable event that structural failure leads to expulsion of a control rod from the core. Such transients should be limited by design and operation so that they cannot result in gross primary-system rupture or disruption of the core, which could impair the effectiveness of emergency core cooling. The reactivity transient problem is complicated by the existence of sizeable positive reactivity effects associated with voiding the borated coolant water, particularly early in core life. In addition, the course of the transients is sensitive to various parameters, some of which remain to be fixed during the final design. Westinghouse representatives reported that the magnitude of such reactivity transients could be reduced by installation of solid burnable poisons in the core to permit reduction of the soluble boron content of the moderator, thereby reducing the positive moderator coefficient. The Committee agrees with the applicant's plans to be prepared to install the burnable poison if necessary. The Committee wishes to review the question of reactivity transients as soon as the core design is set.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactor can be constructed at the Indian Point site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
David Okrent
Chairman

References:

1. Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, Preliminary Safety Analysis Report, Volume 1, and Volume 2, Parts A & B, received December 7, 1965.
2. First Supplement to Preliminary Safety Analysis Report, dated March 31, 1966.
3. Second Supplement to Preliminary Safety Analysis Report, received June 2, 1966.
4. Errata Sheets for Preliminary Safety Analysis Report and First Supplement thereto, received June 13, 1966.
5. Third Supplement to Preliminary Safety Analysis Report, received June 22, 1966.
6. Fourth Supplement to Preliminary Safety Analysis Report, received July 28, 1966.
7. Fifth Supplement to Preliminary Safety Analysis Report, received July 28, 1966.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 30, 1966

MEMORANDUM

To : ACRS Members

From : David Okrent, ACRS Chairman per RSR

Subject: DRAFT 3 OF INDIAN POINT 2 REPORT

Attached is a suggested draft 3 of N as $N \rightarrow \infty$ for Indian Point 2. It is built around draft 2 of N. J. Palladino. It assumes there will be a general letter.

Attachment:
Draft 3 of Indian Point 2 Report.

DRAFT 3
NJP/DO:bmd
7/30/66

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its seventy-fifth meeting, July 14-16, 1966, and its special meeting on August 4-5, 1966, the Advisory Committee on Reactor Safeguards completed its review of the application of Consolidated Edison Company of New York, Inc. for authorization to construct Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the seventy-second and seventy-third meetings of the Committee, and at Subcommittee meetings on March 30, May 3, and June 23, 1966. During its review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants and with representatives of the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed.

The Indian Point 2 plant is to be a pressurized water reactor system utilizing a core fueled with slightly enriched uranium dioxide pellets contained in Zircaloy fuel rods; it is to be controlled by a combination of rod cluster-type control rods and boron dissolved in the primary coolant system. The plant is rated at

2758 MW(t); the estimated gross electrical output is estimated to be 916 MW(e). Although the turbine has an additional calculated gross capacity of about 10%, the applicant has orally stated that there is no planned power stretch in this plant.

The Indian Point 2 facility is the largest reactor that has been considered for licensing to date. Furthermore, it will be located in a region of relatively high population density. For these reasons, particular attention has been given to improving and supplementing the protective features previously provided in other plants of this type.

The proposed design has a reinforced concrete containment with an internal steel liner which is provided with facilities for pressurization of weld areas to reduce the possibility of leakage in these areas. The containment design also includes an internal recirculation containment spray system and an air recirculation system consisting of five air handling units to provide long-term cooling of the containment without having to pump radioactive liquids outside the containment in the event of an accident. Even though negligible leakage is anticipated by the applicant, two independent means of iodine removal within the containment have been provided. These are an air filtration system using activated charcoal filters and a containment spray system which uses sodium thiosulfate in the spray water as a reagent to aid removal of elemental forms of iodine.

The reactor vessel and other components of the system are surrounded by heavy concrete shielding which protects the containment against missiles that might be generated if structural failure of components were to occur during operation at pressure. This includes missile protection against the highly unlikely failure of the reactor vessel by longitudinal splitting or by circumferential cracking in the vicinity of the inlet and outlet nozzles. The Committee believes that protection against such modes of failure is particularly desirable for large reactors in regions of relatively high population density.

It is the policy of the Committee to encourage applicants to seek and develop designs that will enhance the safety of the public. The effectiveness of emergency core cooling systems becomes a matter of particular concern in the unlikely event of a pipe rupture in the primary system. The Indian Point 2 plant, as proposed, is provided with two safety injection systems for cooling the core with borated water in such an emergency. Further attention should be given to the capacity of the systems, as well as to evaluation of the forces imposed on pressure vessel internals during a primary system blowdown and the ability of the components to withstand the forces involved so that impairment of the effectiveness of safety injection flow by failure of this equipment can be avoided.

The Committee believes that these matters can be resolved during the construction stage of the Indian Point 2 station. However, it believes that the Regulatory Staff and the Committee should review details of design, fabrication procedures and the backup analysis for the safety injection system and significant structural members within the pressure vessel, prior to irrevocable construction commitments pertaining thereto.

Subject to the above comments, the Committee believes that the combination of two safety injection systems may be considered as providing an acceptable engineered safeguard against the unlikely accident involving rupture of a primary system pipe. In view of the present state of development of and experience with such emergency cooling systems, and since both safety injection systems are subject to certain simultaneous modes of failure, the Committee believes it prudent to pursue additional steps to enhance the public safety.

The applicant has proposed to install a backup to the safety injection systems, in the form of a water-cooled refractory-lined stainless-steel tank beneath the reactor vessel. This tank is intended to receive portions of the core, were they to melt through the vessel into the water-filled cavity below the vessel in the very unlikely event that all other means for cooling the core failed to function.

The above system appears to have promise as a backup system, and the Committee feels that careful attention should be given to its design and that some experimental evidence should be developed to establish its performance characteristics. Attention should also be given to any problems that might arise as a result of gas evolution if the system is called upon to perform. The Committee would like to review the design details and the theoretical and experimental bases for the design upon completion of the design.

The Committee believes it would also be prudent for the applicant to review the design and fabrication techniques for the entire primary system to assure adequate conservatism throughout and to make use of practical existing inspection techniques which can provide still greater assurance of highest quality. For example, ultrasonic inspection techniques should be used to implement radiographic inspection of pressure vessel welds subject to meaningful ultrasonic inspection. In addition, greater attention should be placed on in-service inspection and the detection of incipient trouble in the primary system. The Committee would like to be advised of the results of such a review and the proposals emanating therefrom.

Special attention should be given to the reliability of the service water system during accident conditions, particularly the possibility that the internal pressure of the containment may exceed the pressure of the water in the cooling coils of the air recirculation system. Attention should also be given to quality control aspects as well as stress analysis evaluation of the containment and its liner. The Committee recommends that the applicant further examine the possibility that missiles might penetrate the reactor building as a result of turbine rotor failure. If necessary, the turbine should be relocated to eliminate any serious radiological hazard to the public from this unlikely failure. The Committee recommends that these items be resolved between the AEC Staff and the applicant as adequate information is developed.

The applicant has made studies of the course of reactivity excursion resulting in the unlikely event that structural failure leads to expulsion of a control rod from the core. The problem is complicated by the existence of sizeable positive reactivity effects associated with voiding the borated coolant water, particularly early in core life. In addition, the course of the transients is sensitive to various parameters, some of which remain to be fixed during the final design. In particular, the possible reactivity worth of an ejected control rod and the magnitude of positive reactivity associated with core voiding must be reviewed and evaluated.

Westinghouse representatives reported that such reactivity transients could be mitigated, if found necessary early during core life, by insertion of solid burnable poisons into the core to permit reduction of the positive moderator coefficient by reduction of the soluble boron content of the moderator. This could be done just before operation if appropriately planned for. The Committee feels that the applicant should be in a position to install the burnable poison at that time if found necessary, and that a review of reactivity transients should be made as soon as the core design is set.

With due regard given to the foregoing considerations, the Advisory Committee on Reactor Safeguards believes that the various problems mentioned can be resolved during construction and that the proposed reactor can be constructed at the Indian Point site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

David Okrent
Chairman

Thus, during the month of July, 1966, there was evolving the potential approach that the ACRS would approve construction permits for Dresden 3 and Indian Point 2 on the basis of greatly improved emergency core cooling systems and of measures to improve the quality of primary systems and in other ways to reduce the probability of a LOCA, all this coupled with a general letter which recommended development of a backup safeguard of some kind. This approach was similar to that adopted in connection with the review of Dresden 2, at which time pressure vessel integrity was the central problem, and which was resolved by the issuance of a general pressure vessel letter together with a letter favorable to construction of Dresden 2.

The ACRS scheduled a special meeting, August 4-5, 1966, in order to pursue its review of Indian Point 2 and Dresden 3. It is noted parenthetically that this meeting occurred during an airplane strike in parts of the United States, and at least one member drove the order of 1,000 miles each way in order to attend the meeting. The summary by Mr. Libarkin says the following about this special meeting.

> Several members expressed the opinion that the demonstration of a workable arrangement for the retention of a molten core was impossible. They felt that core cooling offered the best chance for a solution. Others thought that since a core catcher had been proposed for Indian Point 2, the Committee had no choice but to require such a device on Dresden 3. The Regulatory Staff felt that the best way around the melt-through problem was to prevent core melting. The core cooling systems could be adequately designed during construction and they intended to give no weight to the inclusion of a core catcher.

(It is noted that, at the beginning of the special August meeting of the ACRS, the Regulatory Staff was still proposing to accept the core cooling systems as proposed originnally by Dresden 3 and Indian Point 2).

Mr. Libarkin's summary continued:

> For Indian Point 2, Consolidated Edison repeated the point that the retention basin was intended as a reserve safeguard only. Core cooling was the primary safety mechanism. It was noted that for the double ended pipe break, 25 to 30% of the cladding might melt, with a 12 to 15% metal water reaction. The Indian Point 2 applicant was told at the conclusion of the August 4-5 meeting, that the Committee thought it could write a letter and that the list of points that would be mentioned in the letter would include a review of provisions for coping with the results of core cooling system failure. On Dresden 3, there seemed agreement that the core spray proposal was not satisfactory by itself. Flooding might prove to be a sufficiently independent cooling scheme to be acceptable to the Committee. G. E. pro-

posed to increase the capacity of the core spray systems to allow reflooding the core for a larger spectrum of pipe breaks. Both G.E. and Commonwealth Edison rejected the idea of a molten core retention basin. A plant requiring such a scheme would not be acceptable to Commonwealth and would not be marketed by G.E.

Again, to provide detail on the complex decision-making processes involved, a long excerpt from the minutes of the special August, 1966 bmeeting is on the following pages.

EXECUTIVE SESSION

Procedures

A proposed letter to Mr. Farmer, of the British reactor safety group, was reviewed by the Committee. The attempt by the Committee to make private arrangements with the British safety groups has disturbed Dr. Beck; he sees a possibility of the ACRS opinions expressed being contrary to the AEC policy. He prefers that any such arrangements be made through the Regulatory Staff (RS). This view recalled early conflicts between the ACRS and the RS, which was at that time under the General Manager, e.g., the ACRS having consultants was not favored by the RS, and the RS was against Executive Sessions for the Committee. A complilation, dated 29 July 1966, of Mr. Fraley gave a history of the ACRS operations.

Mr. Plaine believed that the Committee's request for meeting with the British was a reasonable one, and he saw no loss of ACRS prerogative if this were done through the RS. The RS has always been quite willing to arrange meetings for the Committee. Mr. Fraley believed that the AEC might be concerned over problems of confidential industrial information of U. S. Companies being transmitted to foreign groups. A more dignified meeting with any foreign group, than just subcommittee attention, was recommended by Dr. Hanauer.

At a recent Commission meeting, Dr. Okrent was asked about Committee measures to handle the increasing work load; apparently, this reflects comments from applicants, e.g., the Dresden group, on delays. Dr. Okrent replied that he was attempting to keep the ACRS at full strength and hopes for an increase in RS responsibility. He made comments to the Commission on the core melting accident.

INDIAN POINT II

AND

DRESDEN III REACTOR

Executive Session

Dr. Thompson was reported by Mr. Fraley to have said that if the containment were lost for the Indian Point II reactor, the consequences to the public would be the same as for a Dresden III accident. Dr. Okrent saw this conclusion being extended to state that Dresden III presented the same hazards as the proposed Ravenswood Reactor, which he doubts. Dr. Ergen stated that, for intermediate size reactors, sites offer differences in hazards, but for very large reactor accidents, there is probably little difference in consequences.

Dr. Thompson was also reported to believe that any demonstration of the retention of a molten reactor core by a structure underneath a reactor would be next to impossible; the long effort towards such an arrangement for the Fermi Reactor with no results was recalled. The Fermi group finally resorted to an additional coolant system with sodium, which parallels the proposed reliance on core sprays inwater reactors. A fast deluge flooding system for reactors has been mentioned by Dr. Thompson.

Based on studies underway, Dr. Ergen sees a reasonable solution to the core melting problem; although the seriousness of the problem has been recognized by both Oak Ridge National Laboratory (ORNL) and Brookhaven National Laboratory (BNL), it has not been publicized. The fusion of fission products in soil is the subject of studies relating to the re-entry of satellite materials; projecting these results indicate that although the fission products in sand, or in limestone as under the Dresden site, would melt downward, the molten mass might be channeled so as not to violate any confinement seriously. Drilling holes below a reactor might assist in channeling; a venting and air cleaning system would make leak tightness of the confinement of less concern. Pressure resulting from such a molten mass would have to be relieved.

A difficulty in primary circuit integrity and fission product confinement is the tendency for future applicants to continue approved designs. Dr. McKee considers it very necessary for the ACRS to raise new safety issues; however, doing this in a separate letter, rather than in association with a particular reactor appears desirable, e.g., as with the pressure vessel problem. Standarization is probably the trend with reactors, which leaves the General Electric Co. (GE) and the Westinghouse Co. engineers in a quandry as new problems are raised with their accepted designs. The reluctance of the auto industry to respond to the smog control measures in California was recalled by Dr. McKee; complaints have been made for years but the 1968 models automobiles will be the first required to have appropriate exhaust provisions.

The slowness in response to the pressure vessel letter, e.g., by the Dresden Group, was observed by Dr. Hanauer as evidence of the reluctance of the industry to take steps on new safety problems. The rapidly moving technology of the nuclear industry may reveal other serious safety problems, according to Dr. Zabel; consequently, much judgment will be involved in safety assessments. A proper balance between safety and nuclear power needs is preferred by Mr. Etherington; to him alsolute safety is impossible. Dr. Okrent reported Commissioner Palfrey had commented against any steps to neglect safety because of the pressure of economic competition.

Difficulty in demonstrating the usefulness of a molten core collecting facility was predicted by Dr. Mangelsdorf; any requirement for a demonstration could stop the construction of large reactors. To him disproving the usefulness of proposed core melting arrangements would probably be easier. Nevertheless, Dr. Okrent predicted that within two years a design of a reactor facility to retain any melted core should be possible. The GE statement that the core melting problem is industry wide was recalled by Dr. Mangelsdorf; pursuit of a solution through the AEC might be the quickest route, but until this is accomplished, the best technology available, e.g., core cooling arrangements, would have to be accepted. The differences in opinion of the GE and Westinghouse groups on the usefulness of a molten core retainer was noted by Dr. Hanauer. Some believe that the Indian Point II reactor plans for a device to collect any molten core necessitates the same improvement in the Dresden II Reactor. Gas collection tanks might also be required for the Dresden II facility. If no such core melting requirements are set for the Dresden III Reactor, the Committee may find it difficult to enforce such steps on any future facility. Dr. Mangelsdorf desired to avoid any present ACRS conclusion on core melting which would implicitly affect existing reactors.

A threshold, e.g., 750 or a 1000 MW (e), for reactors with acceptable core retention arrangements was suggested by Dr. Zabel. The Committee approval of the Dresden II Reactor without a core collecting arrangement was given by Dr. Ergen as the excuse for GE not proposing such facilities for the Dresden III design; however, others noted that the volume of gas to be handled might be too much for the Dresden III facility. The large volume available in the Westinghouse design for Indian Point II reactor which could accomodate released gases, appeared to Mr. Etherington to lead to the reliance on a core melting arrangement, while the GE group, with a lack of such volume with Dresden III, claims no solution is possible now.

Problems arising from a primary system rupture were reviewed by Dr. Okrent; multiple safeguards of the same type and near a reactor's core might all fail, and, according to him, credit should be given for only one system. On this basis, additional safeguards in an entirely different location and of different type would be required.

Dr. Palladino reported on the subcommittee meeting regarding the Indian Point II reactor, for which the Public Hearing will be on August 31; the subcommittee agreed with the RS that the calculations on the core retaining basin are on a tenuous basis. Nevertheless, Dr. Palladino sees the basin approach as promising. Substantial core cooling arrangement changes are now proposed for this Indian Point II Reactor.

Possible hydrogen explosion hazards for the Indian Point II Reactor are seen as serious by Dr. Mangelsdorf; controlled burning may be required. A maxim in the petroleum industry is that, if an explosive mixture can collect, an explosion will occur. In petroleum reformers, hydrogen is made continuously and steam is present, but the rapid flow and careful control of concentrations avoid explosive situations. Dr. Monson predicted no solution to the problem of maintaining the containment if the core should melt with a subsequent hydrogen explosion; hence, any approval of the Indian Point II Reactor must depend on the assurance of core cooling.

Comments of Dr. Thompson dated 4 August 1966 on the Indian Point II reactor core melting problem were reviewed; no ACRS recommendations on the hazards of melting appeared to him warranted now. Dr. Zabel summarized the ACRS view; since no solution seems possible, only core cooling should be stressed.

After a later session with the applicant, on the Indian Point II Reactor, Dr. Palladino said he was more relaxed on the hazards. Avoiding melting of the core did not seem so difficult to Dr. Monson; more water and a larger supply of emergency diesel power might assure cooling. The use of solid reactivity control materials containing boron, rather than water solutions, was seen by Dr. Monson as always maintaining a negative moderator temperature reactivity coefficient. Proposed drafts of a letter on this reactor were reviewed by Dr. Okrent.

The old criterion that containment, which is a static engineering safeguard, is more reliable than active measures such as core cooling was recalled by Dr. Okrent; to him, the proposals for reliable core cooling contradict this accepted point of view. On many occasions equipment has not operated satisfactory even after passing periodic tests. Nevertheless, Dr. Monson said that a high probability of service when needed must be an acceptable feature for engineered safeguards. Dr. Zabel agreed with Dr. Thompson's letter that the Dresden III and the Indian Point II Reactors are closely related and the letters on each should be considered together. Since any conclusion on the Dresden III Reactor may reflect back on the approved Dresden II Reactor Dr. Mangelsdorf also advised much caution in the letters. A letter on the Indian Point II Reactor at the next meeting was favored by the group.

Drafts of proposed letters on the Dresden III Reactor were reviewed by Dr. Okrent; one included an implicit requirement that modifications be made into regions below the reactor vessel to reduce the consequences of any melting of the core and possible damage to the containment. Either more engineered safeguards or a denial of construction was seen as a requirement of any letter on this Dresden III Reactor.

Engineering decisions in the core cooling procedures require much judgment, according to Dr. Okrent; improved quality control of the primary system is desired by him. Mr. Etherington preferred accepting the best available arrangements against core melting. Dr. Monson has more concern over a satisfactory system ultimately rather than relying on present proposals for retaining a molten core.

Dr. Ergen reiterated that assuming core melting with violation of the containment, but with a clean up system available, would result in only limited damage to the public does not have a firm basis. Possible reasons for requiring no measures against a core which has melted through the reactor vessel were summarized by Dr. Ergen to be: lack of such requirements on past reactors, the low probability, and acceptable consequences. Dr. Ergen saw it as impractical to stop the construction of reactors on the basis of no solution now to the core melting problem; he said that the worst time for an accident is when winds are low, but such occasions allow more time for evacuation measures, e.g., of the Chicago area near the Dresden site. The consensus was that the retention of the core in the pressure vessel must be the primary aim. Lesser returns from additional steps to cool the core cooling system must be sought. Nevertheless, identifying a design goal for cooling appeared difficult to Dr. Monson, and this is needed for the applicant before he can satisfy a Committee request. Another diesel engine appeared to Dr. O'Kelly as needed for the Dresden III design to give further reliability of the core cooling system. The large reactors proposed for the Quad cities and the Browns Ferry locations were observed by Dr. Hanauer as presenting even worse core melting problems; lesser mention of the pressure vessel problem of late was noted.

Dr. Palladino recalled evidence against the effectiveness of core spray systems, e.g., the SL-1 core spray did not operate in a test as planned and pipes to this reactor were sheared during the accident; in addition, a failure of the core spray system occurred with the Senn Reactor. The failure of small pipes, such as used in the core sprays, from normal construction operations or maintenance, e.g., workmen stepping on them, has often occurred.

The engineering safeguard arrangements for the Dresden III Reactor were considered by Dr. Mangelsdorf to be equivalent to the Indian Point II design; he noted that the GE proposal is to spray coolant on the top of the core, which might be more effective than the plan of Westinghouse to fill the core from the bottom. Unpredicted vibrations have given problems with other Westinghouse Reactors, e.g., the Yankee system. The failure of core emergency devices at the Senn Reactor was discovered inadvertently by pieces found elsewhere in the system.

Dr. Mangelsdorf sees much to be learned from a prototype of the core spray system. He doubts if a practical flooding system is yet available and he has little hope for a successful core retention basin arrangement. The Indian Point II Reactor, with no nozzles below the core, provides a different situation than Dresden III from a flooding point of view. Mr. Etherington observed that the desire for flooding is based on distrust of the spray system; to him a very rugged and reliable system could be designed. Therefore, strict requirements on this internal design would seem to assure the reliability.

The use of large tanks of water with rupture disks to allow reactor flooding in emergencies is proposed for the Boston Edison Reactor at the Edgar site, according to Dr. Okrent; perhaps, this massive flooding might be desirable for the Dresden III and the Dresden II designs.

Some felt that the detailed attention by the Westinghouse group to the questions of the Committee on engineered safeguards was taken by the Committee as too much of a credit for the Indian Point II design; in addition, the Dresden site is probably better than the Indian Point location. Mr. Etherington said that the Price Anderson act recognized the maximum credible accident; he sees little difference between the Indian Point II and the Dresden III designs, and will take no position on the Quad Cities Reactor.

The inclination of the RS not to require steps against a core melting accident, surprised Dr. Hanauer; to him ACRS disapproval of the effectiveness of a core retaining basin could be a basis for stopping construction. Because there is much evidence that no water might be available for core cooling with the present designs during an accident, Dr. Hanauer prefers careful attention to avoid this. The consensus of the Committee was that core sprays are probably not good enough now. Flooding of the system might be independent enough to result in Committee acceptability.

Mr. Case joined the Committee briefly near the end of the session on the Dresden III unit and said that after listening to the Dresden III applicant, the RS recognizes additional safety measures are needed for this reactor, e.g., more capacity for the core spray, with redundancy.

Regulatory Staff (RS)

Mr. Case said that the RS considers proposed core cooling systems reliable enough to prevent core melting and, therefore, melting of a core through a reactor vessel. The RS is giving no credit for the core retaining basin underneath the Indian Point II Reactor, because the concept has so many uncertainties. An RS analysis assumes that 25% of a core could melt to the bottom of the vessel without penetration, if cooling from the vessel outside were maintained. Dr. Hanauer predicted that a churning of a molten core mixture would remove any solid uranium dioxide layer which might melt the reactor vessel. Above a six-inch opening in the primary system, a high head pump would be needed to cool the Dresden III Reactor. Although the Indian Point II group has revealed in the public records the proposed core retention basin, by its filing of a license application amendment, the RS is not ready to make this a safety issue yet, e.g., by discussion at the Public Hearing.

The Dresden group has been told that unless both the Dresden II and the Dresden III Reactors are shutdown, should a shared safeguards system fail, redundancy in the equipment would be required; lately the Dresden group has agreed to provide independent safeguard equipment. This redundancy in emergency equipment to prevent core melting can be extended to other items, e.g., valves and instruments; some such equipment need operate only once, e.g., isolation valves. The earlier refusal of the RS to accept certain engineered safeguards now proposed for the Indian Point II and Dresden III design was noted by Dr. Okrent.

A RS analysis of the possible retention of a molten core in a pressure vessel such as with Indian Point II was described. This assumed the outside of the steel pressure vessel at 300 F. with the inside at 2000 F.; at this temperature the contacting uranium dioxide (UO_2) would be solid. Above this solid UO_2, a liquid phase would be at between 4000 to 5000 F. with boiling UO_2 on above. Transfer of heat downward determines the vessel temperature. If 50,000 BTU per hour per square foot downward is assumed, then the tolerable thicknesses of uranium dioxide solid for 4, 2, and 1% of the core power as decay heat are 3, 6, and 12 inches, respectively. The amount of uranium dioxide in the Indian Point II Reactor is 400 cubic feet; this would cover the bottom of the vessel to a depth of about 2 feet.

Mr. Case read a letter from Mr. Joslyn dated 29 July 1966 which agreed to providing two independent core cooling systems for Dresden II and for Dresden III; a review of the Dresden I Reactor in view of present licensing situation was promised. A summary of the system which protects the containment integrity for the Dresden III Reactor from core melting through loss of coolant accidents was reviewed by Mr. Case. This showed the size of break in the primary cooling circuit which could be accommodated by the several emergency systems. For example, the high pressure cooling injection system and core sprays could withstand the loss of coolant from a 5.5 square foot hole; mechanical reliability has been considered. The suppression pool is the source of spray water for the high pressure core injection system of the Dresden III reactor. Since two sets of core sprays inside the vessel must operate, these must not be damaged in any accident. With 9100 gallons per minute of coolant added, and an assumed 4% metal water reaction, ten per cent of the cladding might fail and one per cent of the fuel might melt. In response to Dr. Monson's question, 4500 gallons per minute would definitely flood the core. The diesels, which have other loads to accommodate, can supply 2000 gallons per minute. The Commonwealth Edison Co. was reported by Mr. Case to be concerned about the possibility of any new safety measures added to the Dresden III Reactor being required for the other nuclear plants; the continued addition of safety requirements is vexing GE and Westinghouse engineers because of economic competition. The RS is uncertain as to who in the GE group can commit the Company to further design changes.

Since no effort has been made by the Dresden III Group to analyze the consequences of a melted core, the emergency core coolant systems must be extremely reliable; nevertheless, Mr. Case did not see this system any different in dependability than that proposed for the Indian Point II reactor. A series of mistakes of accidents might lead to some uncertainity in the reliability of the diesel power for the Dresden III plant. At the Indian Point II reactor, three diesel engines are planned, and two are needed for the emergency system; the Dresden III plant now plans to rely on only one diesel and the outside electrical power for emergencies.

The French rely on operating steam boilers for emergency power. Electrical power loss has occurred at the Yankee Reactor and three times at the Carolina Virginia Reactor. Standards for emergency diesel engines and outside electrical power are being formulated by

Mr. DiNunno's group.

INDIAN POINT II REACTOR

Indian Point II Group

The contents of supplements No. 4 and 5 to the license application were reviewed. New calculations have been done by a computer code towards metal water reactions and fuel melting. The nuclear design indicates chemical poisoning as a reactivity control is satisfactory; solid control materials, e.g., boron carbide in the core as fixed units, may be used.

The capability of the vessel and the core structure to retain any slumped fuel during a loss of cooling accident was reviewed. The cracks in solid uranium dioxide, when above 2500 C., tends to close; because of this, little fuel fragmentation and loss is predicted following cladding failure. Water flooding of the reactor pit would penetrate the insulation around the pressure vessel, and a temperature of 300 F. is assumed at the outside of the vessel. Holes in the insulation under the pressure vessel are expected to assure passage of water and a steam phase around the vessel bottom. If 10% of the heat goes to the bottom of the vessel with the remainder flowing upward and recirculating in the covering water, the temperatures should be such as to leave 2½ inches of the pressure vessel steel wall undamaged. With this assumed 300 F. at the outside, other temperatures would be 2400 F. at the inside of the pressure vessel steel, 5000 F. at the top of a solid half inch layer of Uranium dioxide (UO_2), and over 8000 F. above a 3½ inch liquid layer of UO_2. The 2400 F. level is the temperature of the eutectic of the UO_2 and iron oxide mixture. The conductivies of materials have been varied in the analysis by a factor of two and the temperatures given are for the more conservative values. With more favorable heat conduction about five inches, or nearly all the thickness of the steel vessel, would be left.

The UO_2 molten mass should be retained in the vessel, according to this cooling model. If a molten chunk of fuel fell to the bottom of the vessel, it should be chilled by the wall to solid uranium dioxide with liquid on the top, Uranium dioxide, UO_2, gas would be formed which would tend to take the heat upward; about 20 cubic feet of UO_2 gas would produce about 40 cubic feet of steam. This steam should go through the hole which led to coolant loss and into the containment. A steam flow rate of about 30 to 40 feet per second has been assumed in the annular region between the insulation and the vessel wall; no possibility of stagnant steam around the vessel bottom is seen.

The core basin retaining arrangement underneath is seen as a reserve safeguard by the Westinghouse group; the pressure vessel, if it fell, could drop only 2½ feet because of concrete steps.

Dr. Hanauer postulated that fission products might be scrubbed from the melt by the steam and collect on the vessel head with subsequent decay heating; no analysis has been done for no core cooling and no heat transfer through the vessel top. Presumably the uranium dioxide and fission products could boil and collect above. The hydrogen following an accident could collect in the vessel container to give a large amount of explosive gas; the rate of hydrogen production has not yet been studied. Recombination of the oxygen and hydrogen has been considered, but no analysis done. Adding an inert atmosphere could limit the explosive hazard. Iodine removal by both sodium thiosulfate sprays and by charcoal is planned as redundant systems; the spray would be less effective for methyl iodide.

Accident conditions for a coolant line break assumed a negative moderator coefficient and only one low head pump of four operable. Shutdown by voids would occur in one second with the rods inserted in five seconds. Liquid would reach the top of the core in 5.4 seconds with the core uncovered in 7.3 seconds. The low head pump would start within 20 seconds with a deluge of water into the bottom reaching the lower core plate in 280 seconds; with two pumps operating, half the time would be required, and covering of the core should occur in 600 seconds. Only steam cooling was considered for the system, with no credit for any water flowing through the core. The parabolic rate law for the zirconium water reaction was assumed. With 12 to 15% of zirconium water reaction, which would not be too much different for either one or two pumps operating, 25 to 30% of the fuel would be exposed by clad melting. A spray to the top of the core would reduce this fuel exposure; no detailed analysis of the consequences has been done. With all four low head pumps operating, the total flow would be about twice; for a smaller break cooling conditions would be improved.

Acquisition of emergency equipment and installation requires about two years; consequently, space must be planned for such items early. The sensitivity to delay in starting the one low head pump has been also studied for delays of fifteen and forty seconds; these were found not to change the accident situation markedly. Siphon breakers in the coolant lines are believed to preclude any loss of water faster than it is being added.

For a 4 inch pipe rutpre with only one high head pump operating the rods would insert in two seconds, safety injection would begin in 50 seconds, the top of the core would be uncovered in 675 seconds. Uncovering of the core to five feet below the top would be in 850 seconds with recovery of coolant in 950 seconds. No cladding should melt for this situation, and the zirconium water reaction would be less than one percent.

Study of the literature indicates that failure of a turbine in a massive way would not give missiles which could penetrate the Indian Point II containment. Reference was made to a Commonwealth Edison accident which gave four large turbine pieces; two landed in the condenser and two were 150 yards apart and in opposite directions. These pieces were probably going at 450 feet per second when the turbine broke and 150 feet per second after penetrating the turbine casing. Ninety percent of the energy probably went into the casing of the turbine and the remainder into the kinetic energy of the metal peices. In view of this experience with the Commonwealth Edison failure, a twenty percent figure for kinetic energy of a turbine fragments was assumed with no credit for building retention. Reports of two turbines failing by over speeding when valves did not close were found, and failure is usually at 70 to 100% of over speed; both failures were early in turbine life. Foreign particles left in valves during installations and from inferior oil were the cause. No failure of a Westinghouse built turbine has been experienced.

The core structure contains about 220,000 pounds of UO_2 and 44,000 pounds of zirconium. The UO_2 volume, 400 cubic feet, could fill the vessel bottom part way; water would be around the vessel to a level of several feet above. Water inside the vessel would be 20 feet above the core at the bottom. Any crust on the molten uranium dioxide should be broken by vapor pressure. Dr. Zabel observed that solids in the circulating water of the emergency cooling arrangement would contain abrasive materials which might lead to difficulties during the 400 days of operation required following any accident.

The failure of radiographic methods to detect flaws in thick metal sections, which ultrasonic methods did locate, was mentioned by Dr. Okrent. The shear wave ultrasonic technique of testing is to be used for this reactor vessel; the forged stainless steel pipe will be tested ultrasonicly. During manufacture, the inspection of the vessel is to be over and above that required by Section 3 of the ASME code; careful inspection of the plates will be done. Pump casings will also be radiographed; the grain size on the pump casings preclude ultrasonic testing. Any welds found developing graphite will be removed and repaired. Much preventive maintenance is planned for this reactor plant. Careful surveillance and inspection of the pressure vessel on reactor shut down is planned; complete inspection of the vessel inside will be possible. The present program towards surveillance of pressure vessels by the Babcock & Wilsox Co. (B&W) is being sponsored by the Ensada group.

Release of steam through a relief valve of the pressurizer would amount to a leak of perhaps a quarter million pounds per hour. Dr. Monson explored the possibility of additional cooling with all the diesel power and the two low head and three high head pumps in operation to keep the core from melting; approximately 9100 gallons a minute of coolant would be supplied and ten percent of the clad melting was predicted by the Westinghouse group. If there were a negative reactivity temperature coefficient and 9500 gallons per minute were added, the group still expected five percent melting. The moderator temperature coefficient would be positive only in the first of a fuel cycle.

Dr. Monson proposed more capability for emergency cooling and assurance of a negative temperature coefficient to prevent melting or limit melting to a very small amount. The accident situation with two emergency diesels operating was predicted by the Indian Point group not to be much different than discussed above; however, no available diesel would result in a very serious situation. If no melting of the fuel is a goal, then the Westinghouse group believed that other measures than excessive cooling would be required, e.g., better distribution of the coolant.

The Indian Point group considers that the outside power is quite reliable for emergencies; nevertheless, the power loss of the Northeast of last fall did indicate otherwise. Four outside systems provide five power lines, and the other power unit at the site is a possible emergency source; this is considered better than at more isolated locations where perhaps only one long line may offer the emergency connections.

The tubes of the heat exchangers are to be ultransonically tested and welded into the tube sheet. Service water is used directly in the fan coolers. Pressure in the fan coolers tubes is to be five to ten psi; hydrostatic testing is done for 300 psi. Although the 40 psi integral leak test of the containment indirectly tests the fan coolers; individual leak testing on these coolers will be done. In addition, monitoring of the coils for radioactivity release would be a further leakage check. Severance of a fan cooler lines would lead to a leakage from the containment of 20% of the volume per day, 50 times the design valve, which could cause a 25 rem exposure at the site boundary in a half an hour. Fan coolers could be isolated remotely if need be.

If thirty percent of the fuel melted, the pressure vessel should hold, but the Indian Point II group was non-committal about retention of larger amounts. Fifty percent of the core gamma radiation was assumed to be absorbed in water to release hydrogen; in several weeks, the hydrogen concentration might be in the percent range in the containment. If the UO_2 remained in one mass, more of the gamma rays would be absorbed in it, with only perhaps a twenty-fifth as much available for water decomposition; then many months would be required to reach explosive levels of hydrogen in the containment. Much hydrogen might be dissolved in the water and be released later. The most favorable solubility data to promote an accident was assumed; the water was estimated at 180° F.

Dr. Okrent concluded that more emergency core cooling assurance seemed advisable, the role of the positive moderator coefficient will need more Committee consideration, and improvement in the quality of the primary system may be in order; a letter on this reactor is probable next week.

DRESDEN III REACTOR

Dresden III Group

A flow chart showed the core spray circuits, the high pressure coolant injection system (HPCI), the isolation condenser arrangement, and the containment cooling circuits.

Mr. Bray said that the design basis for the emergency core cooling arrangements is for no core melting, all breaks dual protected, no external power sources available, and testing of the equipment possible. The proposed systems include the two core sprays, the HPCI system, feed water flow, core flooding, and operator depressurization. A bar chart showed size of break (log arithm scale) for which the different emergency cooling systems would be effective. Bars for core sprays, feed water availability, HPCI, and operator depressurization were included. This was for a break below the water line with liquid flow from the primary system and for the other case of a break above the water line with a steam flow. The size of break went from about .02 square feet to 5 square feet. For the liquid flow situation, core sprays one and two were effective down to about 1.5 square feet or a six inch diameter pipe opening. While for a steam flow break, these were effective down to about .3 square foot. For the liquid flow loss, only the core sprays could satisfy the needs for a large break. The feedwater system, the HPCI and the operator activation of depressurization would be capable of handling a small break; these three cover a longer range for the steam loss accident. The two loops for the core sprays are to be completely redundant. The isolation condenser system is also a high pressure arrangement.

A chart showed the pressure of the system against time following loss of coolant. A steady drop from about a thousand psi to about 600 psi would occur with a more rapid drop following. This was for a break of .15 square feet with no feed water introduction. The core would be uncovered for about 500 seconds, with core sprays starting at about 600 seconds. The situations for intermediate size breaks of .01, .03, and .06 square feet were given. Percent of feed water flow varied from 3 to 50. The time to reach the high dry well pressure changed from 55 to 5 seconds. The time to uncover the core went up to six minutes.

The GE group has been conscious for some time of the serious problems of cooling of very large reactor cores. Many relative tests dating back some years have been reviewed. Increasing the capacity of the cooling system has been considered. Uncertainties in the behavior of molten materials in water exist. A continuing program to simulate blowdown accidents and develop reliability data is underway; steam binding tests are also to be done.

Core cooling is believed assured by a physical separation and mechanical protection of the components, component redundancy, quality control, mechanical integrity, control instrument logic, design for surveillance, and a very reliable design and operating analysis. All lines of the core cooling systems are to be adequately designed for expansion requirements. A six inch movement of the pressure vessel is believed possible before any line breaks, holes in the concrete structure, rather than pipe elasticity, determine this.

The core spray water would drain through a line break, into the dry well and then to the pressure suppression system; little chance for escape is seen. The possibility of vibrations deactivating a spray system as has occurred with other reactors was suggested by Dr. Hanauer. Perhaps spray tests, e.g., with the Jersey Central Reactor, could throw light on reliability. The Big Rock Reactor had vibrations which might have interferred with a core spray arrangement. Nevertheless, early tests with a new plan are believed by the GE group to capable of revealing such difficulties. Designs of the current GE plants are similar; but careful analyses and tests are used to include possible improvements towards safety. The General Electric (GE) group stated that all steps towards a safe plant are taken with each new design; evolution in research and development and plant experience provide improvements.

The use of one diesel for the Dresden III plant was considered by Dr. Hanauer as inconsistent with the duplication of other parts of the spray arrangement. Nevertheless, the GE group considers the diesel very reliable and sees outside power availability providing redundancy. Dr. Palladino indicated that the one diesel might need maintenance, which could require shutdown of the plant at other than refuelings. The emergency cooling system is considered by the GE engineers to be adequately powered. Simplicity is desired so as not to complicate the spray arrangement, e.g., care is to be taken to see that no loose bolts or other pieces can enter the system.

Recently a tornado did disrupt all of several outside power lines to the Dresden I facility; the diesel emergency unit did operate. Now additional lines of emergency power are from the south. Mr. Joslyn could not recall a case where a power plant accident resulted in the loss of outside power; with a turbine failure, there was still no outside power loss. Later, he recalled three shutdowns of plants from internal incidents with no loss of outside power; switch yard errors have resulted in such loss.

Post accident cooling of perhaps a month was mentioned; this would be the required time for a decision as to whether to unload the fuel of the plant. No estimate was given by GE as to how long the plant might have to be cooled if the fuel was not unloaded from the core.

Mr. Joslyn said that the conventional plants of the Commonwealth Edison system, as well as a nuclear plant, can be shutdown at any time the operator in charge deems advisable. The Dresden I experience with the tornado showed good operator action in an emergency. The Dresden I plant has been shutdown rapidly and safely with no plant damage. Flanged connections might lead to problems during fast shutdown. The Dresden III plant is designed for a drop of 545 F. to 307 F. in 10 minutes and then at a 100 F. per hour to a 100 F.; the ASME codes are the guide. If there were a more severe drop in temperature, e.g., the 545 to 350 F. in five minutes; the ASME code would not be met. Nevertheless, the fatigue criteria would probably allow ten such cycles. Data from thermcouples would be analyzed in these metallurgical predictions. Inspection would be performed particularly in the Flange areas, which would be accessible. Mr. DiNunno said that heating and cooling rates may be determined by a variety of components.

Feed water failure would trip the isolation condenser arrangement with a 15 second delay. The condenser capacity can take all the heat except that from a transient. Safety valves have a capacity of 40% of the reactors steam flow. Maximum pressure expected in a reactor is 1100 psi. The high pressure injection system is to back up this condenser arrangement. With no outside heat removal ability, there is a 25 megawatt-day heat sink in the pressure suppression system with use of the core sprays, which might be a five day supply.

If the control rod worth the most in reactivity were dropped from the reactor, a 4000 megawatt second transient might occur, only a one to 2 psi peak pressure is forseen in the vessel because of the cushioning of the steam above; somewhat less conservative assumptions might give 5 psi.

A test with a hot spray pipe about 20 feet long and in a pressure vessel has been made. Depressurization was carried out and water was supplied by a pump. Pressure traces of recorders against time were shown for the pump head, the vessel pressure, and the sparger spray pressure. Variation in the slope of the change in pressure of the sprays may be from boiling in the sparger. Although no actual blowdown test on any reactor has been done, many loops have been tested. The intent is for a full scale test of the spray system and higher temperature tests are planned.

The system proposed is believed adequate by GE to keep the core from melting; advanced concepts are under study, but the GE group is not inclined to speculate on any other schemes, e.g., use of massive flooding of the facility. Complete core flooding has been considered, but there are associated problems. Flooding systems encounter difficulties in predicting the flow passages. Water flooding yields steam from the bottom, and, when only two thirds of the core heighth is covered, sufficient cooling from both the steam and the water should be available; over the range tested, large variations in flow have not affected temperature much. Core flooding is not seen by the GE group as effective for cooling as is a spray. However, Dr. Okrent told the group that the present core cooling arrangements are probably insufficient; flooding might be an adequate addition.

Later the GE group said that the plan would be to use the core sprays for flooding; this would be either more sprays or minimizing the leaks, and perhaps some cladding melting would accompany this. Mr. Joslyn doubted if those changes in design could be done before the meeting of next week; however, he hoped that the flooding concept might be sufficient to allow Committee approval. Neither GE nor the Commonwealth Edison Group want a basin for retention of molten core; a plant that would require such measures would not be considered suitable by the utility. Core melting in the vessel. but no penetration, would be acceptable.

As an interesting historical aside, it is noted that the very first section of the minutes of the Special August meeting mentions that Dr. Beck was disturbed by an attempt by the ACRS to make private arrangements with British safety groups. It also mentions a meeting between ACRS Chairman Okrent and the AEC Commissioners. What became known much later, from a brief history of the ACRS prepared by the AEC Chief Historian, R. G. Hewlett, in 1974, was that during this very difficult period in which a course of action was sought by the ACRS concerning loss of cooling accidents and the "China Syndrome," Dr. Beck was interceding with the AEC Commissioners concering what he considered inappropriate activities by the ACRS. We quote from Hewlett as follows:

> With the rapid expansion of projects utilizing nuclear energy, the role of the ACRS became increasingly amorphous and expansive. This trend prompted a letter from Dr. Clifford Beck, Deputy Director of Regulation, dated July 19, 1966, to the files entitled "Current Trends in ACRS Activities." Beck argued that the present trend was similar to the situation in the late 1950's when the ACRS was heading toward a role independent of the AEC, with its own expanding staff, a proliferation of consultants, and direct lines of communication with applicants and others outside the agency.
>
> In particular, Beck attributed to these tendencies a lack of common basis of technical understanding between staff and the ACRS because of differing consultant sources, a needlessly increasing ACRS staff, and an increase in ACRS involvement, and compliance activities. Among other recommendations, Beck requested an ad hoc task force, similar to that established in 1954, made up of AEC and ACRS personnel, to examine the present and future relationships of the two organizations.
>
> Subsequently the Commission met with Dr. David Okrent (Chairman of the ACRS) on August 3, 1966 to discuss these "current trends." No solutions were proposed to the problems Beck had enumerated in his letter of July, 1966*, but the statutory guidelines on ACRS activities were under continual review. In November of 1967 an amendment to §182b of the Atomic Energy Act was circulated by the Director of Regulation, proposing a modification of the requirements for mandatory reviews and reports by the ACRS. The Chairman of the AEC, Glenn T. Seaborg, sent a final legislative package to the Bureau of the Budget on December 20, 1968. The expressed purpose of the new legislation was to make the statute flexible enough to permit the ACRS to omit its review if the Commission and the Committee agreed. This amendment would assist the ACRS in its deliberations as the standardization of designs increased.

*The existence of Beck's letter was not disclosed to Okrent nor were Beck's concerns.

> Dr. Zabel's* appointment as Chairman of the ACRS in 1968 seemed to herald a new era of cooperation between the AEC and the ACRS. Mutually satisfactory liaison procedures were established between the two bodies. In January, 1969 it was decided by the Commission that the Director of Regulation was to be responsible for resolving serious deficiencies in the Quality Assurance programs of applicants, not the ACRS.** This decision was part and parcel of the move to free the ACRS of responsibility for routine matters.

Also, of some interest is a Project Status Report (on the following page) prepared by an ACRS Staff engineer prior to the Special August, 1966 meeting. This report notes that the Regulatory Staff had already given notice of a construction permit held on August 31, 1966 despite the complex new issues raised by the "China Syndrome," despite lack of resolution of the matter as of August 3, 1966, and despite previous requests by the ACRS that the Regulatory Staff not notice such hearings until completion of the ACRS portion of the review.

At its 76th meeting, August 11-13, 1966, the ACRS completed action on both the Indian Point 2 and Dresden 3 construction permit reviews.*** As an integral part of this action, the Committee also decided to write and completed preparation of a general letter concerning problems of primary system integrity, the loss of coolant accident, and the possible consequences of failure to cool the core.

At the August meeting, General Electric had proposed for Dresden 3, and for Dresden 2 as well, two core spray systems and a flooding system, any of which would meet a no-clad melting criterion, and each of which was operable with emergency on-site power. The ACRS agreed that a letter could be written on Dresden 3 although the final design of the emergency cooling system would require additional review.

The ACRS reports dated August 16, 1966 on Dresden 3 and Indian Point 2 are duplicated on the following pages, as well as the draft general letter agreed to at the August meeting.

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* Chairman are selected by the ACRS members themselves, and not appointed.

** As a later chapter will discuss, inspection and quality assurance proved to be rather deficient during the next two years.

***The ACRS members present at the July meeting were D. Okrent, Chairman, S. Bush, H. Etherington, F. Gifford, S. Hanauer, H. Mangelsdorf, J. McKee, H. O. Monson, H. Newson, A. O'Kelly and N. J. Palladino. C. Zabel participated in the August meetings; H. Newson did not attend. Members H. Kouts and T. Thompson missed all the July and August meetings.

MCG - 8/3/66

Project: Indian Point 2

Status : Construction Permit Review - Letter Requested

Background: On December 6, 1965, Con Ed submitted the Preliminary Safety Analysis Report for Indian Point Nuclear Generating Unit No. 2 along with the application for a construction permit for the facility. Since that time, five supplements to the Preliminary Safety Analysis Report have been submitted. Subcommittee meetings regarding the project were held on March 30, May 3, and June 23, 1966, and the project was considered at the April, May and July ACRS meetings.

DRL Analysis: DRL Report No. 3, which was issued prior to the July ACRS Meeting, concluded that the Indian Point 2 facility could be built and operated without undue risk to the health and safety of the public. Report No. 4 has now been issued by DRL. In this Report, DRL concludes that the proposed emergency core cooling system for Indian Point 2 provides sufficient capacity, redundancy and reliability to preclude significant core damage and to protect containment integrity in the event of credible loss-of-coolant accidents. DRL also states they believe the reactor pit crucible should be considered as only a backup to the emergency core cooling system and that sufficient evidence has not been presented by the applicant to demonstrate its effectiveness under assumed accident conditions.

Questions:

1. In their Fourth Supplement, Con Ed has presented additional information concerning radiolytic decomposition of water following a primary system piping failure. This matter was first raised by Dr. Parker of ORNL. DRL is in the process of obtaining Dr. Parker's opinion concerning the information presented by Con Ed. DRL might be asked their conclusion concerning the significance of the possible radiolytic decomposition of water following a loss of primary system integrity.

2. The Fourth Supplement contains information regarding the modified isolation valve seal water system. Conspicuously absent from this information is a discussion of the operating experience with the type of valves proposed for installation. Con Ed might be asked concerning this matter.

3. On page 6 of DRL Report No. 4, DRL indicates that they are willing to consider that two independent systems have been provided to prevent core melting following a small piping break -- the high-head and the low-head safety injection systems. It appears questionable that the low-head system alone would prevent core melting in the event of the worst size small pipe break. DRL states that Con Ed will be prepared to discuss this item at the Committee meeting.

4. DRL has already given public notice that the hearing to consider the issuance of a construction permit will be held on August 31, 1966. It is quite unusual for DRL to give such a notice prior to issuance of the ACRS letter. The Committee has previously asked the Regulatory Staff to terminate such practice, and they indicated that they would do so. DRL might be asked the reason for the notice being issued prior to the ACRS letter.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

August 16,1966

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON DRESDEN NUCLEAR POWER STATION - UNIT 3

Dear Dr. Seaborg:

At its seventy-fourth meeting, on June 8-11, seventy-fifth meeting, on July 14-16, a special meeting on August 4-5, and its seventy-sixth meeting on August 11-13, 1966, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Commonwealth Edison Company to construct a third nuclear power plant at the Dresden site, near Morris, Illinois. Unit 3 will include a boiling water reactor to be operated at 2255 MW(t) power level with pressure suppression containment. Unit 3 would be similar to Unit 2. The Committee had the benefit of discussions with representatives of the applicant, the General Electric Company, Sargent & Lundy, the Babcock & Wilcox Company, and the AEC Staff, and of the documents listed. A Subcommittee of the ACRS met to review this project at the Dresden site on June 2, 1966, and in Washington on July 7, 1966.

In its report on Dresden Unit 2, dated November 24, 1965, the Committee recommended that the AEC Staff follow development work by GE to resolve particular design problems. The Committee recommends that the Staff continue to follow the development work in connection with both Units 2 and 3, particularly with respect to operation with jet pumps, testing of emergency cooling methods, and studies of reactivity transients to assure no impairment of emergency cooling effectiveness as a consequence thereof.

The Committee also urged that the designers pay particular attention to the design of the pressure vessel, and of the high pressure steam lines with their isolation valves and fittings. The Committee reiterates its opinion on this matter in connection with Unit 3.

The Committee notes that the applicant has made improvements in the requirements for pressure vessel inspection during fabrication and urges that the applicant pursue vigorously the implementation of adequate in-service inspection techniques.

The effectiveness of emergency core cooling systems is a matter of particular importance in the unlikely event of a pipe rupture in the primary system. The applicant proposes the following improved complex of emergency cooling systems:

1. a high pressure coolant injection (HPCI) system,

2. a high-volume flooding system to permit rapid injection of water into the reactor vessel following blowdown to a low pressure,

3. two core spray systems,

4. a system that will make river water available to the feedwater pump for emergency cooling.

The applicant advised the Committee that equivalent changes in the emergency core cooling systems of the Dresden 2 unit would be made. Three diesel-driven generators will be installed to serve Units 2 and 3.

The Committee concurs that the proposed systems should increase the reliability and effectiveness of emergency core cooling. Complete details of the systems are not available, but the Committee believes that these matters can be resolved during construction of this facility. The Committee believes that the Regulatory Staff and the Committee should review details of design, fabrication procedures, plans for in-service inspection and the analyses pertaining to the emergency core cooling systems, as soon as this information is available and prior to irrevocable construction commitments pertaining thereto.

Careful examination of the forces during blowdown on various structural and functional members within the pressure vessel is necessary to assure sufficient conservatism in the design. The Committee recommends that the AEC Staff satisfy itself fully in this respect.

The Committee believes that the combination of emergency cooling systems has a high probability of guarding against core meltdown in the unlikely accident involving rupture of a primary system pipe. In view of the present state of development of such emergency cooling systems, however, and since the cooling systems may be subject to certain low-probability interrelated modes of failure, the Committee believes that the already small probability of primary system rupture should be still further reduced by taking additional measures as noted below. The Committee would like to review the results of studies by the applicant in this connection, and the consequent proposals, as soon as these are available.

1. Design and fabrication techniques for the entire primary system should be reviewed thoroughly to assure adequate conservatism throughout and to make full use of practical, existing inspection techniques which can provide still greater assurance of highest quality.

2. Great attention should be given to design for in-service inspection possibilities and the detection of incipient problems in the entire primary system during reactor operation. Methods of leak detection should be employed which provide a maximum of protection against serious incidents.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactor can be constructed at the Dresden site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

David Okrent
Chairman

References Attached.

References (Dresden 3)

1. Dresden Nuclear Power Station, Unit 3, Plant Design and Analysis Report, Volumes I and II, Commonwealth Edison Company, undated, received February 14, 1966.

2. Letter dated May 3, 1966 from L. F. Lischer, Commonwealth Edison, to Dr. R. L. Doan, AEC, transmitting Amendment No. 1, Answers to AEC Questions, including replacement and additional pages to Plant Design and Analysis Report.

3. Amendment No. 2, Answers to AEC Questions, undated, received May 20, 1966.

4. Letter dated May 26, 1966 from Murray Joslin, Commonwealth Edison, to Dr. R. L. Doan, AEC, transmitting Amendment No. 3, Answers to AEC Questions, including replacement pages to Plant Design and Analysis Report.

5. Supplement to Construction Permit Application, Reactor Vessel Non-Destructive Testing of Plate, dated June 10, 1966.

6. Letter dated July 8, 1966 from Murray Joslin, Commonwealth Edison, to Dr. R. L. Doan, AEC, with attachments.

7. Letter dated July 26, 1966 from F. A. Hollenbach, General Electric, to Mr. E. Case, AEC, with attachment.

8. Letter dated July 29, 1966 from Murray Joslin, Commonwealth Edison, to Mr. Edson Case, AEC.

9. Letter dated August 10, 1966 from W. D. Gilbert, General Electric, to Mr. E. Case, AEC, with attachment.

10. Letter dated August 12, 1966 from M. Joslin, Commonwealth Edison, to Dr. Richard L. Doan, AEC.

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1. Ltr dtd 5/12/66 from DRL to Commonwealth Edison Co.
2. DRL Staff Analysis dtd 5/26/66 (OUO).
3. N. M. Newmark and W. J. Hall comments dtd June 1966.
4. N. M. Newmark and W. J. Hall comments dtd July 1966.
5. DRL Staff Analysis "Report No. 2" dtd 8/3/66.
6. DRL - Dresden Unit 3 - "Systems which Protect Containment Integrity from Core Melt through Due to Loss of Coolant Accidents", received 8/5/66.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

August 16, 1966

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its seventy-fifth meeting, July 14-16, 1966, and its special meeting on August 4-5, 1966, the Advisory Committee on Reactor Safeguards completed its review of the application of Consolidated Edison Company of New York, Inc. for authorization to construct Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the seventy-second and seventy-third meetings of the Committee, and at Subcommittee meetings on March 30, May 3, and June 23, 1966. During its review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants and with representatives of the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed.

The Indian Point 2 plant is to be a pressurized water reactor system utilizing a core fueled with slightly enriched uranium dioxide pellets contained in Zircaloy fuel rods; it is to be controlled by a combination of rod cluster-type control rods and boron dissolved in the primary coolant system. The plant is rated at 2758 MW(t); the gross electrical output is estimated to be 916 MW(e). Although the turbine has an additional calculated gross capacity of about 10%, the applicant has stated that there are no plans for power stretch in this plant.

The Indian Point 2 facility is the largest reactor that has been considered for licensing to date. Furthermore, it will be located in a region of relatively high population density. For these reasons, particular attention has been given to improving and supplementing the protective features previously provided in other plants of this type.

The proposed design has a reinforced concrete containment with an internal steel liner which is provided with facilities for pressurization of weld areas to reduce the possibility of leakage in these areas. The containment design also includes an internal recirculation

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containment spray system and an air recirculation system consisting of five air handling units to provide long-term cooling of the containment without having to pump radioactive liquids outside the containment in the event of an accident. Even though the applicant anticipates negligible leakage from the containment, two independent means of iodine removal within the containment have been provided. These are an air filtration system using activated charcoal filters, and a containment spray system which uses sodium thiosulfate in the spray water as a reagent to aid removal of elemental iodine.

The reactor vessel and various other components of the system are surrounded by concrete shielding which provides protection to the containment against missiles that might be generated if structural failure of such components were to occur during operation at pressure. This includes missile protection against the highly unlikely failure of the reactor vessel by longitudinal splitting or by various modes of circumferential cracking. The Committee favors such protection for large reactors in regions of relatively high population density.

The Indian Point 2 plant is provided with two safety injection systems for flooding the core with borated water in the event of a pipe rupture in the primary system. The emergency core cooling systems are of particular importance, and the ACRS believes that an increase in the flow capacity of these systems is needed; improvements of other characteristics such as pump discharge pressure may be appropriate. The forces imposed on various structural members within the pressure vessel during blowdown in a loss-of-coolant accident should be reviewed to assure adequate design conservatism. The Committee believes that these matters can be resolved during construction of these facilities. However, it believes that the AEC Regulatory Staff and the Committee should review the final design of the emergency core cooling systems and the pertinent structural members within the pressure vessel, prior to irrevocable commitments relative to construction of these items.

The applicant stated that, even if a significant fraction of the core were to melt during a loss-of-coolant accident, the melted portion would not penetrate the bottom of the reactor pressure vessel owing to contact of the vessel with water in the sump beneath it.

The applicant also proposes to install a backup to the emergency core cooling systems, in the form of a water-cooled refractory-lined stainless steel tank beneath the reactor pressure vessel. The Committee would like to be advised of design details and their theoretical and experimental bases when the design is completed.

In order to reduce still further the low probability of primary system rupture, the applicant should take the additional measures noted below. The Committee would like to review the results of studies made by the applicant in this connection, and consequent proposals, as soon as these are available.

1. Design and fabrication techniques for the entire primary system should be reviewed thoroughly to assure adequate conservatism throughout and to make full use of practical, existing inspection techniques which can provide still greater assurance of highest quality.

2. Great attention should be placed in design on in-service inspection possibilities and the detection of incipient trouble in the entire primary system during reactor operation. Methods of leak detection should be employed which provide a maximum of protection against serious incidents.

Attention should also be given to quality control aspects, as well as stress analysis evaluation, of the containment and its liner. The Committee recommends that these items be resolved between the AEC Regulatory Staff and the applicant as adequate information is developed.

The applicant has made studies of reactivity excursions resulting from the improbable event that structural failure leads to expulsion of a control rod from the core. Such transients should be limited by design and operation so that they cannot result in gross primary-system rupture or disruption of the core, which could impair the effectiveness of emergency core cooling. The reactivity transient problem is complicated by the existence of sizeable positive reactivity effects associated with voiding the borated coolant water, particularly early in core life. In addition, the course of the transients is sensitive to various parameters, some of which remain to be fixed during the final design. Westinghouse representatives reported that the magnitude of such reactivity transients could be reduced by installation of solid burnable poisons in the core to permit reduction of the soluble boron content of the moderator, thereby reducing the positive moderator coefficient. The Committee agrees with the applicant's plans to be prepared to install the burnable poison if necessary. The Committee wishes to review the question of reactivity transients as soon as the core design is set.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactor can be constructed at the Indian Point site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
David Okrent
Chairman

References:

1. Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, Preliminary Safety Analysis Report, Volume 1, and Volume 2, Parts A & B, received December 7, 1965.
2. First Supplement to Preliminary Safety Analysis Report, dated March 31, 1966.
3. Second Supplement to Preliminary Safety Analysis Report, received June 2, 1966.
4. Errata Sheets for Preliminary Safety Analysis Report and First Supplement thereto, received June 13, 1966.
5. Third Supplement to Preliminary Safety Analysis Report, received June 22, 1966.
6. Fourth Supplement to Preliminary Safety Analysis Report, received July 28, 1966.
7. Fifth Supplement to Preliminary Safety Analysis Report, received July 28, 1966.

August 16, 1966

To : ACRS Members

(Signed) R. F. Fraley

From : R. F. Fraley, Executive Secretary
Advisory Committee on Reactor Safeguards

Subject: PROPOSED ACRS REPORT ON PROBLEMS ARISING FROM
PRIMARY SYSTEM RUPTURE, DRAFT #7

At the 76th ACRS Meeting it was concluded that the attached draft letter would be discussed with H. L. Price and the Commissioners at the 77th ACRS Meeting.

A copy of the draft is attached for your information. A copy will also be provided to Mr. Price for his use and use by the Commissioners in preparation for this meeting.

Attached:

Proposed ACRS Letter on Problems Arising from Primary System Rupture, Draft 7, dtd 8/16/66.

Copy:

Dr. D. Duffey

DRAFT 7
RCM/RJP/DO:js
8/16/66

Subject: PROBLEMS ARISING FROM PRIMARY SYSTEM RUPTURE

Dear Dr. Seaborg:

In its continuing review of the safety of large nuclear power reactors, the Advisory Committee on Reactor Safeguards has paid considerable attention in recent months to the general problem of a primary system pipe rupture (or other cause of significant primary coolant leakage) followed by a functional failure of the emergency core cooling system. The probability of either of these types of failure occurring is considered to be very low, and the chance that both types of failure might occur within the same event is even more remote. Nevertheless, as more and more reactors come into existence, particularly reactors of larger size and higher power density, the consequences of failure of emergency core cooling systems take on increased importance. Accordingly, the Committee believes that action should be taken along the lines indicated below.

1. Additional precautions should be taken to reduce still further the low probabilities of failure of the primary coolant system and the emergency core cooling systems.

a. In the primary coolant system, careful and systematic analysis of all elements of the system should be made from the standpoint of possible contribution to leakage or rupture. All elements, active and passive, should be designed, fabricated, installed and tested with such care as to assure an operational system not marginal in any respect related to reliability.

Deliberate surveys of new technological developments which might enable greater assurance of quality in fabrication and installation should be conducted, and use of such new or additional techniques as are appropriate should be instituted promptly. As one example, for pressure vessel welds on which ultrasonic inspection techniques can be used effectively to supplement radiographic inspection, both should be used.

Current practice for assessing probable operability and reliability of the primary system in service should be reviewed, and design features incorporated to improve inspectability and testability. Efforts should be increased to develop improved means of leak detection and to assure appropriate operator response to signs of leaks. Operating procedures in the circumstance of incipient difficulty should be especially conservative.

b. Added conservatism should be used in evaluating possible effects of a primary system rupture with respect to rendering the core unamenable to emergency core cooling. Such effects should be recognized early during the design stage and be adequately protected against. For example, all vital structures within the reactor pressure vessel must be conservatively analyzed and designed to withstand the maximum forces resulting from blowdown associated with a major rupture. ~~Two or more independent systems should be~~

c. For emergency core cooling, a very conservative design basis and high system quality are needed. Two or more independent systems should be provided.

Preferably, these should employ two different design approaches to provide maximum assurance that this engineered safeguard will function adequately under all conditions. Redundancy of all vital active components is required in each system, as are abundant capacity and adequate speed of response. Again, all elements should be designed, fabricated, installed, and inspected and tested with especial care, utilizing advanced techniques. The maximum practicable inspectability and testability during reactor operation should be provided. Experimental confirmation of the efficacy of this important safeguard under accident conditions should be pursued vigorously, in detail.

d. Appropriate design measures should be taken so that no possible mechanism exists by which a primary system rupture, through the effects of blowdown or by other means, might render the emergency core cooling system incapable of performing its function.

2. Design of water-cooled power reactors should develop in such a manner as to provide in the near future even greater certainty of adequate protection in the event of primary system rupture.

In the remote circumstance that failure of both the primary system and all emergency core cooling systems of a reactor were to occur, large portions of the reactor core could be expected to melt and the assurance of containment integrity would thereby be made substantially more difficult.

DRAFT 7 - Primary System Rupture 4

The ACRS believes that, with proper attention to the considerations of 1., above, a well-engineered set of emergency core cooling systems using the concepts employed in current applications for construction permits can exhibit very high probability of coping effectively with a primary system pipe rupture. Nevertheless, because experience with emergency cooling systems is limited, and because systems using current concepts necessarily are subject to certain low-probability modes of failure related to primary system rupture, the Committee believes it prudent to provide still greater protection of the public by some independent means, particularly for reactor sites nearer to population centers. Progress toward this objective will require an evolutionary process of design and a vigorous program of research, both of which should begin immediately and be aimed at reaching a high state of development in approximately two years. Future reactors relying solely on currently employed types of emergency core cooling systems to cope with the unlikely accident involving primary system rupture will be considered suitable only for rural or remote sites.

As discussed in the Committee's report to you dated November 24, 1965, the problem of providing adequate emergency core cooling is considerably complicated in the unlikely event of reactor pressure vessel rupture, and the Committee suggests that attention be given to this aspect of the core cooling problem.

DRR 7 - Primary System Rupture 5

The Committee believes that, in view of the large number of reactors to be built in the future, it is prudent to undertake the above measures to assure continued protection to the public, and recommends that they be implemented as rapidly as possible.

Sincerely yours,

David Okrent
Chairman

In its reports on Indian Point 2 and Dresden 3 the ACRS empahsized the need for additional measures to reduce the probability of a loss of coolant accident and the need for improved emergency core cooling systems. With regard to the ECCS, the ACRS took the unusual step of recommending that "the Regulatory Staff and the Committee should review details of design, fabrication procedures, plans for inservice inspection and the analyses pertaining to the emergency core cooling systems, as soon as this information is available and prior to irrevocable* construction commitments pertaining thereto." Prior to that time, construction permit approvals had been based largely on a commitment to meet rather general criteria, and the plant, as built, might or might not prove satisfactory to the regulatory groups. The requirement imposed by recommending ACRS review before irrevocable commitments were made, was, in a sense, the forerunner of requiring an increased knowledge of most design aspects at the construction permit stage.

The draft general letter, after recommending further measures to help prevent a LOCA and a very conservative approach to the design of ECCS, discussed the potential for large scale core melt. It went on to make a controversial recommendation as follows:

> ....because experience with emergency core cooling systems is limited, and because systems using current concepts necessarily are subject to certain low-probability modes of failure related to primary system rupture, the Committee believes it prudent to provide still greater protection of the public by some independent means, particularly for reactor sites nearer to population centers. Progress toward this objective will require an evolutionary process of design and a vigorous program of research, both of which should begin immediately and be aimed at reaching a high state of development in approximately two years. Future reactors relying solely on currently employed types of emergency core cooling systems to cope with the unlikely accident involving primary system rupture will be considered suitable only for rural or remote sites.

In connection with the issuance of the general letter on pressure vessels, the ACRS has received rather adverse comments from the Atomic Energy Commission to the effect that the AEC had not been consulted in advance and not been notified that the Committee planned to issue such a general letter. Hence, this time the Committee decided to forward the general letter to Mr. Price and to the Commissioners so that they could see it and comment prior to its formal issuance by the ACRS.

*emphasis added

The general letter was a way of asking that, as rapidly as possible, some alternate method, over and above adequate emergency core cooling systems of the general type being used or proposed, be developed, so that the problem of the "China Syndrome" would be attacked in depth with diverse approaches. And the plan to issue such a report was part of the overall package agreed to by the ACRS at its August, 1966 meeting. In fact, the basis by which some members agreed to the issuance of letters favorable to the construction permits of Dresden 3 and Indian Point 2 was that there would be such a general letter.

As was then the custom, all the discussions taking place between the ACRS and the Regulatory Staff, or with Commonwealth Edison and with Consolidated Edison, had been in closed session. The only thing on the public record which indicated that some new provision for full-scale core melt had been considered in any way was the amendment to the Preliminary Safety Analysis Report submitted for Indian Point 2 in late June. In that amendment they proposed to put a core retention structure under the reactor vessel. It's interesting to take a look at the Public Safety Evaluation published for Dresden Nuclear Power Station Unit 3 on August 31, 1966 by the Regulatory Staff. There is no hint anywhere in this report that the "China Syndrome" and the inability of the containment to withstand core melt had been a major issue. It is stated that the ACRS met with the Applicant on June 10th to discuss the overall design of the station and particular features of safety significance, that the ACRS met with the Applicant and the Regulatory Staff on August 5 and August 12 for further discussion on the emergency core cooling systems for the proposed plant, and that following this meeting, the ACRS reported its views by a letter report dated August 16, 1966.

Depending on one's point of view, one might consider this a less than candid review of what had transpired. One might equally well say that the ACRS letter was not completely candid, since it did not directly address the interrelationship between core meltdown and the containment failure in Dresden 3 report. However, that was a point which was to be made explicit in the general letter on Problems Arising from Primary System Rupture, which report was never issued, as we shall now discuss.

The 77th meeting of the ACRS was held on September 8-10, 1966. Members Thompson and Kouts, who had not attended the July and August meeting, were present at this meeting, as was a new member, J. Hendrie. The minutes show that there was a meeting between the Committee and the full Atomic Energy Commission, which included Chairman Seaborg and Commissioners Johnson, Nabrit, Ramey, and Tape.

Chairman Seaborg referred to the proposed letter on primary system failure. He said that the impacts on the industry might be serious, and he felt that any letter should await more study. To Dr. Seaborg, the letter failed to recognize the current efforts to meet this problem with large reactors. Mr. Price said that the core melting problem is one for the industry to pursue rather than for the Regulatory Staff. Dr. Kavanaugh, the Assistant General Manager for Reactors in the AEC, objected to the tone of the letter and, in particular, to its lack of recognition of the efforts under way. He saw the close relation of the time schedule for reactor designs and the site problems as a difficulty with any steps toward protection against primary system failure. Making public such a letter without the much related correspondence might lead to misunderstanding by the public. Commissioner Johnson noted that parts of the proposed letter might be considered decision-making, which is the prerogative of the Commissioners.

To Commissioner Johnson, more facts are needed before outlining measures to avoid or cope with this primary system hazard. Assemblying a task force group of experts from the AEC laboratory to assist the ACRS with this problem has been considered by Dr. Kavanaugh; no AEC conclusion on such an approach had been reached. After more discussion, a task force to develop the problem of primary system failure was again suggested by Chairman Seaborg, prior to any such letter from the Committee. Mr. Shaw of the AEC reported being impressed by the response of applicants to suggestions of the Committee and the Regulatory Staff; consequently he preferred no letter now.

After its discussions with the Commission and the members of the AEC staff, the Committee discussed at length in executive session the draft letter of August 16, 1966 concerning primary system rupture. Where agreement had existed in August, divided opinion was now present, and the presence of several members who had not been there in August added to the division of opinion.

Although several members felt that they had agreed to the letters issued on Indian Point 2 and Dresden 3 only on the basis that a general letter of the type under discussion would also be issued and, in fact, this exact statement is attributed to one member in the minutes, the final conclusion of the Committee was not to send the general letter, but rather to endorse the recommendation of the Atomic Energy Commission that a task force be established. And a possible major change in safety for light water reactors, namely the conscious development and provision of steps to mitigate core meltdown, was not undertaken.

The complexity of the problem is well illustrated by the minutes of the ACRS Executive Session on this matter, which follow:

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PRIMARY SYSTEM RUPTURE

Executive Session

A draft of 16 August 1966 concerning primary system rupture was referred to by Dr. Palladino who cautioned the Committee that any measures required should be consistant with past ACRS action. Dr. Hanauer noted that the Indian Point II Reactor probably did not meet the requirements of the proposed letter. The letter was considered by Dr. Ergen to be too inclusive; more attention to the known problems, e.g., core melting and possible hydrogen explosions, and defining the problems might be more productive. However, Dr. Zabel considered current reactors as forcing the Committee to face the issue.

Dr. Hanauer observed that emergency precautions must include equipment which will not fail during the extended operation required following an accident. The Committee's position appeared to be that the past emergency measures were satisfactory for rural reactors, but something additional is needed for the suburban locations. Dr. Bush admonished the Committee to use much care in identifying emergency systems as separate types. Dr. Monson believes that the independency of engineering safeguards required by the Committee means physical independence rather than different types of systems. Simultaneous failure of emergency equipment, e.g., of both core sprays and the flooding arrangements, appears possible to Dr. Okrent; this would decrease the degree of independence of such systems. Although Dr. Kouts presumed that any advise of the Committee towards this primary system problem would be made public; Dr. Okrent believes special requests to the AEC would be necessary to assure this.

According to Dr. Hanauer, research is still needed to see if the core would melt through the base of the reactor structure. Although one can assume a heat source and predict core melting, little is probably known of the consequences. Dr. Zabel considered two years as probably a short time for any answers to these questions. Dr. Okrent senses that Dr. Kavannaugh plans no special efforts towards the core melting problem now. The necessity for all the considered measures of emergency action, e.g., multiple methods of core cooling, is not clear to Dr. Zabel; hence he advised the Committee to hedge on stating any primary system requirements. Dr. Palladino recalled that the only core spray tested in a reactor, in the SL-1, did not work as designed, and the only spray inspected carefully was shown to have broken; since the basis of approval for several recent reactors has been reliable core sprays, he sees a need to collect more information before proceeding.

Dr. Kouts believed that if any one engineering safeguard was considered by the Committee as offering complete protection, then the industry would sieze on this and contend that no other safeguard is needed. Because progress towards more safety in reactors has been made by the industry in the past following Committee comments, Dr. Zabel believes that a letter may not be needed. The measures by the Indian Point group towards flooding the reactor, after offering some reluctance to such steps earlier, were seen by

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Dr. Palladino as evidence of the resourcefulness of the industry and a tendency to take safety steps following informal comments of the Committee. Nevertheless, Dr. Hanauer saw the pressure vessel letter of last fall as a parallel case which had been quite productive of industrial reactor safety efforts. Specific request for research by the industry rather than vague comments seemed desirable.

Dr. Thompson, saw little in the proposed letter on primary system failure except the requirement of two independent safeguards; the problems of core melting are believed to have been known by the industry for eight to ten years, and no solution has been forthcoming. Dr. Thompson advised a clear letter, if any. Dr. Hendrie believed that if the Committee has a position then the problem is only one of expression. Dr. Gifford advised the Committee that it regulated reactors and does not design them; however, Dr. Hendrie believed designers must have some specific information of ACRS requirements otherwise satisfying the Committee will be impossible. Bringing the primary system problem to the attention of the AEC informaly appears desirable. Dr. Okrent said that the draft had been shown to the Commission and discussed with them, which amounts to informal advice. There was divided opinion in the Committee regarding the advisability of having shown the letter draft to the Commissioners; Dr. Mangelsdorf and Dr. Kouts considered this a mistake. Several of the Committee were against presenting any further drafts of the letters to the Commission for a review; only a final letter to the AEC seemed desirable, with only a invitation to the RS for editorial comment.

After the session with the Commissioners, Dr. Ergen observed that efforts towards complying with the proposed letter were underway; therefore the letter may not be needed. Mr. Plaine said that discussions towards such safety items need more exactness; he related that the methods of communication to the Commissioners are letters, advice to the applicants, and drafts which are not formerly issued. To him, all these avenues must be retained because of possible use with other issues.

Dr. Mangelsdorf considered any action by the Committee now towards primary systems to be in ignorance; the AEC task force effort appeared a desirable path, and action of the Committee should await this. Dr. Thompson agreed to this point of view; the principle fabricators of pressure vessels and reactors are aware of the problem and response has been evident. Any general letter now appeared to Dr. Thompson to offer a path for interveners in reactor cases. However several doubted if the industry would respond without any letter. Dr. Newson recalled the reluctance of the applicants to make changes for the Dresden II and III reactors and the Indian Point II system; to him this indicated a letter would be useful. Dr. Newson believed that complete flooding might be considered as a replacement for containment in view of the claim that this was an assurred method to prevent fission product release. Dr. Newson recalled that he had agreed to the letters on the Indian Point II, the Dresden II, and Dresden III Reactors on the basis that a general letter regarding primary circuit failure would follow.

Many ideas on core retention are available, but Mr. Etherington saw many uncertainties in the proposed scheme, e.g., the behavior of very hot materials in water. Dr. Thompson recalled the problem of core melting arose with the Fermi Reactor perhaps ten years ago; no solution was forthcoming. This history indicates to him that the Committee should be cautious in recommending any action which might interfere with the steps proposed by the AEC over the next few months. Perhaps the Committee should establish a task force itself, according to Dr. Okrent, but he recognized that the ACRS is not an operating organization. Dr. Newson said that a task force for such technical studies can be chosen so as the conclusion is predetermined. Dr. Zabel said that a task force approach could be arranged to relieve the Committee from any decision; the proposed off shore island location of a reactor, which was handled in this way, was seen by him as a parallel.

Dr. Hanauer proposed that since the Committee does not seem to have a unified position, acting now might be unwise and delay progress towards solution of this primary circuit hazard. Dr. Thompson suggested that a position was needed before any letter could be prepared; perhaps the lack of conservatism in reactor designs should be recognized. An analysis might indicate where failures are possible and then criteria could be developed to limit the consequences of circuit failure. Dr. Gifford saw the problem as centering on large reactors in population centers; this pressure from the public forces the Committee attention. Dr. Hanauer agreed that the issue was one that must be faced soon.

Although there was some feeling towards tabling the problem, this was not favored because of the departure of Dr. Thompson, Dr. Newson, and Dr. Kouts at this meeting. Because of the economic competition to sell nuclear reactors, Dr. Thompson saw the only time to speak on such an issue was with individual cases rather than through a general letter; he recalled that complete measures to retain any pressure vessel failure would have probably been taken for the Indian Point II Reactor if there had been a Committee request. A reputation of capriciousness, or resistance to reactor construction, by the Committee was predicted by Dr. Mangelsdorf, if prudent action is not taken; the Committee's position and reputation with the industry could be undermined. Dr. Kouts believed that if the Committee lost its technical respect then the usefulness of the review procedure would be gone. Dr. Mangelsdorf considers the Committee to have lost ground by its actions with the proposed letter.

The severe competetion in the field, now about 2 billion dollars of reactors are on order, means that the reactor producers are risking much in projections on performance; any feature of a reactor which has had a safety review is assumed by the reactor builders to be satisfactory for future reactors. Dr. Thompson believed that the industry has been confused by the Committee actions, in particular by the pressure vessel letter and the subsequent approval of the Indian Point II facility. To him, the industry is seriously interested in making reactors safe and are taking steps beyond what the RS has required in the past. Dr. Thompson recalled that one or two of the Commissioners had expressed a desire for reactors to be in cities; now, seems to him, to be a time for consistency of the Committee's actions. Dr. Thompson believed the Committee has a good reputation which it should be very careful to retain.

Dr. Mangelsdorf saw no accusation of Committee capriciousness yet, but he saw this as developing if the Committee was not careful; those unfavorable to the Committee are well organized. Dr. Ergen recalled that the Oak Ridge group considered the Committee's actions on the High Flux Isotope Reactor as being difficult to understand; basic principles seemed to have been changed during the safety review. Development of an atmosphere of resistance to Committee pressure might result from the proposed letters; nevertheless, several admitted that ACRS criticism must be expected from any agitation of the Industry. Dr. Thompson said he had been told by the AEC staff that the Committee's recent actions on primary circuit failure have not been well founded. He believed the industry had been left in a quandary by the Dresden II letter and the pressure vessel letter; the chain of Committee actions did not seem to be consistent. Dr. Thompson recalled that the Committee might be considered to have forced the Westinghouse group to provide a core retention basin under the Indian Point II reactor, and then to have asked for proof of operation.

Dr. Gifford foresaw continuing new safety problems with reactors, and criticism from the industry and the AEC staff is to be expected. He saw a danger with the Committee spending so much time on introspection following pressure by the industry rather than applying its efforts towards safety. Dr. Bush believed several months delay would be necessary to formalize any Committee thoughts on this problem; he considers the new developments and Committee requests as leaving the industry uncertain as to what is needed, which necessitates a consistent Committee position. Mr. Etherington believes that some are willing to accept the small risk of serious consequences from a large accident, while others are unwilling to compromise in this way. Mr. Etherington expressed a need for guidance as to responsibility of the Committee, particularly as to any distinction between a fatality a day over a long period of time as contrasted to an equivalent loss from one accident. He considers the primary circuit issue clear to the applicants. To him, many questions raised are unnecessary and their avoidance would save time. Nevertheless, Dr. Okrent recommended more firm Committee procedures with a position to avoid a repetition of any unsafe designs. Dr. Okrent considered the past pressure vessel letter and the comments towards the core melting accident as having markedly affected the safety of reactors.

Dr. Zabel desired more discussion towards the list of required equipment for reactors (hardware table) before such a letter; Dr. Thompson reccommended no general letter for at least six months. An earlier Committee letter which was against reactors in the cities without many more safety precautions was recalled by Dr. Okrent; but Dr. Thompson believed these comments were towards reactors truly in cities, e.g., the Ravenswood or the Boston Edison Reactor, and not towards suburban sites. Past efforts by the Committee regarding reactors in cities was recounted by Dr. Kouts; a particular difficulty was defining a metropolitan area. An answer to the question of the threshold number of individuals which might be killed before special precautions are needed is desired by Dr. Gifford.

Following a motion by Dr. Kouts and Dr. Palladino, it was agreed to

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draft a letter to the General Manager or the Commissioners recommending a task force effort on the problems connected with the primary circuit failure accident, e.g., from melting of the core. Perhaps the General Manager route might avoid some publicity, but Mr. Plaine saw the Commissioners as probably a better channel. Dr. Mangelsdorf favored verbal communication to accomplish the safety measures towards the primary circuit failure only. The Quad Cities reactors appears to be the next large system to come before the Committee, which is a few months away; Dr. Mangelsdorf saw this allowing task force action, and a letter would not hasten the matter. However, Dr. Hanauer doubted if a task force could be effective within that time. Dr. Kouts noted the eight drafts with indecision, which is evidence of the lack of Committee opinion and indicates that the problems need more engineering information before an intelligent answer can be developed. Dr. Thompson saw continued discussion as promoting a reasonable conclusion; to him the nuclear industry is more persistent now in pursuing safer designs because of the huge investment. Dr. Thompson reiterated his preference for including safety advice on primary circuits into letters on a specific cases; this should promote use of the latest information on safety available.

Dr. Newson believed a fool proof core cooling system could be designed. Mr. Fraley saw quality control as the main concern to the RS. New cooling arrangements are of concern to both the RS and the safety research group but core melting is mainly the concern of the latter. Mr. Fraley sees the Commissioners as confused over this distribution of interest and efforts against primary circuit failure.

Dr. Zabel cautioned against unenforceable rules, e.g., the identification of sites, as surburban and rural sites, is difficult; items needing further study are the heating from fission products and the reactions between metals, oxides, and water. Dr. Hanauer sees much difference between the surburban and the rural sites. Dr. Zabel considers the informal actions of the Committee as much more important towards safe reactors than the written items. Dr. Newson reiterated that perhaps on core melting, the containment would be of no value. But Dr. Thompson believed that containment would always be needed because of possible fuel failure with rapid nobal gas release.

Dr. Palladino and Mr. Etherington agreed on the need for criteria for construction permits. Dr. Monson suggested restricting this to current items, e.g., acceptable core cooling arrangements. Dr. Bush and Dr. Mangelsdorf favored a table of emergency safeguards equipment for reactors, which, if used consistently, would assist the safety of the designs. Mr. Etherington believed that if a reactor, e.g., the Dresden III facility, is given Committee approval for construction without precautions against a core melting accident, this should not be raised again at the operating stage. To Dr. Thompson, this recalled the old problem of construction permits and operating permits which was considered by the Mitchell panel. Mr. Fraley said that the difficulty is being assurred of the applicant's following through with the proposed construction plans.

Dr. Hanauer judged that, since the Fermi Case, once a construction permit is granted the AEC is ultimately obliged to issue an operating license. Mr. Plaine said that the AEC had always recognized lack of complete information on a reactor design at the construction permit stage;

the AEC statute states that permits may be issued on the basis that satisfactory design will be developed. This has led to provisional construction permits and the precedent is well established. Dr. Hendrie also considered as established this internal policy for the Commission of accepting items at the construction permit stage, but with possibility of changes later. Dr. Kouts recalled Dr. Kavannaugh considering the core melting problem as one for reactor safety research, but other problems, e.g., the core cooling arrangements, was more one of design.

The reactor design cirteria subcommittee has been asked to report on the table of equipment required for emergency protection of primary circuits. Dr. Monson reported on a subcommittee luncheon meeting at which a preliminary listing of areas requiring criteria was prepared; these included the cooling system, turbine orientation, pressure vessel fabrication requirements, prestressed concrete structures. Dr. Thompson believed that the industry fears the requirement to alter old designs on the basis of late Committee demands; the basis is the abrupt change in reactor requirements implicit in the pressure vessel letter.

After being shown the proposed letter on primary circuit failure, Mr. Price joined the Committee briefly to say that, although he had no fault with the letter, he believed the Commissioners had already promised the task force arrangement to study of the problem; therefore, inclusion of ACRS comments in the summary letter recognizing the task force appeared more appropriate. This summary letter route would avoid public attention to the problem. Mr. Price was apprehensive over the public reaction to such an ACRS letter. Dr. Thompson motion that the Committee's comments on primary circuit failure be handled in this summary letter fashion was passed. Dr. Kouts referred to a 1962 summary letter, as well as others, as a precedent for this mode of Committee advice with no public attention.

Regulatory Staff (RS)

With Dr. Kavannaugh present, Mr. Price said that Dr. Morris was relieving Dr. Doan, who will stay with the RS for a while as a consultant. Mr. Price had no particular comments on the proposed draft of the primary circuit letter at the time; his questions concerned requirements for approved reactors and sites and additional requirements for future suburban and city sites. Mr. Price noted that few rural or remote reactors are being proposed. Mr. Price desires any letter to make it clear as to what is a rural or a remote site.

Dr. Doan said that the Virginia Electric Power Company Reactor will have about three miles of relative isolation, but perhaps 50,000 people will be at a somewhat further distance; this is an example of the difficulty in identifying a site as remote or rural. Mr. Price suggested circulating the proposed information on primary circuit requirements, and related other ACRS letters such as on pressure vessels and engineering safeguards, to the industry as a guide. To Mr. Price, two years is a short time for any research development program towards these difficulties.

We shall see in a succeeding section that there was an insignificant effort expended by the AEC (and NRC) in the decade following 1966 on safety research and development on core melt or on measures intended to ameliorate the consequences of core melt.

However, a major program on LOCA-ECCS and on primary system integrity was initiated as a result of the ACRS action in 1966. A new regulatory approach of great emphasis on preventing core melt, from any cause, rapidly evolved. And the MCA lost much of its meaning, although the prescription of Part 100 continued to be used in site evaluation.

The next month, at the October 1966 meeting, the ACRS wrote a safety research letter, in which it palced great emphasis on studying phenomena related to large molten core masses, and on an improved ECCS. However, the letter did not include a specific recommendation that some new engineered safety feature be developed for use a few years hence in light water reactors. About a year later the task force came out with a report the end result of which was to provide a mechanism whereby any further work aimed toward the development of a means to reduce the effects of core meltdown could be put aside as unnecessary by those who wished to argue against such efforts.

2.10 REACTOR SITING: 1966-68

Had the pressure vessel issue not arisen in 1965, it is likely that the Dresden 2, Brookwood, and Millstone 1 construction permit reviews would have adhered to emphasis on Part 100 in the manner previously developed in 1963-64, namely the substitution of engineered safeguards for distance. As a result of the pressure vessel letter on November 24, 1965, the question of how to implement the letter became part of the Brookwood and Millstone Point 1 reviews.

A difference in philosophic approach between the Regulatory Staff and the ACRS clearly emerged during the 69th ACRS meeting, January 6-8, 1966, when Harold Price, the Director of Regulation, said that if any applicant is forced to protect against pressure vessel failure, all other commercial reactors must comply, regardless of location. Clearly, the ACRS thinking differentiated among sites with regard to the possible requirements for such protection or other additional safety features (besides those dictated by the ritual of meeting Part 100). Then, the emergency of the "China Syndrome" problem and its resolution produced a revolutionary change in regulatory review practices, with an ever-increasing emphasis on measures to prevent core melt. As we shall see, it also made much less likely a favorable recommendation for a large LWR at a site substantially more populated than Indian Point. And, within ACRS, it built up an increasing emphasis on still greater measures to prevent core melt for reactors at "borderline" sites, like Indian Point, although the Indian Point 2 reactor itself was accepted. Indian Point 2 never seemed to be a strong candidate for rejection based on considerations of site characteristics, despite its relatively large surrounding population density and the fact it was about 25 miles from New York City.

From bits and pieces of discussion in the minutes of various ACRS meetings, one can deduce that the likelihood of a serious accident was estimated to be in the neighborhood of 10^{-5} per reactor year. In fact, ACRS member Etherington, back around 1965 or 1966, estimated the possibility of pressure vessel failure to be in the vicinity of 10^{-6} per vessel year, an estimate which was remarkably close to that published after considerable study some eight years later. In any event the thinking was that the probability of a serious accident was likely to be small.

The comparison of rural versus suburban versus metropolitan sites with regard to the risk imposed on society was complex. For relatively smaller radioactivity releases, the rural or remote site clearly had large advantages. For the postulated accident involving very large releases of radioactivity outside the containment, there was a considerable body of thinking (though not unanimous) that there might not be that big a difference between a rural site and a relatively more urban site, at least in terms of the total effects, even if the early mortalities were larger for the urban site. This thinking was without the benefit of the more elegant studies done six or eight years later for the Reactor Safety Study, but had the benefit of studies like those done by the Brookhaven group for WASH-740.

When faced with the clear identification that core melt would lead to containment failue for the Dresden 3 and Indian Point 2 reactors, the Regulatory Staff proposed to continue with approval of those two reactor designs on the basis originally advanced by the applicant. The ACRS on the other hand, chose to look very intensively at what improvements in engineered safeguards were possible, and sought to establish whether they could find improvements that would make these two reactors acceptable.

Looking back at the situation some dozen years later, it's not obvious why more time wasn't taken in trying to fully develop the new regulatory approach. Nevertheless, an approach was developed during the summer of 1966 which, in essence, created a major change in the engineered safety requirements for light water reactors, and really set light water reactor safety on a new path. It had become important to make the probability of core melt much lower than it was, whatever it had been; and the first two major steps, which were taken in connection with Indian Point 2 and Dresden 3, were: 1) to require much improved quality in the primary system, much more inspection and much more leak detection in order to reduce the probability of a loss of coolant accident; and 2) to require a much improved emergency core cooling system in order to reduce the probability that a LOCA would lead to core melting. This was the beginning of a continuing series of efforts, looking in ever-expanding directions for possible causes of initiating events that could lead to core melt, and seeking out measures to reduce the probability of such events. Pressure vessel failure had become one of many possible sources of containment failure, and it had to take its place with other possible sources of large radioactivity release.

During the months following the August 1966 decisions on Dresden 3 and Indian Point 2, the ACRS devoted considerable effort to reactor siting. At the 77th meeting in September 1966, the ACRS requested one of its subcommittees to prepare a draft of possible design requirement criteria for use in future construction permit reviews, and a first draft was discussed at the 78th meeting, October 6-8, 1966. At the same meeting the Committee also discussed at some length the matter of emergency evacuation capabilities in the event of a serious nuclear accident.

As member Joseph Hendrie put it in the discussion on design criteria, the ACRS was fundamentally enlarging the scope of credible accidents. And this was to show clearly in the next reviews. During the next few months Turkey Point Units 3 and 4, Quad Cities Units 1 and 2, and the Palisades reactor were all reviewed for construction permits. In addition to examinaing an improved ECCS, a much broader and deeper look at possible accident sources other than the LOCA was undertaken.

A special ACRS meeting was held December 2-3, 1966, particularly to provide time for discussion of reactor design criteria and similar matters. At this meeting the Committee adopted the following motion:

> That it be recognized that there are differences in reactor sites which justified differences in safeguards provisions. In view of this, for the Committees' use in delineating reactor sites, three types should be recognized:
>
> City - roughly characterized by Edgar and Ravenswood;
>
> Rural - roughly characterized by Dresden and San Onofre;
>
> Intermediate - roughly characterized by Indian Point and Millstone.

The Committee decided to try to develop design requirements as a function of site type for several safety concerns, including emergency power, tornado and hurricane protection, earthquake protection, decay heat removal, reactor scram, turbine orientation, primary system integrity, instrumentation, emergency core cooling systems, and containment requirements. Although the Committee never issued a report in which it recommended differences in design and engineered safeguards as a function of site types, the existence of this effort is, in itself, interesting. And the act of working on such possible requirements generated ideas as to different levels of safety, and led to a deeper examination of the adequacy of previously accepted measures.

At the special meeting in December 1966, the ACRS also discussed the slow pace at which improvements in pressure vessel quality were being initiated by industry code groups and adopted the following motion:

> The ACRS Chairman shall ask Mr. Price and Mr. Shaw whether members of their groups can and will work with one or more ACRS members to develop additional requirements for Section III prssure vessels; these are to be considered by the ACRS within two or three months.

The ACRS discussed pressure vessel inspectability* at the special December meeting and at the 80th meeting, December 8-10, 1966. At the latter meeting, the Commitee adopted the following position:

*Access for inspectability was a particularly awkward problem for BWR's.

With regard to the inspectability of reactor pressure vessels for pressurized and boiling water reactors, the following is the position of the ACRS.

1. The interior of the vessel, including the bottom head, should be accessible for general visual observation on a scheduled periodic basis. Such observation has as its objective detection of mechanical damage or structural failure of reactor internals.

2. Practical means of access should be provided to essentially 100% of a reactor pressure vessel surface, either from the inside or outside or a combination thereof; the purpose of this access is to permit thorough inspection of the vessel at appropriate intervals by visual means and ultrasonic or other suitable methods.

3. The ACRS realizes that it may take time to achieve these aims, but expects to see them fully achieved in plants for which construction permit applications are filed more than one year after announcement of this position.

4. The foregoing should be announced formally to the nuclear industry within the next several months. The target data for the announcement should be selected at the February, 1967 ACRS Meeting.

The two positions adopted on pressure vessels represented one aspect of the new emphasis on accident prevention.

On December 13, 1966 the Public Service Electric and Gas Company of New Jersey submitted an application for a construction permit and operating license for its proposed Burlington Nuclear Generating Station Unit #1. The proposed site was located on the east bank of the Delaware River estuary, adjacent to the city of Burlington, approximately 17 miles northeast of downtown Philadelphia and 11 miles southwest of Trenton, New Jersey. The population surrounding the Burlington site represented a significant increase compared to previously licensed sites for reactors. It was proposed to build a 3083 MWt PWR, which represented an increase of 12% over the Indian Point 2 reactor. The containment system and engineered safety systems were proposed to be basically those of Indian Point 2. The following table compares the surrounding population distribution for several reactor sites.

Distance in Miles	Burlington	Indian Point 2	Turkey Point	Oyster Creek
0-1	4,700	1.080	-0-	200
0-2	18,600	10,800	-0-	1,600
0-5	119,400	53,000	-0-	4,600
0-10	536,200	155,500	42,000	32,800
0-20	3,904,000		232,000	136,000
0-25	4,502,000	1,393,000		

The application for the Burlington reactor site, coupled with the knowledge that plans were under way for proposing a reactor for the Bolsa Island site off a heavily populated coastal area of California, made it clear that the pressure was still on from the industry and from the developmental side of the Atomic Energy Commission to move reactors into more populated areas.

At the 81st meeting, January 12-14, 1967, the ACRS held its first meeting on the proposed Browns Ferry BWR's, which at 3300 MWt had a power 46% higher than Quad Cities, and about five times larger than any operating BWR. So, the trend toward still larger reactors was continuing. And, while the Browns Ferry site was one having a low surrounding population density, experience told one that similar reactors would soon be proposed for much more heavily populated sites.

The Browns Ferry Review occupied a major portion of the 82nd meeting held February 8-11, 1967, a special meeting held February 28, 1967, and the 83rd meeting held March 9-11, 1967. At the March meeting, the ACRS finally arrived at a decision and wrote a letter report to Chairman Seaborg which included many Committee reservations* and dissenting remarks by ACRS member Hanauer. The Browns Ferry letter is reproduced on the following pages.

*The Browns Ferry report represented the birth of the so-called "asterisked items", later to become the ACRS generic items.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C.

March 14, 1967

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON BROWNS FERRY NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its eighty-third meeting, March 9-11, 1967, the Advisory Committee on Reactor Safeguards completed its review of the application of the Tennessee Valley Authority for authorization to construct Browns Ferry Nuclear Power Station Units No. 1 and No. 2. This project was previously considered at the eighty-first and eighty-second meetings of the Committee, January 12-14, 1967 and February 9-11, 1967, respectively, at a special meeting on February 28, 1967, and at subcommittee meetings on November 26, 1966, January 4-5, and January 28, 1967. Representatives of the Committee visited the site on February 27, 1967. During its review, the Committee had the benefit of discussions with representatives of the Tennessee Valley Authority, General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Browns Ferry Units are to be located in Limestone County, Alabama, on the shore of Wheeler Lake approximately 30 miles west of Huntsville. Each Unit includes a boiling water reactor to be operated at a maximum power level of 3293 MWt, the highest power level for any reactor reviewed for a construction permit to date. The average core power density is about 40 percent higher than for the previously reviewed Quad-Cities boiling water reactors. The increase is achieved by flattening the power density distribution and employing an approximately 20 percent higher fuel element maximum linear heat rate. The margins between thermal operating limits and fuel element damage limits are thereby reduced. In relation to margin on critical heat flux, the applicant uses new heat transfer correlations developed from recent experimental data.

The complex of emergency core cooling systems for Browns Ferry is similar to that proposed for the Quad-Cities reactors. Each reactor is provided with a high pressure coolant injection system; a low pressure coolant injection, or flooding, system; and two core spray systems. Because of the higher core power density and power level, substantial increases have been made in the flooding system and core spray system capacities. The Committee feels that the emergency core cooling systems proposed have a high probability of preventing core meltdown in the unlikely event of a loss-of-coolant accident. It notes, however, that although calculated peak fuel temperatures in such an accident are similar to those for the Quad-Cities reactors, the calculated number of fuel elements reaching undesirably high temperatures is greater. Also, the time margin available for actuation of the systems is less. Because of these factors and the importance of the effective functioning of emergency core cooling systems, the Committee believes the adequacy of these systems should be further corroborated by the following two measures:

1. Analysis indicates that a large fraction of the reactor fuel elements may be expected to fail in certain loss-of-coolant accidents. The applicant states that the principal mode of failure is expected to be by localized perforation of the clad, and that damage within the fuel assembly of such nature or extent as to interfere with heat removal sufficiently to cause clad melting would not occur. The Committee believes that additional evidence, both analytical and experimental, is needed and should be obtained to demonstrate that this model is adequately conservative for the power density and fuel burnup proposed.*

2. In a loss-of-coolant accident, the core spray systems are required to function effectively under circumstances in which some areas of fuel clad may have attained temperatures considerably higher than the maximum at which such sprays have been tested experimentally to date. The Committee understands that the applicant is conducting additional experiments, and urges that these be extended to temperatures as high as practicable. Use of stainless steel in these tests for simulation of the Zircaloy clad appears suitable, but some corroborating tests employing Zircaloy should be included.

The applicant stated that the control systems for emergency power will be designed and tested in accordance with standards for reactor protection systems. Also, he will explore further possibilities for improvement, particularly by diversification, of the instrumentation that initiates emergency core cooling, to provide additional assurance against delay of this vital function.

Steam line isolation valves are provided which constitute an important safeguard in the event of failure of a steam line external to the containment. One or more valves identical to these will be tested under simulated accident conditions prior to a request for an operating license.

Operation with a fuel assembly having an improper angular orientation could result in local thermal conditions that exceed by a substantial margin the design thermal operating limits. The applicant stated that he is continuing to investigate more positive means for precluding possible misorientation of fuel assemblies.

The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly by inlet coolant orifice blockage or by other means, to be remote. However, the resulting effects in terms of fission product release, local high pressure production, and possible initiation of failure in adjacent fuel elements are not well known. Information should be developed to show that such an incident will not lead to unacceptable conditions.*

A linear heat generation rate of 28 KW/ft is used by the applicant as a fuel element damage limit. Experimental verification of this criterion is incomplete, and the applicant plans to conduct additional tests. The Committee recommends that such tests include heat generation rates in excess of those calculated for the worst anticipated transient and fuel burnups comparable to the maximum expected in the reactor.*

The Rod Block Monitor system should be designed so that if bypassing is employed for purposes other than brief testing no single failure will impair the safety function.

The diesel-generator sets for emergency power appear to be fully loaded with little or no margin (on the design basis of one of three failing to start). They are required to start, synchronize, and carry load within less than thirty seconds. The applicant stated that tests will be conducted by the diesel manufacturer to demonstrate capability of meeting these requirements. Any previously untried features, such as the method of synchronization, will be included in the tests. The results should be evaluated carefully by the AEC Regulatory Staff. In addition, the installed emergency generating system should be tested thoroughly under simulated emergency conditions prior to a request for an operating license.

The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system and of inspection during service life. Because of the higher power level and advanced thermal conditions in the Browns Ferry Units, these matters assume even greater importance. The Committee recommends that the applicant implement those improvements in primary system quality which are practical with current technology.*

The Browns Ferry Units have been designed to provide the same accessibility for inspection of the primary system as for the Quad-Cities plants. A detailed inspection program has not yet been formulated by the applicant. The Committee will wish to review the detailed in-service inspection program at the time of request for an operating license.

Considerable information should be available from operation of previously reviewed large boiling water reactors prior to operation of the Browns Ferry reactors. However, because the Browns Ferry Units are to operate at substantially higher power level and power density than those on which such experience will be obtained, an especially extensive and careful start-up program will be required. If the start-up program or the additional information on fuel behavior referred to earlier should fail to confirm adequately the designer's expectations, system modifications or restrictions on operation may be appropriate.

The Advisory Committee on Reactor Safeguards believe that the items mentioned above can be resolved during construction of the reactors. On the basis of the foregoing comments, and in view of the favorable characteristics of the proposed site, the Committee believes that the proposed reactors can be constructed at the Browns Ferry site with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

The following are additional remarks by Dr. Stephen H. Hanauer. "It is my belief that the substantial increase in power and power density of the Browns Ferry reactors over boiling water reactors previously approved should be accompanied by increased safeguard system margins for the unexpected. The emergency core cooling system proposed should in my opinion be redesigned to provide additional time margin and to reduce the severe requirements for starting of large equipment in a few seconds. The dependence on immediate availability of a large amount of emergency electrical power, using diesel generators operating fully loaded in a previously untried starting mode, is of special concern, as are the high temperatures and numerous fuel-element failures predicted even for successful operation of the emergency core cooling system in a large loss-of-coolant accident."

Sincerely yours,

/s/ N. J. Palladino

N. J. Palladino
Chairman

* The Committee believes that these matters are of significance for all large water-cooled power reactors, and warrant careful attention.

References Attached

During the very intense discussions on Browns Ferry, at least two other ACRS members besides Hanauer had serious reservations about the review. The minutes of the special February meeting record the following:

> Acceptability of the Browns Ferry Proposal
>
> Mr. Palladino observed, with respect to Dr. Zabel's comments, that he had been assuming that the Committee would approve the proposal, with appropriate reservations. At Dr. Hendrie's suggestion, he called for a soft vote on the feasibility of the proposal.
>
> Nine members felt the proposal could be approved, four abstained from voting and two did not feel they were able to approve the proposal, even with reservations.
>
> The Committee voted to ask Dr. Hanauer and Dr. Zabel to discuss their conclusion that the facility should not be built.
>
> Dr. Zabel pointed out that there has been a gradule escalation in reactor systems and that his feeling was that there was some point beyond which it would not be prudent to go. There were many questions about this proposal for which there will not be answers for a long time, even assuming an extensive research program. He could also forsesee other systems of the same type being proposed at much worse sites.
>
> Dr. Zabel felt that at times the Committee has been willing to accept some uncertainty on the basis that enough leeway had been provided, but he had the uncomfortable feeling that such was not the case for this particular machine. In addition, attempts at alleviating the situation have only made it worse by forcing the use of unproven systems such as the large diesel-generators. Simply stated, Dr. Zabel felt that this was the place at which he would like to call a halt.
>
> Dr. Hanauer told the Committee that his feelings about the proposal were largely related to his experience with the ACRS within the past year. First, on Dresden 3 and Indian Point 2 the Committee wrote letters approving construction and had intended to write a general letter on core cooling. Instead the Task Force was convened. The Committee is now treating these as proven types and others of the general class get approval with little difficulty. Dr. Hanauer objected that he finds himself "stuck" with such systems regardless of the prudence of the initial approvals.

Secondly, the first of the bigger power reactors are presently being reviewed for operating licenses. In none of these are the problems which were of concern during the construction permit review even close to being solved. Dr. Hanauer has, therefore, concluded that it is no longer appropriate to assume that serious problems can be resolved between the construction and operating license reviews.

Finally, the nuclear industry has grown to the point where the Committee is not only concerned with Browns Ferry, but with many similar systems, and Dr. Hanauer felt that future similar proposals would be approved, based on experience with Dresden 3. He also commented that an industry with a two billion dollar backlog of profitable business has different responsibilities for demonstrating the adequacy of its hardware than does one existing on an AEC dole.

In view of the continued lack of resolution of problems identified on Dresden 3, in view of the departure represented by the proposal, and in view of the apparent reluctance of the ACRS to put teeth into its construction permit recommendations at the operating license stage, Dr. Hanauer did not feel that he could approve of the Browns Ferry proposal on the basis of existing technology.

Later in the meeting, Dr. Zabel stated that while he did not feel able to approve of the Browns Ferry proposal, he would not dissent to a Committee letter with appropriate reservations. Dr. Hanauer, however, indicated that he would dissent and agreed to put this dissent in writing.

The Committee agreed to tell TVA, through the Regulatory Staff, that the Committee would like to inform them of its reservations. The Committee also agreed that the applicant would be allowed time to caucus following the discussion. The Committee would also caucus, discuss its reservations with the Staff in order to attempt resolution of any disagreement, and finally inform the applicant of its decision.

The minutes of the March 1966 meeting record a tentative dissent of a different nature, as follows:

<u>Acceptability of the Double-Ended Pipe Break MCA</u>

During the discussion on the Browns Ferry letter, Dr. Bush submitted the following paragraph for the Committee's consideration and suggested that it might form the basis of a dissent to the letter, on his part.

> Dr. S. H. Bush expresses a general concern that the acceptance of a double-ended pipe break, common to all water reactors, may lead to inherently less safe Emergency Core Cooling Systems. Such a pipe break is an admittedly incredible event in large austenitic stainless steel piping when the normal failure is by limited circumferential or longitudinal cracking. The acceptance of the double-ended pipe break determines the time of blowdown and the sizing of pumps, valves and, more significantly, the diesel-driven emergency power generators. This break leads to severe requirements for short startup times of very large diesel-generators, representing a yet-undeveloped technology. A more realistic failure model, based on smaller break sizes, should permit a more rational sizing of ECCS equipment.

The Committee decided to study the matter of the double-ended pipe break on a generic basis, and member Bush did not attach the remarks to the Browns Ferry report.

The minutes of the 83rd meeting also record that the ACRS discussed the matter of emergency plans with the AEC Commissioners.

> Mr. Palladino observed that, at present, emergency plans were usually geared to the individual needs of existing facilities and were not adequate for wide application. Several factors have brought this to the Commission's attention, including the recent trend of power reactors toward population centers, the tendency toward multi-reactor siting, the recent large increases in design power, the increase in the handling of radioactive material which will result from the growth of the nuclear industry, and actions with local officials because of the possibility of public relations problems. Mr. Palladino concluded by saying that the Committee was not suggesting that off-site drills be held, but rather that the situation be studied; the study perhaps going beyond evacuation capabilities to include the need for development of a wide-range stack monitor, cloud tracing techniques, training programs for local fire and police departments and the development of widespread hospital capability for treating irradiated patients. The Committee's feeling is that in some sense evacuation is being depended upon and the AEC should be made aware that a study might be indicated.

At the 84th meeting, April 6-8, 1967, and the 85th meeting, May 11-13, 1967, the ACRS again discussed metropolitan siting. It was proposed that a limit on acceptable population density for the next five years be recommended to the AEC which, in effect, closely enveloped the existing sites for LWR's. At the 85th meeting, the ACRS agreed to tell Mr. Price, the Director of Regulation, that the Committee was considering such a limit in the form

$$P_{TOT(R)} \leq 4000(R^2) \text{ using current population}$$

$$\leq 5000(R^2) \text{ using population projected 25 years hence}$$

$$5 \leq R \leq 25 \text{ in miles}$$

where $P_{TOT}(R)$ is the total population within a distance R miles of the reactor.

The minutes of the Subcommittee meeting on Metropolitan Siting of Reactors held May 30, 1967 indicate that both the Subcommittee members and the Regulatory Staff were favorable toward proposing some such interim criterion, to be applicable for about five years, by which time some operating experience should be available with the larger reactors under construction. Also, this would have provided time for safety research and the development of improved safety systems. Appropriate population limits would also apply for distances less than 5 miles, and consideration was given to rejection of the site if any 22 $1/2^{o}$ sector exceeded the average limit significantly.

The minutes of the 86th meeting, June 8-10, 1967, state that ACRS Chairman Palladino reported to the Committee on a discussion which he had held with Chairman Seaborg of the Atomic Energy Commission concerning the issue of metropolitan siting. Chairman Seaborg expressed concern over the use of a simple formula to establish a criterion for siting of reactors for several reasons, one of which was apparently its effect on the proposed Metropolitan Water District reactor on Bolsa Island. Dr. Glenn Seaborg had raised several pertinent questions such as how one handles a large population very close to a site. He indicated that one must consider situations in which a site has particularly favorable meteorology or a relatively large, unpopulated area close to the site with a large population center further out. Dr. Seaborg felt that such situations should be covered by any criterion established.

In short, Dr. Seaborg had encouraged the Committee to proceed slowly and be kept informed. The minutes indicate that the Committee discussed the matter considerably in the executive session without arriving at any agreement. It then met with Mr. Price and senior members of the Regulatory Staff. Mr. Price reported that he had met with the Commissioners and had gotten the idea that they were generally unhappy with the idea of a fixed threshold limit line and with an arbitrary formula. They would prefer a more flexible approach and were influenced by several considerations. First of course, the Metropolitan Water District site at Bolsa Island was beyond any of the threshold limits considered. There had already been a great deal of publicity given this project in obtaining Congressional approval. In addition, the proposed criterion did not credit sites that were very good within less than 5 miles. Mr. Price felt that, based on these meetings, it would be difficult to convince the Commissioners to promulgate such a criterion any time soon.

Following Mr. Price's report, there ensued considerable discussion between the ACRS and the Regulatory Staff. The meeting finally terminated with the position of continuing to proceed, as before, on a case-by-case basis. During the discussion one point which was made was that recent testimony before the Joint Committee on Atomic Energy (JCAE) had, in effect, imposed a moratorium on metropolitan siting.

In fact, in the ACRS testimony presented by Palladino and Okrent to the JCAE in April 1967, the Committee stated:

> **The ACRS believes that placing large nuclear reactors close to population centers will require considerable further improvements in safety, and that none of the large power reactors now under construction is considered suitable for location in metropolitan areas. The Committee believes that, in addition to favorable experience with reactor construction and with operation of reactor systems, components and safeguards in these reactors now under construction, further improvements in design are required to make accidents, large and small, still more unlikely; and the consequences limiting safeguards must be made more fool-proof, and provide protection from the consequences of accidents of still lower probability.**

Testimony by the AEC at the JCAE hearings also indicated that the time was not ripe for metropolitan siting. But, no quantitative definition was given of a metropolitan site.

In July 1967, the ACRS actively began reviewing the proposed Burlington site. The minutes of the 87th meeting, July 6-8, 1967 give some insight into the course of events.

87th Meeting, July 6-8, 1967
Burlington Station/Metropolitan Siting

Mr. Palladino reported on a conversation with Mr. Price. There has been a great deal of interest in the Burlington project, including a letter to Chairman Seaborg from the governor of New Jersey asking why the ACRS had not yet reviewed the project. Mr. Price asked that the Committee make every effort to have the project on its August, 1967 agenda. He noted, however, that Mr. Price would be reluctant to have the Regulatory Staff's conclusions in writing since such documents have been difficult to keep private. It could prove embarrassing if a controversy should arise between the ACRS and the Staff.

Dr. Monson observed that Mr. Price had made similar statements to the Committee's staff. He noted that, as Subcommittee Chairman, he had several points to raise with the Committee. Firstly, this will not be a normal review since the Committee has been asked to perform a site review after submission of a complete application. Also, the question being put to the ACRS is actually one of the acceptability of the site on the basis of population distribution, alone.

Mr. Palladino and Dr. Hanauer thought that a complete site review had been requested. Dr. McKee noted that there are many site-related questions, e.g., the supply of Delaware River oysters and the Philadelphia municipal water supply.

Dr. Monson observed, however, that the only reason for a review of the site at this time is the population distribution, which is the only thing of significance in the kind of partial review which is being considered. If a complete site review were to be done, the entire facility would have to be considered. Dr. Monson posed two questions:

1. Is a Subcommittee meeting necessary or is this properly a question for the full Committee?

2. Should there be a site visit?

Dr. Hanauer thought that if all sites such as Burlington were to be ruled out, a Subcommittee meeting was not needed; if more information is required about Burlington, then the Subcommittee should meet.

Dr. Monson thought there was no point to a site visit if only the population question was of importance. Mr. Mangelsdorf felt that the Committee would be criticized for reporting unfavorably on a site it had not visited, if the application were to be rejected. Dr. Hendrie and Mr. Palladino agreed.

Dr. Monson posed a third question. If the Subcommittee meets, should the Staff be asked for a prior report with a statement of position. Dr. Monson thought that the reasons forwarded by Mr. Price for not doing this applied equally well to all projects. At present, the Committee has only a descriptive report which says that, while the questions raised by the high population density nearby should be resolved, the Regulatory Staff has not decided to consider the Burlington site a metropolitan area. Under these conditions, it seemed clear to Dr. Monson that it will be an ACRS decision which puts a moratorium on sites in high population-density areas.

Dr. Hendrie felt he could understand the Staff's quandary, however, Mr. Palladino recalled that the Committee and the Regulatory Staff had agreed to discuss differences before issuing formal statements.

Dr. Okrent moved the following:

> The Burlington Subcommittee hold a meeting at the site to determine the validity of the population distribution figures which have been presented and such other site information as is appropriate. The Regulatory Staff need not have a written position concerning the Burlington site prior to the August 1967 ACRS meeting, although the Committee would like an oral statement of position from the Regulatory Staff at that meeting.

Dr. Isbin seconded the motion and it was passed without dissent.

Dr. Monson then proceeded with his statement. The Burlington site is located on the Delaware River, in New Jersey. It is 17 miles from the Philadelphia-Camden area with about 2 million people. There are 16 communities of more than 25,000 people each within less than 25 miles. Trenton, New Jersey, containing about 125,000 people is 12 miles away. The applicant has stated that due to the cost of transmission, there are no other suitable sites. In addition, air pollution problems have ruled out the construction of a fossile-fuel plant at the Burlington site.

Dr. Monson then quoted from a series of newspaper articles and public statements by Commissioner Ramey, Chairman Seaborg, etc., which indicated that the Bolsa Island project was very highly favored. In Dr. Monson's view the Bolsa Island site, which essentially duplicates Burlington with respect to total population density, is even worse than Burlington because of the concentration of people to the landward side and the continuous, daytime on-shore winds. Since the wind direction frequency is more randomly distributed with respect to the people at the Burlington site, it has a lower effective population density.

Dr. Monson also recalled for the Committee many recent statements by the Director of Regulation and the Chairman to the effect that AEC policy is to not site power reactors in metropolitan areas and not to allow them to encroach on such areas until there have been significant offsetting improvements in reactor technology. Dr. Monson thought that the inconsistency of this statement with the apparent predetermined approval of the Bolsa Island site made it difficult to reject the Burlington site on a population basis. He suggested that this matter be discussed with the Commissioners.

Dr. Isbin felt that there were significant differences between the two projects. The Burlington proposal includes conventional reactors under private ownership. Since the Bolsa Island facility will be heavily supported by the AEC, it might be a good starting point for establishing requirements which will allow the use of high population density sites.

Dr. Okrent thought that the ACRS should formulate its opinions on the Burlington project before any discussion with the Commission. Since the Bolsa Island design has not been completed there may be some willingness to go quite far on other features to allow the use of the site. Dr. Okrent did not agree with the conclusion that since a decision has already been made on Bolsa Island, if indeed it has, then the Committee must accept it and therefore accept the equally undesirable Burlington site.

Dr. Hendrie agreed that the Committee should decide its position on Burlington before any discussions with the AEC. Dr. O'Kelly thought the same.

Mr. Palladino obtained agreement that the general opinion was not to discuss the question with the Commission during the August meeting.

Mr. J. E. McKee did not participate in the discussion concerning the Bolsa Island project.

As noted in the meeting minutes, a letter dated July 3, 1967 was written by the Governor of New Jersey to Chairman Seaborg of the AEC, expressing concern about the prolonged schedule for the regualatory review of Burlington 1. The Governor in his letter pointed out the need for power in his state from this facility and that this had a bearing on the public health and safety. He endorsed construction of nuclear plants as opposed to fossil fuel burning plants as a solution step to the problem of the air pollution in the area.

The ACRS Subcommittee on Burlington visited the site and held a meeting on August 9, 1967. The topics discussed included population distribution and evacuation capability. Public Service of New Jersey stated that they had no site better than Burlington, that no sites with, say, half the population density of Burlington existed except in the pine woods of central New Jersey where no cooling water was available. Public Service also stated they were proposing no special design or operating features to compensate for the high population density.

The Regultory Staff had submitted a report to the ACRS in late July which estimated parametrically the off-site doses in terms of the classical Part 100 recipe, namely 100% of the noble gases and 50% of the iodine released to an intact containment, with various assumptions on iodine removal within the containment, etc. The report said that unless significant removal of radioiodine is achieved in the containment, doses could exceed Part 100 guidelines within a matter of a few hours for several thousand people. But the report contained no conclusions concerning the acceptability of the site.

However, at the 88th meeting, August 10-12, 1967, Mr. Price told the ACRS that the Regulatory Staff thought Burlington was a poor site.

The minutes of this discussion follow:

> <u>Regulatory Staff</u>
>
> **The Committee was joined by Mr. Price, Drs. Beck and Mann, Mr. Case and Dr. Morris. Mr. Case noted that there was agreement among all those present from the Regulatory Staff as to the position which he would report to the Committee on the Burlington proposal.**

Generally, the Staff thinks that the proposed site is a bad one. If it is not in fact a metropolitan site it is so close as to be indistinguishable. In view of the Committee's statements, both written and in testimony before the JCAE over the past two years, and in view of positions taken bv the AEC during the same period, the industry generally should have been aware that the Burlington site represented a poor choice.

Mr. Price felt that the applicant will say that no other sites are available within their service area. There is a need for power and in particular for nuclear power. Mr. Price noted, however, that in February 1966 Public Service Company officials talked with Dr. Mann concerning siting of nuclear power reactors. Two sites were discussed, one at Burlington and an alternate site in Delaware. At that time they were told plainly that the Burlington site was bad and indications were that the decision would be to apply for a permit to construct a power reactor at the site in Delaware. In December of 1966, and without further discussion, the company publicly announced its intent to construct a reactor on the Burlington site and followed this announcement with a formal application.

In February 1967, Mr. Price met with Mr. Baker and Mr. Smith, the Vice President and General Manager, respectively, of Public Service Company. Drs. Beck, Mann and Morris were present. While an attempt was made not to prejudge the case, Mr. Price did what he could at that time to inform them that the site would probably be considered a poor choice. Mr. Price urged that if the Committee reaches a different conclusion this matter be discussed until either the Committee or the Staff have been convinced of the other's view.

Mr. Price also noted that he had had a telephone call from Mr. Ramey. Mr. Ramey was concerned as to how the application had found itself in this condition and why the situation had not been handled in the same way as the Boston Edison application with respect to the Edgar Site. Mr. Price informed the Commissioner that the situation had arisen because the company elected not to follow the approach taken by Boston Edison.

Mr. Beck added that in the view of the Regulatory Staff it seems as though, during the past three years, there has been substantial progress in improving the safety of large reactors, at least so far as paper studies are concerned. The Public Service

Company is, in addition, proposing something which may improve the situation still further.

However, their proposal represents a long step without substantial operating experience in any of the larger power reactors. This represents a considerable jump beyond the Indian Point site and Dr. Beck concluded this site should not be approved. Mr. Price made one last point with respect to the argument that no appropriate sites are available within the Public Service Company service area. He pointed out that there is nothing preventing them from building outside their service area. While efficiency might be hurt by such an approach, only conventional engineering and money would be involved. He pointed out as an example that the site in Delaware originally proposed as an alternate and on which the Peach Bottom reactor has been constructed is not in the Public Service Company's service area. They are involved in Peach Bottom in conjunction with several other utilities.

Dr. Isbin asked, with respect to the lack of operating experience, if negative results come in during the next four years on some of the items identified as potential problems in recent Committee letters, are not the companies who have engaged in the projects taking the risk. Dr. Morris replied that they are taking the risk but predicted that considerable pressure will be put on the AEC to approve plants in any case. Mr. Price added that after 15, 20 or 30 such large power reactors come into operation the country would be heavily dependent on their power production capability.

Dr. Beck felt that there should be some reasonable assurance in each case that the project can go forward as proposed.

Dr. O'Kelly asked if the Staff's conclusion had been based on the type of reactor proposed for Burlington. Mr. Price replied that any reactor in an area of similar population density would probably not be suitable. Those things which the ACRS, Commissioner Ramey, he himself and others have been saying repeatedly were necessary to construction close to metropolitan areas have not yet been done.

Mr. Mangelsdorf asked if this case bore any relation to the Bolsa Island project. Mr. Price replied that at least the Bolsa Island site is a good one for approximately 1-1/2 miles from the reactor. The Burlington site is bad after 500 feet. He noted also that the present schedule proposed for the Bolsa Island project is absolutely impossible and that some additional time will be available.

Dr. Zabel observed that no opinion was expressed in the Staff evaluation. Mr. Price replied that that had been deliberate. The Staff's opinion had just been presented orally to the Committee.

Dr. Monson noted that the schedule for the Burlington plant called for power generation by May, 1971. He asked if any large power reactors other than Turkey Point 3 and 4, Indian Point 2 and Palisades would be in operation, particularly PWR's. Dr. Beck thought there would not be others which had been operational for a long enough period by May of 1971 to assist in reaching any conclusions on the Burlington proposal.

Dr. Beck felt that the applicant was amenable to thinking in other terms. Mr. Price agreed and had suggested that only an amendment for a new site would be necessary if this site proved unsuitable. This would mean that all of the time so far expended in the review of the Burlington proposal would not have been wasted.

Mr. Palladino stated that the Committee had agreed that the applicant should be told at this time that the site is not considered appropriate. Mr. Price agreed with this position but suggested that some consideration be given to how answers which would be necessary to future public inquiries. He suggested that it would be preferable not to have to say in the future that the applicant had been flatly turned down. He did note that both the ACRS and the Regulatory Staff have decided the site is no good and that someone must tell the applicant. He suggested, however, that he would be in a more "livable" position if the applicant was told that the Advisory Committee on Reactor Safeguards sees no way that the proposed site could be approved. He pointed out that he was thinking in terms of future public announcements.

After discussion with the applicant, the ACRS* met in executive session and adopted the following position:

*The ACRS members at the 88th meeting were the following: N. J. Palladino, Chairman, S. H. Bush, H. Etherington, W. L.Faith, F. A. Gifford, S. H. Hanauer, J. M. Hendrie, H. S. Isbin, H. G. Mangelsdorf, H. O. Monson, A. A. O'Kelly, D. Okrent, W. R. Stratton and C. W. Zabel.

> **The Committee believes that it has now received essentially all of the information necessary to evaluate the Burlington site and has given careful consideration to this information. It is the unanimous opinion of the Committee that it does not see how the site can be approved.**

The Committee met with the applicant and advised the applicant orally of its position. Mr. Price noted that a public response from the ACRS and the Regulatory Staff could be avoided. Mr. Bake of Public Service replied he perferred that there be no letter and noted that the company would have to reconsider its position.

The ACRS recorded its position, as stated to Public Service of N.J., in its monthly summary letter to the Chairman of the AEC. At that time, the summary letters were not routinely made public by the AEC. However, the fact that the Burlington site had been rejected became generally known.

Public Service withdrew its application for the Burlington site. The Bolsa Island site, which was similar to Burlington regarding population distribution, except for the first 1 1/2 to 2 miles, still remained for consideration soon. Interestingly, in the testimony by AEC Commissioner Ramey to the JCAE on April 4, 1967, and in the testimony by Mr. Price and Dr. Beck, no mention was made of Bolsa Island as a site which presented problems arising from the large surrounding population density, although the seismic design aspects of the proposed project were discussed. The one exception arises in a comment by Congressman Hosmer.

> **Back in my mind I have the question of the Bolsa Island reactors, two reactors, which will be in the same general location (the Los Angeles basin). I would believe that every effort would be made to make these compatible with their intended locality.**

To which Mr. Price replied "that is right."

At the same hearings, Congressman Hosmer asked "Do you think you could really justify building reactors with lower safety standards in remote areas than wold be required in metropolitan areas anyway?" To which Dr. Beck replied,

> **We are not, in fact, suggesting that reactors be built to any different standards at one place or another. We are building reactors to the best standards we know, at any location. We say that there are still some residual uncertainties in reactors.**

> The actual experience with reactors in general is still quite limited and with large reactors of the type now being considered, it is non-existant. Therefore, because there would be a large number of people close by and because of lack of experience, it is not a matter of difference of standards; it is a matter of judgment and prudence at present to locate reactors where the protection of distance will be present.

During part 2 of the 1967 hearings by the JCAE on Licensing and Regulation of Nuclear Reactors (September 12-14, 1967), statements supporting metropolitan siting of reactors were made by several representatives of the nuclear industry. Mr. Jack Horton, speaking on behalf of the Edison Electric Institute testified,

> The siting of nuclear power plants in metropolitan areas is important to the electric utility industry and the public which it serves. Utilities serving metropolitan areas must have their sources of generation close to the load they serve if they are to continue giving reliable service. Therefore, siting of nuclear power plants in metropolitan areas must be a key factor in the design of our future electric power systems. We believe the AEC is moving toward this goal.

A. E. Schubert of General Electric responded,

> We agree that requirements for metropolitan siting need to be defined. To this end, we believe it would be helpful for the AEC to establish a government-industry task force, which would include appropriate ACRS and Regulatory Staff membership, to make recommendations for metropolitan siting criteria.

And J. C. Rengel of Westinghouse testified,

> We believe that a change is called for in AEC policy on siting of nuclear power reactors in or near metropolitan areas. We believe that utility groups can make an economic case for such locations and plants can be designed which can be constructed and operated safely in metropolitan areas.

When asked by Congressman Hosmer, "Are you talking about, say, Queens, in New York City"? Mr. Stern of Westinghouse said "We are talking Queens." Mr. Stern went on to discuss "tne importance of eliminating the possibility of out-leakage beyond the boundary of the containment itself" for metropolitan sites, but did not discuss the potential for accidents which could lead to containment failure.

The year or so following the 88th ACRS meeting in August 1967 involved a variety of complex matters relating to the siting of reactors in populated areas. The ACRS attempted to provide guidance on what might make sites more populated than Indian Point acceptable for reactors; and also, an effort was made to develop some relatively simple basis for comparative site evaluation, particularly with regard to surrounding population. The probable need to review the Bolsa Island site was always present. Construction of the Zion reactors north of Chicago had been proposed. And Consolidated Edison had proposed for consideration three somewhat more populated sites than the Indian Point 2 site, as possible locations for large boiling water reactors.

Further down the road would be Newbold Island site which Public Service of New Jersey would propose as its alternate to Burlington.

At the 89th meeting, September 7-9, 1967 the ACRS decided that the Metropolitan Siting Subcommittee should continue its efforts toward the development of criteria for more populated sites. At the 90th meeting, October 5-7, 1967, Mr. Price, Director of Regulation, informed the ACRS that the Jersey Central Power and Light Company had proposed the location of a reactor at its Union Beach site. Jersey Central was so much in favor of this that they had suggested approaches such as double containment, etc. They had also requested some informal reaction from the Regulatory Staff. Mr. Price observed that Union Beach seemed to be a better site than Burlington for the first few miles out, but then it becomes much worse. Later during the 90th meeting the ACRS voted to have its Chairman inform Mr. Price that "if the proposed Union Beach facility is similar in design to those now being reviewed, the Committee would have great difficulty reaching a favorable conclusion," as was the case with the original Burlington proposal. The adopted motion also expressed the Committee's willingness to consider new departures, in connection with densely populated reactor sites.

The minutes of the 90th ACRS meeting take note that in testimony before the Joint Committee on Atomic Energy representatives of the nuclear industry had suggested that a panel be established to determine metropolitan siting criteria.

At the 94th meeting February 8-10, 1968, the ACRS decided that, in view of the negative attitude of the Commissioners, further efforts to try to develop quantitative population limits on site acceptability would be de-emphasized and that the Metropolitan Siting Subcommittee would be asked to develop for full Committee consideration means, methods and techniques for studying the entire metropolitan siting question. The effort to set up a "hardware table" which defined greater safety requirements for more densely populated sites was discontinued at that time. And the Subcommittee concentrated first on a method of comparing sites.

At the 95th meeting, March 7-9, 1968, the ACRS discussed with Consolidated Edison certain aspects of the Indian Point 2 reactor, including the core catcher, which the Applicant had proposed in the middle of the construction permit review and which the ACRS had not accepted as being adequate, but which had been left in the design. The question implicit in the discussion during this particular meeting with the ACRS was whether the Applicant could remove the core catcher. We will come back to this controversial and frequently emotional matter in considerable detail in the Section "China Syndrome, Part 2." What is important is that there had been no further technical development by Westinghouse of a core catcher system, and there was no more assurance in 1968 than there had been in 1966 that the design proposed could cope with core melt and prevent containment failure. In effect, it was a take-it or leave-it kind of core catcher that was still in the construction permit application.

At the 96th meeting, April 4-6, 1968, the ACRS again reviewed the question of the core catcher for Indian Point 2; and it began its review of the application for two large PWR's at Zion, a site having roughly similar population characteristics to the Indian Point site. In his discussions with the ACRS, Mr. Price took the approach that, if a core catcher was going to be required, or to be left, in Indian Point 2, then it would be required for Zion 1 and 2. This difficult subject was discussed in considerable detail during the 96th meeting. The ACRS decided to defer making a decision on Indian Point 2, until after the Committee had further benefit of a report from the Metropolitan Siting Subcommittee which was to provide a comparison of the Zion and Indian Point 2 sites.

As if life were not already complicated enough, during the winter and spring of 1968, the ACRS was completing a long, very difficult and controversial (within the ACRS) review of the first, large, high-temperature gas-cooled reactor (HTGR), the Fort St. Vrain reactor, which was at a relatively remote site. The Committee did not reach a unanimous position, and the ACRS letter to Chairman Seaborg dated May 15, 1968 on Fort St. Vrain included a dissent by member Hendrie, who did not approve of this reactor without a containment building, and additional remarks by member Okrent not opposing contruction of this reactor but expressing concern for the construction of large reactors of the Fort St. Vrain type at more populated sites without additional features to cope with major accidents including various modes of failure of the reactor vessel.

The ACRS report on Fort St. Vrain is included for persepctive.

COPY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 15, 1968

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: FORT ST. VRAIN NUCLEAR GENERATING STATION

Dear Dr. Seaborg:

At its ninety-seventh meeting, May 9-11, 1968, the Advisory Committee on Reactor Safeguards completed a review of the application by the Public Service Company of Colorado (PSC) to construct a nuclear unit at its Fort St. Vrain site in Weld County, Colorado. Previous consideration had been given to this project during the Committee's eighty-seventh meeting, July 6-8, 1967, eighty-ninth meeting, September 7-9, 1967, ninety-first meeting, November 2-4, 1967, ninety-fifth meeting, March 7-9, 1968, ninety-sixth meeting, April 4-6, 1968, and special meeting, April 27, 1968. Subcommittee meetings were held May 25, 1967 at the site, June 12, 1967 at La Jolla, California to review General Atomic's research and development programs, and August 8, 1967, September 6, 1967, November 1, 1967, December 6, 1967, and March 6, 1968. During its review, the Committee had the benefit of discussions with representatives of the Public Service Company of Colorado, General Atomic Division of Gulf Oil Corporation (GGA), and the AEC Regulatory Staff and their consultants. The Committee also reviewed the documents listed.

The Forst St. Vrain unit will be located about 35 miles north of Denver, Colorado between the South Platte River and St. Vrain Creek. It is a High Temperature Gas-cooled Reactor (HTGR) cooled with helium and designed to produce 842 MWt (330 MWe) with an ultimate capacity of 879 MWt. The prestressed concrete reactor vessel (PCRV) proposed for this plant is the first in the United States. This PCRV is to contain not only the core, but the entire primary coolant system. The plant utilizes a confinement building equipped with ventilation filters for removing particulates and iodine from the building exhaust.

The PCRV has inside dimensions of about 31 feet in diameter and 75 feet in height with walls ranging from 8 to 18 feet in thickness. A 3/4 inch thick carbon steel liner is to provide a leak tight barrier. This liner is covered with a thermal insulation which, in combination with water cooled coils behind the liner, limits the temperature in the concrete.

This vessel contains the 740° F to 1470° F, 700 psi helium coolant. The pressures are contained by the PCRV which is strengthened by reinforcing bars and a series of axial and circumferential prestressing tendons. The penetrations in bottom and top heads each contain two steel closures designed to Section III of the ASME Code. A system of strong-backs is designed to prevent the possibility of a blowout of the closures on the larger penetrations.

Since this is a first-of-a-kind reactor, the Committee believes that particular attention must be paid to final design, construction, and quality control. Specifically, the Committee believes the following are critical:

(1) Careful attention to good construction procedures is essential during construction of the PCRV in light of its vital function in this plant.

(2) There should be quality control groups representing PSC and GGA. These groups should be staffed with well qualified personnel not responsible to the construction organization. Clear channels of authority should exist that will insure continuing attention to meeting rigorous quality standards.

There are several items of safety oriented research and development that the applicant has underway that are critical to the safety of this system:

(1) It is essential that the integrity of the prestressing tendon system be maintained throughout the life of the plant. The Committee recommends that the AEC Regulatory Staff follow research and development programs on corrosion protection for this system and evaluate the proposed method prior to installation. Before the beginning of operation, a systematic program of surveillance should be developed appropriate to the method of corrosion protection used.

(2) Since forced circulation is essential in cooling the HTGR, the gas circulators perform a vital function. Assurance must be obtained through appropriate research, development, and analytic studies that these circulators will perform satisfactorily.

(3) The applicant has indicated that additional information is to be developed on the thermal insulation as a function of time, temperature, fluence, vibration, and impurity levels.

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(4) The steam generators are made up of modular units, placed within the PCRV. Leakage in conventional steam generators is not uncommon. In the Fort St. Vrain steam generators, leakage might have the undesirable effect of introducing water into the reactor; therefore, the design, testing and construction programs necessary to establish the reliability of these units require increased and special attention.

In certain low probability accidents leading to injection of water into the PCRV, there is a possibility of a substantial increase in PCRV pressure. The Committee recognizes that the steam generator dump systems and the proposed pressure relief valve both serve to control this overpressurization. However, the Committee believes that a second pressure relief valve is required.

A system of instrumentation is to be installed in the concrete walls to monitor the response of the PCRV during operation. The Committee concurs with this approach and suggests that suitable attention be given to location and redundancy of instruments to insure continued monitoring of all critical regions throughout the life of the PCRV.

Because of the nature of the liner and thermal insulation design, the liner is not accessible for inspection after startup. The Committee believes particular attention must be paid to the inspection of the liner during construction, and that the applicant should continue to investigate possible methods of detecting incipient failure of the liner.

The Advisory Committee on Reactor Safeguards believes that the above items can be resolved by the applicant and the AEC Regulatory Staff during construction. The Fort St. Vrain site provides an acceptable degree of isolation when considered in relation to the proposed high integrity PCRV. Based on these factors, the Committee concludes that the Fort St. Vrain unit can be constructed at this site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Additional remarks of Drs. Okrent and Hendrie are attached.

Sincerely yours,

/s/
Carroll W. Zabel
Chairman

Dr. David Okrent makes the following additional remarks:

"The Fort St. Vrain Station will have the first prestressed concrete reactor vessel designed and constructed in the United States, although such vessels have been built abroad. However, even abroad, only limited experience exists with these vessels. Not all of that experience has been favorable, and none of the existing experience covers more than a fraction of the operational life of the vessels. Only a limited amount of safety research work has been done in connection with various failure modes of these vessels, or on the effects of anomalies and errors in design, construction, or operation. History teaches us that errors and misjudgements have been and will be made in the design and construction of vital components. The chance of such errors is increased when a long experience with design, construction, and operation is not available. At this time, it is not clear to me that significant faults in a PCRV would necessarily be detected prior to the loss of integrity of the vessel. The inaccessibility of the vessel liner, cooling tubes, and thermal insulation compound this difficulty.

"From the standpoint of reactor safety, the Fort St. Vrain design is especially vulnerable to vessel failure because a single structure serves as both reactor vessel and secondary containment. I believe it acceptable to construct the Fort St. Vrain station, in view of the remote character of the site, the moderate power of the reactor, the apparent great conservatism in the design of the reactor vessel, and the fact that only one unit is involved. However, I believe that it would not be prudent at this time to construct larger reactors of the Fort St. Vrain type at more populated sites without additional safety features to cope with major accidents involving various modes of failure of the reactor vessel."

Dr. Joseph Hendrie makes the following additional remarks:

"I believe the Fort St. Vrain reactor should be contained in a building of such design pressure and leakage characteristics as to protect the public in the event of a major failure of the reactor vessel. I do not agree with the applicant's argument that the present design of the reactor vessel provides both primary and secondary containment of the reactor in an adequate manner. The great merit of the traditional secondary containment building is that it is a separate and independent barrier to protect the public from the effects of failures of the primary system. In the Fort St. Vrain design, this essential separation is lost, and the safety of the public depends upon the integrity of a single structure. The applicant concludes that a significant loss of integrity of the reactor vessel is impossible, due to the reinforced, prestressed concrete construction. This may be a correct conclusion, but in a matter as important as the public safety I believe it should be supported by a substantial amount of favorable experience in the construction and operation of high-temperature, gas-cooled reactors with concrete vessels. In the absence of such experience, I believe the Fort St. Vrain reactor, and any similar units that might be proposed in the near future should have secondary containment buildings."

It is of interest also to note that in meetings between the ACRS and the AEC Commissioners in March, 1968, Chairman Seaborg expressed his concern over the costs and economics of safety. He cautioned against going to extremes without consideration of costs. It was noted that some utilities had said they did not dare to discuss cost features with the ACRS. Dr. Seaborg expressed his concern about piling one safeguard on to back up another. He referred to the core catcher crucible at Indian Point 2. And the Commissioners suggested an improved dialogue between industry, the Regulatory Staff and the ACRS, and that it would be helpful if general problems could be solved on a non-case-by-case basis.

Actually, a rather considerable number of light water reactors had received favorable construction permit reviews in 1967 and early 1968, including H. B. Robinson 2, Browns Ferry 1 and 2, Monticello, Point Beach Unit No. 1, Vermont Yankee, Peach Bottom 2 and 3, Three Mile Island Unit 1, Fort Calhoun Unit No. 1, and Oconee Units 1, 2 and 3, among others. All of these had sites with surrounding population densities far smaller than Zion or Indian Point.

2.11 ZION

In the summer of 1967 Commonwealth Edison filed an application for construction of two 3250 MWt PWR's at Zion, Illinois, (population 14,000) between Chicago and Milwaukee, six miles north-northwest of Waukegan, Illinois (population 55,719 at the time). The preliminary report on Zion by the Regulatory Staff to the ACRS, dated September 13, 1967, notes "Our review of the site will emphasize population distribution in the vicinity. A preliminary comparison reveals a definite similarity to the distribution around the Indian Point site." The Regulatory Staff basically completed its review of Zion and issued a report to the ACRS, dated March 18, 1968, which was favorable to construction with a few (typical) minor reservations, including one requiring charcoal filters in the containment to further reduce the idoine inventory during an MCA. Although the first ACRS Subcommittee meeting on Zion was not held until March 21, 1968, the proposed reactors were the focus of discussion within the Committee prior to that time. For example, at the 93rd meeting, January 11-13, 1968 the ACRS discussd in executive session the progress (or lact thereof) in resolution of the growing list of asterisked items, initiated with the Browns Ferry review, a year earlier. Member Okrent urged Committee action to obtain resolution of these items and moved that their resolution be accomplished (if possible) prior to completion of review of the next site like Indian Point 2, that is, Zion. Member Mangelsdorf moved that a Subcommittee be set up to review the progress of the asterisked items, and this compromise position was adopted, including urging that resolution be pursued.

At the first meeting of the Zion Subcommittee, a major interest of the ACRS members related to what addtional measures they might recommend for the Zion reactors which were essentially a replica of the recently approved Diablo Canyon reactor, the latter being at a very remote site. Various potential topics for improvement were identified, including partial protection against pressure vessel failure, improved protection against sabotage, increased margins in engineered safeguards, pressure vessel cavity flooding (to cope with the recently identified potential for reactor vessel failure from the thermal stresses (at low pressure) due to injection of cold water following a LOCA). It was questioned by member Hanauer whether it was appropriate to use Part 100 as a basis for designing such a plant.

According to the analysis of the Regulatory Staff, there was no meaningful difference in population distribution between Zion and Indian Point. Zion was on a lakefront and its average population distribution was similar to Indian Point 2; however, in some directions it had significantly higher population densities (4 times) e.g., if one considered the sector which encompassed Waukegan. The ACRS asked its Metropolitan Siting Subcommittee to provide an independent assessment of the relative population characteristics of the Zion and Indian Point sites.

Zion recieved an initial hearing by the full ACRS at the 96th meeting, April 4-6, 1968. Excerpts from the minutes provide some insight into the thinking.

Member O'Kelly asked if the Committee's philosophy would be something like this: that the Zion site is about the same as the Indian Point site and therefore should be approved. Member Joseph Hendrie said he felt that if the Zion site was acceptable for some undefined reactor, then the Committee should determine what additions must be made to the Diablo Canyon reactor to make it acceptable at the Zion site.

Although the ACRS was still waiting for a report from its siting subcommittee, the general consensus seemed to be along the lines of the comments by O'Kelly and Hendrie. However, the seemingly strong trend toward improvements in safety for Zion was undercut by frequent references by some members to the core-catcher proposal made for Indian Point 2 by the applicant (which had not been accepted by the ACRS as an adequate means, as proposed, to cope with a large LOCA followed by core melt). These members stated that the Committee's desire for "something more" might force the Applicant into something which was ill-considered.* This entire matter was somewhat of a sore point within the Committee, since other members felt "let-down" by the ACRS decision to accept the Task Force recommended by the AEC, instead of issuing a general letter back in September, 1966. Now this feeling was reinforced by the Task Force report which came out in early 1978, and 1) endorsed the existing approach to safety as adequate, and 2) weakly supported any research and development on means to cope with core melt.

Chairman Zabel advised the Regulatory Staff that the present ACRS feeling was that the Diablo Canyon reactor at the Zion site was not adequate, and it would not be approved without additional safeguards. The ACRS identified several areas of interest to be discussed at the next Zion Subcommittee meeting.

During March, April, and May, 1968, the Metropolitan Siting Subcommittee, chaired by Dr. Monson, held three meetings during which the primary emphasis was on the development of a basis for comparing the population characteristics of sites, and the determination of the relative population characteristics of Zion and Indian Point 2, although Bolsa Island ws also a recurring consideration. The bases considered did not place much emphasis on the MCA. Rather, more serious accidents involving lethal doses beyond the site boundary were postulated, and alternative methodologies for estimating the integrated consequences considered.

There were a wide range of approaches considered and a wide range of opinions expressed. For example, at the meeting on March 29, 1968, member Hendrie pointed out that the concern regarding metropolitan siting was the subjecting of a large number of persons to a small probability of a severe accident. He believed that this concern leads to the need to consider the population distribution in angular sections. It was his opinion that a site with a certain population density per square mile equally distributed around the site is not significantly different from one with a 45^{o} sector with the same population density per square mile and few or no persons located in the other 315^{o} sector. On the other hand, Dr. Monson thought

*See appendix for additional discussion

the site with persons located only in a 45^{o} sector would be eight times better than a site with the same population density all around the site.

There was considerable discussion on whether to include meteorology into the comparative site evaluation, and the consensus was not to.

The minutes of the Subcommittee meeting on April 18, 1968, show a difference of opinion between consultant Ergen and member Monson. Ergen saw little difference between the more populated Burlington site and Indian Point, if one postulated a massive release of radioactivity. Monson thought there would be a definite difference in this regard.

When comparisons were made of Indian Point and Zion, on an integrated population basis, Zion appeared to be slightly better. If only the worst sector was used, Zion was "better" than Indian Point from 1 to 4 miles, but worse between 4 to 20 miles.

By the May 8 1968 Subcommittee meeting, a methodology which weighted persons close in more importantly, and looked at both sectors and integrated population, had been chosen for interim purposes. The Subcommittee concluded that Zion was slightly better than Indian Point from the population point-of-view (and hence acceptable).

The strong precedent set by the original acceptance of Indian Point 1 in 1956 and Indian Point 2 in 1966 was clear. Only now and then do the minutes record an opinion by a member expressing unhappiness with the earlier acceptance of Indian Point 2. And rarely does there appear the opinion that there should be a re-examination of the acceptability of the Zion site, even if it is equivalent to or "better" than Indian Point 2.

Although major accidents were postulated for use in making comparisons of different sites, no quantitative estimate of the probability of such large accidents or of the risk (a summation of the product of probability and consequences over all accidents) was available. Nor did there exist a quantitative risk acceptance criterion (except that indirect one that might be derived from the AEC Commissioners opinion on Malibu, that even though a fault had not moved in the last 14,000 years, displacement along the fault should be considered in the design of that proposed reactor).

At the 98th meeting, June 5-8, 1968, the Siting Subcommittee reported its findings to the full Committee, and the ACRS concluded that the Zion site ws comparable to Indian Point and acceptable. Considerable discussion was held with the Applicant concerning various safety matters, with emphasis on the possible need for a system to cope with vessel failure from thermal shock. The Applicant agreed to include provisions in the design which would make it feasible, at a later date, to add a slow flooding capability for the vessel cavity. Whether Zion included substantive additional safety features compared to Diablo Canyon is difficult to ascertain.

The ACRS agreed at the June 1968 meeting that it would prepare a letter favorable to construction of the Zion Station, with one member indicating he would have additional remarks which were aimed primarily at future reactors proposed for populated sites similar to or worse than Zion.

Extensive and controversial discussion ensured within the Committee, including a) the propriety of adding remarks concerning future reactors and b) the possibility that the Committee would adopt a position similar to the remarks with regard to more populated sites than Zion-Indian Point, and write a general letter on the subject.

Discussion of the matter was carried over to the 99th meeting, July 11, 12, 13, and 21, 1968, at which time the Committee decided to agree on issuing only a report on the Zion reactor, which is duplicated on the following pages. Also duplicated on the following pages is a personal letter from member Hendrie to member Okrent.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 24, 1968

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ZION STATION UNITS 1 AND 2

Dear Dr. Seaborg:

At its ninety-ninth meeting, July 11-13 and 21, 1968, the Advisory Committee on Reactor Safeguards completed its review of the application by the Commonwealth Edison Company for authorization to construct nuclear generating Units 1 and 2 at its Zion Station in Zion, Illinois. This application was considered also at the ninety-sixth, ninety-seventh, and ninety-eighth meetings, on April 4-6, 1968, May 9-11, 1968, and June 5-8, 1968, respectively. Members of the ACRS visited the site on June 6, 1967, and Subcommittee meetings were held at the Argonne National Laboratory on March 21, 1968, and in Washington, D. C., on April 17 and May 29, 1968. During its review, the Committee had the benefit of discussions with representatives of the Commonwealth Edison Company and their consultants, with the Westinghouse Electric Corporation, and with the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents referenced in this report.

The Zion Station is located on the west shore of Lake Michigan in Zion, Illinois. Zion has a population of 14,000, and Waukegan, Illinois, with a population of 65,000, has its nearest boundary 3.6 miles from the site. The site comprises 250 acres.

Each of the two 3250 MWt pressurized water reactors is similar in design to the Diablo Canyon reactor. The containment for each reactor is a prestressed concrete vessel similar to previously approved designs (e.g., Turkey Point, Palisades, and Point Beach). The reactors to be built at the Zion Station are the largest reactors reviewed to date for construction in a region of relatively high population density.

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The applicant has considered the possibility of reactor vessel failure as a-result of thermal shock caused by emergency core cooling system action in the unlikely event of a loss-of-coolant accident during the later portions of vessel life. He has conducted engineering studies which have established the feasibility of a cavity flooding system that could flood to a level above the top of the core and thereby provide additional protection in the event of such failure. He stated that this system would be installed at a future time if studies now under way indicated that vessel failure as a result of thermal shock could occur. The present design provides for reactor cavity flooding to about two feet above the bottom of the core. Additionally, the reactor cavity has been designed, as at Indian Point 2, to limit vessel movement in the highly unlikely event of failure of the reactor vessel by longitudinal splitting during operation. The Committee continues to favor such protection for large reactors in regions of relatively high population density.

The applicant has proposed using signals from the protection system for control and override purposes. The Committee reiterates its belief that control and protection instrumentation should be as nearly independent of common failure modes as possible, so that the protection will not be impaired by the same fault that initiates a transient requiring protection. The applicant and the AEC Regulatory Staff should review the proposed design for common failure modes, taking into account the possibility of systematic, non-random, concurrent failures of redundant devices, not considered in the single-failure criterion. In cases where hypothesized control or override failure could lead to the need for action by interconnected protection instrumentation, separate protection instrumentation channels should be provided or some other design approach be used to provide equivalent safety.

The applicant described programs for development and utilization of instrumentation for prompt detection of gross fuel failure and for detection of primary coolant leakage.

The Committee continues to emphasize the need for quality in the manufacture, storage, and installation of the reactor and primary system components. The applicant described the quality assurance program that he and his contractors intend to carry out for this purpose. In this connection, the applicant described the testing program for engineered safety features, including a full flow test of the emergency core cooling system delivering water to the reactor vessel. The Committee recommends that the applicant give further consideration to testing the containment spray systems with full flow to the spray nozzles at least once at an appropriate time during construction.

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The applicant described his emergency plans for the Zion Station, which are based partly on experience acquired in developing plans for the Dresden Nuclear Station.

The Committee continues to call attention to matters that warrant careful consideration with regard to reactors of high power density and other matters of significance for all large, water-cooled power reactors. In addition, attention is called to safety-related questions specifically identified for the Diablo Canyon reactor class. The applicant reviewed his research and development program designed to resolve safety-related problems and stated that he expects resolution of these problems before operation of the reactors. System modifications or restrictions on operation may be appropriate if the startup program, additional operating experience, or the research and development should fail to confirm adequately the proposed safety margins.

The Committee believes that the items mentioned can be resolved during construction and that, if due consideration is given to the foregoing, the nuclear Units 1 and 2 proposed for the Zion Station can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Additional remarks by Dr. David Okrent are appended. The matters discussed by him were considered by the Committee during its meetings. The Committee believes that the status of these matters, as they pertain to the Zion units, is satisfactory.

Sincerely yours,

/s/
Carroll W. Zabel
Chairman

Attachments:
1. References
2. Additional Remarks of
   Member David Okrent

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References - Zion Station

1. Letter from Commonwealth Edison Company, dated July 12, 1967; Application for Construction Permit and Operating License; Volumes I and II of Preliminary Safety Analysis Report, Zion Station

2. Letter from Commonwealth Edison Company, dated August 15, 1967; Amendment No. 1 to Application

3. Letter from Commonwealth Edison Company, dated November 28, 1967; Amendment No. 2 to Application; Volumes III and IV of PSAR

4. Letter from Commonwealth Edison Company, dated December 20, 1967; Amendment No. 3 to Application.

5. Letter from Commonwealth Edison Company, dated January 29, 1968; Amendment No. 4 to Application; Volume V of PSAR

6. Letter from Commonwealth Edison Company, dated March 1, 1968; Amendment No. 5 to Application

7. Letter from Commonwealth Edison Company, dated April 4, 1968; Amendment No. 6 to Application

8. Letter from Commonwealth Edison Company, dated April 17, 1968; Amendment No. 7 to Application

9. Letter from Commonwealth Edison Company, dated May 3, 1968; Amendment No. 8 to Application

10. Letter from Commonwealth Edison Company, dated June 6, 1968; Amendment No. 9 to Application

11. Letter from Commonwealth Edison Company, dated June 27, 1968; Amendment No. 10 to Application

July 24, 1968

Additional Remarks of Member David Okrent

While I am not objecting to a construction permit for the Zion reactors, I am suggesting that in connection with its issuance there are certain matters that warrant consideration and resolution before construction is completed.

In its report of November 24, 1965, on reactor pressure vessels, the ACRS recommended that further attention be given "to methods and details of stress analysis, to the development and implementation of improved methods of inspection during fabrication and vessel service life, and to the improvement of means for evaluating the factors that may affect the nil ductility transition temperature and the propagation of flaws during vessel life". The ACRS also recommended that "means be developed to ameliorate the consequences of a major pressure vessel rupture" and suggested as a possible approach the provision of "adequate core cooling or flooding which will function reliably in spite of vessel movement and rupture". The ACRS went on to state that "the orderly growth of the industry, with concomitant increase in number, size, power level and proximity of nuclear power reactors to large population centers will in the future make desirable, even prudent, incorporating in many reactors the design approaches whose development is recommended above".

Since November, 1965, considerable additional emphasis has been placed by the nuclear industry and the AEC on providing still greater quality in pressure vessel fabrication. An important research program is under way by the AEC to provide a better understanding of the behavior of thick-walled, steel pressure vessels. Our reactor vessel operating experience, although limited, has been good.

On the other hand, some questions have arisen in connection with specific design and fabrication aspects of pressure vessels. Resolution is required concerning the potentially adverse effect on vessel integrity of thermal shock arising from operation of the emergency core cooling system in the unlikely event of a sizable primary system leak, and questions exist with regard to the behavior of highly irradiated, thick section, pressure vessel walls in the presence of flaws and at significant vessel pressure.

Increasing attention has been given to the development of in-service inspection techniques and to the provision during reactor design of the necessary accessibility for thorough in-service inspection. Both industry and AEC regulatory groups are currently working on access and periodic inspection requirements for water reactor primary systems, including the pressure vessel. Means of remote, volumetric inspection of pressure vessels in service are under development by the nuclear industry, as are other flaw detection devices.

I believe that, with regard to water reactors of current design to be sited in less populated areas, the efforts under way to provide improved vessel quality and adequate, thorough, in-service inspection, in conjunction with satisfactory resolution of the thermal shock matter, with acceptable results from safety research programs on irradiation effects, subcritical flaw growth, etc., in thick-walled vessels, and with deliberate conservatism and thoroughness in pressure vessel design and fabrication practice, should provide an acceptable basis for dealing with safety questions arising from pressure vessel integrity.

The Zion site has a relatively large surrounding population density. For large water reactors proposed for such a site, I believe that, in addition to the above steps, careful consideration should be given in the initial engineering design to provision of the capability to cope with a loss in primary system integrity arising from a leak or split in the pressure vessel wall. Such provisions should include necessary steps to maintain the containment integrity. It appears likely that means to maintain the general core geometry and to provide the necessary emergency cooling water would be required. It is important that such provisions, if they are to be implemented, provide a significant degree of additional protection, albeit not perfect or complete, and that they should not, of themselves, provide a means of detracting from the integrity of the pressure vessel. It is to be expected that the development of means to deal with a loss of primary system integrity arising with the pressure vessel will be a process of evolution. Careful and thorough study should lead to a definition of those potential areas of degradation in pressure vessel integrity for which protective meausres are practical and appropriate. In view of the very low probability of a pressure vessel rupture, the design of these protective features could be based on fairly realistic rather than highly conservative analyses. A reliance on off-site power sources in connection with these protective features may be acceptable, if the capability of the external power system to withstand sudden, unexpected shutdown of the reactor can be clearly demonstrated and periodically verified.

For the Zion reactors, where the engineering design is now well along and could not be readily modified without major delays and significant additional costs, I believe that the applicant should study what provisions could be made, within the limitations of the existing design, to provide further protection against a loss in primary system integrity arising from a limited size leak or split in the pressure vessel wall, particularly in the region that receives the highest neutron irradiation dose during reactor lifetime.

I also believe that, at this time, additional conservatism in design, construction and operation is desirable for the Zion reactors, as compared to similar reactors at less populated sites. To be most effective, this additional conservatism should be part of the applicant's basic philosophic approach. The following aspects might be included:

1. Both for the primary coolant system and for other features of vital importance to the protection of the health and safety of the public, additional conservatism in design and further steps to assure quality of construction and continued integrity and reliability during operation should be used, where practical.

2. Safety issues remaining to be resolved between the start of construction and the initiation of operation at power should be minimized; well-defined research and development programs, adequate to clearly resolve the issues in timely fashion, should be committed. Where questions remain to be resolved, and where complete resolution may not be accomplished by the time of reactor operation, the reactor design should proceed on the basis of incorporating the appropriate safety provisions.

3. Since it is highly unlikely that a clear demonstration of the efficacy of the several engineered safety systems and other protective features under representative accident conditions will occur as a consequence of actual accident experience in the reasonably near future, it is desirable that extra margins be provided in the design of the usual engineered safety systems, particularly those for which some degree of uncertainty or some problem requiring resolution remains.

4. Additional, detailed examination of potential accidents leading to moderate releases of radioactivity to the environment (small accidents) should be made, and steps be taken to reduce still further the probability of occurrence of such accidents.

In my opinion, additional steps such as these, which are taken to protect the health and safety of the public with regard to reactors to be sited close to population centers, need not necessarily be applied to reactors in less populated sites.

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.
UPTON, L.I., N.Y. 11973
TEL. AREA CODE 516 YAPHANK 4-6262

REFER:

July 16, 1968

Dr. David Okrent
Senior Physicist
Laboratory Director's Office
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Dear Dave:

Since everybody else is giving you advice on the Zion matter, I don't see why you should be spared my two cent's worth. I've been trying to answer, for myself, three questions about your additional remarks on Zion. I think I have, and the answers may be of interest to you.

I ask myself:

a) Do I now approve of, or object to your additional remarks?

b) What is the reason for the answer to a), and why, since I strongly support your position in general, do I refuse to join you in these additional remarks?

c) What position do I take on Zabel's move to prevent your (and Mangeldorf's) additional remarks from being attached to the Zion letter?

The answer to a) is that I object, in a very mild way, to your additional remarks as they now stand. The answer to b), the reasons for objecting, and for not joining you in the remarks go as follows. First of all, the remarks as written seem to have a very limited applicability to the Zion application itself. I know you don't agree, and the fact that we differ on this matter is a principal reason for this lengthy discourse.

As I read the additional remarks, they say for Zion,

1) do what is practicable (and reasonable?) within the limits of the existing design to deal with vessel leaks and ruptures, and

2) take the steps #1 through #5 listed on pages 6-7 of Draft 2 (attached to letter Draft 7).

But since you do <u>not</u> oppose the Zion application, which your first paragraph specifically says, and since the application does not now include these items, I must conclude that these items do <u>not</u> really apply to Zion in your view. (As a side note, the steps 1-5 you list are only inferred to be appropriate for Zion through the reference, middle of page 6, to sites with population densities equal to, etc., Zion. I think some, maybe most, people would conclude the steps 1-5 do <u>not</u> apply specifically to Zion.)

Now I know the syllogism above will be offensive to you (note Webster's alternate definition of syllogism as a "...specious or crafty argument."), but it is a difficulty for me, and for others too, I am sure. From my standpoint, the additional remarks would be much clearer and would squarely meet the test of applicability to Zion if they said something like:

"I believe that the applicant should, in addition to the matters cited in the ACRS report, 1) do what is practicable within the limits of the existing design to deal with pressure vessel leaks and ruptures, (etc.), 2) take the steps enumerated below (steps 1-5, etc.)."

In this form the additional remarks would be a clear dissent on the Zion application approval. They might be tempered by a remark that these matters are resolvable during construction, in your view.

I know the above introduces an "applicability" criterion for additional remarks with which you may disagree. But I do think some of the Committee reaction to your remarks is brought on by such feelings about applicability. (Later on, in discussing the answer to my third question, I will tell you what I think of such a criterion.)

There remains the question of how to treat the matters in your additional remarks pertaining to future reactors, and this raises the second reason for my objection to the remarks as they now stand. Basically this is that you are preempting the Committee in two important areas: first in the area of requiring protection against vessel failure, and second, in the area of Metropolitan Siting, where you suggest that steps along the lines you suggest might open the way to sites in high population areas. I don't think I am being completely rational with this objection, because you have given the Committee ample time to consider your point of view, and if they will not agree with it, you have a right and a duty to express it individually. But I am frustrated by a) agreeing with you, b) being personally unwilling to join in these remarks on the Zion letter, and therefore, c) being unable to participate in the expression of views on vessel failure protection. So, rational or not, I wish you were not going to put the "non-Zion" sections of your remarks into the additional remarks on Zion. (I think Palladino and some others share this feeling.) Furthermore, I have the feeling that the presence of these remarks, appended to the Zion letter over the protests of many Committee members, may really hamper future acceptance by the majority of the position that protection against vessel failure should be provided at poor sites.

If you were to ask what I suggest doing (obviously I am going to tell you whether you want to know or not), it is as follows:

A) Confine additional remarks for the Zion letter to those areas listed as 1) and 2) above,* and consider whether they should not have the more direct dissent form suggested above.

* previous page, 2nd last paragraph.

B) Work with the Committee toward incorporating the "non-Zion" portions of the present remarks in a general letter. Since this is unlikely to produce the desired general letter in the near future, I suggest it only as a prelude to an individual letter to Seaborg on vessel failure in which I would join you, and in which some others might join, notably Palladino. Your current drafts of such a letter would suit me just fine, and we could allow the rest of the Committee to make "additional remarks" if they wanted to.

Now as to the answer to my third question to myself, about Zabel's attempt to deny your additional remarks a place with the Zion letter. I have been trying to think of a reasonable basis for limiting the scope of individual remarks on case letters. I conclude there is none. Even the test of applicability of the remarks to the case at hand, which I do apply for myself and which is the central reason I do not join with you on the Zion remarks, I would not want as a general rule for the Committee. The problem with such a general rule, or any general rules related to individual remarks, is that the majority will have to decide how to apply the rule in each specific case, and the nature of the situation is that the majority is automatically prejudiced against the individual remarks. So I am against Zabel's move, and I will try hard to prevent anything along that line.

I have tried to think what I might do, if I were in your position and were denied the Zion letter as a vehicle for my individual views. Being unfortunately inclined to emotional outbursts when frustrated, I expect I would resign with a fiery letter to Seaborg, and would subsequently regret it. I judge you to be considerably more self-possessed than I am, and if Sunday's meeting goes badly, I will count on you to fall back on the general letter approach and on a joint individual letter on the vessel failure subject if necessary.

Best regards,

Joe

Joseph M. Hendrie
Associate Head, Engineering Division
Nuclear Engineering Department

JMH:ecd

APPENDIX

On the Removal of the Core-Catcher from Indian Point 2

A very detailed description of the difficult and very controversial discussion within the ACRS concerning core-catchers in general, and their potential applicability to Indian Point 2 and Zion 1 and 2, in particular, can be found in the minutes of the 95th meeting, March 7-9; 1969, 96th meeting, April 4-6, 1968; 97th meeting, May 9-11, 1968; and 98th meeting, June 5-8, 1968.

At the 95th meeting, AEC Chairman Seaborg expressed his concern to the ACRS about piling one safeguard on to back up another, referring to the core crucible at Indian Point 2. At the same meeting, Consolidated Edison came to see the Committee, nominally to report on its ECCS in accordance with the August 16, 1966 ACRS letter, but more specifically to get ACRS approval of removal of the core catcher which was formally included in the plant in June-July, 1966.

A very complex discussion ensued. The applicant stated they had done only one to two man years of work on the core-catcher, and some members expressed disappointment that so little effort had been made to examine its possibilities, despite the admittedly difficult problems involved. Some members felt that a decision on the core-cathcer should be related to how effective the new ECCS really was, a matter not yet evaluated. Others were in favor of dropping the core catcher as having no value, independent of other considerations.

The applicant first said he planned to drop the core-catcher, but then said it was not his intent at that time to request ACRS approval for such action.

The Regulatory Staff basically treated the core-catcher as if it did not exist, even on paper.

The ACRS finally took no action at the March meeting.

At the April meeting, both Zion and Indian Point 2 were discussed. Mr. Price said that the Regulatory Staff did not know how to design a core-catcher and so he did not see any need for it. Dr. Morris of the Staff stated the Zion site was acceptable without a core-catcher; however, if a core-catcher ws required for Indian Point 2, then it would have to be included for Zion.

In preliminary vote in Executive Session, one member favored a core-catcher for Zion, eight did not, while six abstained.

At the same meeting, Mr. Price reported that he had received a phone call from Consolidated Edison Company reporting that they had decided to include a core-catcher for Indian Point 2. The reasons for the decision were complex. Apparently, the applicant was reluctant to initiate a formal request for its removal. Some ACRS members thought that the tone of the questions in the March meeting may have suggested an ACRS position in favor of the proposed device.

The discussion and arguments over a core-catcher for Zion and Indian Point occupied much of the May meeting. Three members felt so strongly about the issue that they implied they would write their own letters that month against a core-catcher for Indian Point 2, if the Full Committee failed to act. The deadline for action was later withdrawn when it was agreed that the matter would be given priority at the June meeting and resolved for both Indian Point and Zion.

At the June meeting, the issues discussed on Zion were not related to a core-catcher, which the ACRS agreed they would not recommend, but on the measures included in the design to enable a later addition of a cavity flooding device intended to assist in keeping the core cool, should thermal shock lead to a reactor vessel leak following a LOCA, and on additional remarks by one member dealing with other matters. In effect, the Applicant, by provision of the capability for cavity flooding, had added a special site-related safety feature, albeit nothing approaching a core-catcher in its objective.

With regard to Indian Point 2, the ACRS reached the position (with one negative vote) that "the core-catcher was not an essential engineered safety feature for this rector", but did not request that it be removed from the design. The Regulatory Staff and Applicant were advised orally of this decision. The Applicant indicated that construction of the core catcher would be terminated and that it did not need a letter report from the ACRS on the matters discussed.

During the series of meetings, there had been indications to the Applicant that, were the ACRS to write a report, it might have to include consideration of the asterisked (or generic) items which had arisen since its August, 1966 report on Indian Point 2. Hence, for this and other reasons, the Applicant's final decision to ask that no ACRS letter be written at that time was understandable.

The ACRS asked that the Final Safety Analysis Report be submitted as soon as possible, so that the Applicant's final decision to remove the core-catcher would be made public as soon as possible, since there was to be no letter on the matter, and at that time ACRS meetings were not public.

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At the June, 1968 meeting in which the ACRS reached a consensus on the accepability of Zion, the Committee also decided it would advise Consolidated Edison that the ACRS saw no need for the core-catcher to be kept in the Indian Point 2 reactor design (and, implicitly, was therefore avoiding the Regulatory Staff requiring it for Zion). This rather difficult decision was clearly based on different reasons for the various Committee members. Some felt it was unnecessary. Some would have liked a core-catcher (or some other means to maintain containment integrity or mitigate the effects of core melt), if they had thought that a practical design was available. However, the applicant had done essentially no further analysis or development beyond the very modest amount performed in 1966, and the AEC had not pursued an independent R&D program.

At the same June, 1968 meeting the ACRS heard a first presentation by Consolidated Edison for some proposed sites for 2 large BWR's, the sites being somewhat more densely populated than Indian Point..

At the 99th meeting, July 11-13, 21, 1968, there was some discussion between the ACRS and Mr. Price concerning the Bolsa Island review. In addition to population density questions, seismic design was a major question for this proposed facility, and the Bolsa Island Project had taken the somewhat unusual step of appointing a "blue ribbon panel" to make recommendations as to what constituted adequate seismic design.

During the same time period the Regulatory Staff and the ACRS had worked on the development of seismic and geological siting criteria, and a draft had been agreed to by the ACRS some months earlier. Its release in the Federal Register for public comment had been held up by Mr. Price, however, while awaiting comments from Mr. Milton Shaw, Director of the Division of Reactor Development, AEC.

At the 99th meeting, Mr. Price was asked if the seismic criteria would be put out for public comment in the near future. Member Okrent pointed out that these criteria may be connected with Bolsa Island, although the review could probably be carried out without them. Mr. Price felt that it would not be wise to judge this or any other facility on the basis of criteria which had not yet been issued. He stated frankly that the delay in issuing the criteria was partly due to the fact he wanted to know whether or not they would rule out the Bolsa Island site.

The Regulatory Staff had submitted a non-commital report on Bolsa Island to the ACRS dated July 8, 1968 just before the 99th meeting. It summarized the recommendations of the "Blue Ribbon Panel" and gave some population comparisons with Indian Point, Zion and Burlington.

There was no discussion of this report at the July, 1968 ACRS meeting. At the 100th meeting, August 8-10, 1968, ACRS Chairman Carroll Zabel advised the Committee that Mr. Price had called him to advise the Committee that the Bolsa Island project had been terminated because of rising project costs.

Interestingly, a year earlier, Dr. Donald Hornig, the Science Assistant to the President, had sent the following letter on the next page to Chairman Seaborg of the AEC, and received the response indicated.

However, no request was made by the AEC to the ACRS for an early recommendation concerning the suitability of the Bolsa Island site, although the population information existed. And the Regulatory Staff refrained from completing any position on its suitability in this regard.

At the 100th meeting, August 8-10, 1968, the ACRS reviewed Consolidated Edison's proposed Trap Rock, Montrose, and Bowline sites, all in the general vicinity of Indian Point but each having its own demographic characteristics. The Metropolitan Siting Subcommittee had reviewed the three sites for population considerations at a meeting held August 7, 1968 and, using a comparative approach based on estimating off-site casualties and latent cancer effects on an average, rather than worst sector basis, concluded that the Trap Rock site was fairly similar to Indian Point. The Subcommittee recommended to the full Committee that the Trap Rock site was not unacceptable on the basis of population alone, and that the ACRS not take a position on Montrose or Bowline which were both less desirable than Trap Rock.

Consolidated Edison had proposed a secondary containment having a design pressure of 10 or 15 psi, instead of the normal, "no-strength" containment building placed around the pressure suppression containment system of a BWR. A cryogenic off-gas system to trap and store noble gas fission products normally released from routine effluents was also proposed.

In its report to the ACRS of May 31, 1968, the Regulatory Staff stated "These sites are somewhat worse demographically than Indian Point 2; however, we believe they are in the Indian Point 2 class of sites. We do not consider them in the Burlington class of sites." The Staff went on to say, "we anticipate the applicant would need to put more emphasis on design and performance adequacy of the containment and engineered safety features than, for example, was required at Peach Bottom." Presumably, this Staff emphasis was in terms of meeting the guidelines of Part 100 for the MCA and other "design basis accidents."

The Committee discussion at the 100th meeting shows a very considerable divergence of opinion. Some members felt the three sites were relatively indistinguishable, in contrast to the Subcommittee conclusion. Some felt all might be acceptable for a PWR like Indian Point 2, although there was concern for a BWR with the primary system extending outside containment. At least one member stated that he did not like the Indian Point 2 site, felt he had been taken in when

2-325 DR-1376

THE WHITE HOUSE

WASHINGTON

August 25, 1967

Dear Glenn:

It has been obvious from the beginning that the site proposed for the large nuclear power and water desalting plant in California (the MWD project) could present a very serious safety and licensing problem, because of its close proximity to a large population center, its possible seismic implications, and novel features of the man-made island. We have all hoped, of course, that these problems could be resolved before any substantial amounts of money were irrevocably committed to the project.

May I inquire about the present status of these considerations? Is any thought being given currently to possible alternative sites? In this connection, I understand that a proposal to build a nuclear power plant near Burlington, New Jersey was recently turned down on grounds of proximity to population, although this problem appears at first glance to be less serious in the Burlington case than it would be in Los Angeles.

Sincerely,

Don

Donald F. Hornig
Special Assistant to the President
for Science and Technology

The Honorable Glenn T. Seaborg
Chairman
Atomic Energy Commission
Washington, D.C.

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September 7, 1967

Dr. Donald F. Hornig, Director
Office of Science and Technology
Executive Office of the President
Washington, D. C. 20506

DR. 1376

Dear Don:

Commissioner Ramey has informed me that he has been in touch with Bob Barlow about your letter of August 25 concerning the island site proposed for the large nuclear power and water desalting plant (the MWD project) and that it has been decided that an oral briefing would provide the best mechanism to supply you with the information you requested.

Mr. Barlow has suggested 10:00 a.m. on October 11 as the first time which appears to suit both of our schedules. I am looking forward to meeting with you at that time.

Cordially,

(Signed) Glenn T. Seaborg

Glenn T. Seaborg

RECEIVED SEP 11 3 09 ... COMMITTEE ON ... EQUIPMENT he approved it, and felt the ACRS should be more demanding now.

The Committee had available to it a long list of safety matters related to BWR's and these rather than the site characteristics, served as the principal focus for the discussion with the Applicant.

From the minutes of the Committee caucus, it is not clear that all members agreed that the Trap Rock site was acceptable. However, a Committee position was adopted, and subsequent to the Committee caucus, the ACRS Chairman read the following statement to the representatives of Consolidated Edison.

> The ACRS feels that the Trap Rock site is not unacceptable on the basis of population alone for a reactor of the approximate size proposed. The Montrose and Bowline sites seem less desirable than Trap Rock, but the Committee makes no statement now as to their acceptability. The Committee feels that considerable attention should be given to emergency plans, including evacuation, particularly in view of the high close-in population. With regard to the reactor design matters discussed thus far, the Committee continues to be concerned with the fact that a part of the primary system is outside containment, and will watch the detailing of the steam line break accident and associated matters with great interest. The Committee emphasizes the importance of in-service primary system inspections and of the provisions for adequate access for such inspections.
>
> The Committee feels that at difficult sites such as these, more careful study and evaluation of the problems associated with possible reactor vessel splitting are needed. Other items noted in previous Committee letters will be of even greater interest in this case, in view of the difficulties of the site. The Committee feels that a more conservative approach on the part of the applicant, leading to fewer items to be resolved after the construction permit stage, is appropriate for a plant at this kind of site. The Committee believes that the double containment and the off-gas systems are useful approaches.

Not too long after the 100th ACRS meeting, Consolidated Edison advised the Regulatory Staff that their schedule for these two proposed BWR's had become somewhat indefinite, and, in fact, a construction permit review was never completed.

However, at Trap Rock, as at Indian Point and Zion before, the ACRS was continuing its emphasis on trying to get a higher level of safety at poor sites, either by decreasing the probability of accidents or by improving the measures to deal with accidents. This was in contrast with the Regulatory Staff which continued to make adherence to Part 100 the only aspect of safety which was dependent on population density for an accepted site.

As it turned out, by its 100th meeting the ACRS had dealt with all but one of the most populated sites it would review through the end of 1977. namely Newbold Island, although Midland and Limerick would also pose some population-density-related questions.

In the months following the site review of the Trap Rock, Bowline and Montrose sites for Consolidated Edison in August, 1968, the ACRS Metropolitan Siting-Site Evaluation Subcommittee continued to work on an improved comparative site population index and on possible criteria for reactors at sites worse than Indian Point - Zion. At a meeting on December 4, 1968, the Site Evaluation Subcommittee heard a presentation by representatives of the Atomic Industrial Forum (AIF) concerning their viewpoint that there was a need for metropolitan sites for nuclear reactors. In particular, the AIF representatives emphasized the very high cost of transmission lines.

With regard to the development and choice of a site population index, the Subcommittee examined a host of potential approaches and finally adopted a relatively simple, arbitrary method of comparing other sites to a synthetic site having the average characteristics of Indian Point and Zion; and the ACRS used the method to provide itself with a rough comparative population index for all sites.

The site population index (SPI) changed somewhat over the period of time it was used. A later version was described by ACRS consultant (and former member) W. Ergen in a memorandum to file, as follows:

> The Site Population (or SPI) index considers the number of people who have to be evacuated as a consequence of the postulated accidents, and the exposures resulting from these accidents. It is understood that other protective measures might be substituted for evacuation. An individual is assumed to have to be evacuated if he otherwise would receive more than 25 R. In counting the man.rem of exposure, the index does not include the man.rem that would be received by evacuees and the doses below 1 R per person.
>
> The smallest accident considered causes 25 R at 0.5 miles, the largest accident considered 25 R at 6 miles. Between these two limiting accidents, there is a continuance of accidents, the probability of an accident being independent of the size of the release. The dose decreases proportionally to $r^{-1.5}$m where r is the distance from the reactor. In the largest accident considered, the dose at 50 miles is $25 \cdot (50/6)^{-1.5} = 1.039$ R. This was chosen as the cut-off point, see above, so that people outside the 50 mile circle do not have to be considered.

The reference site has a population density, which is, at every point, equal to the arithmetic mean of the population densities at the corresponsing points of the Indian Point and Zion sites.

For each accident considered, the number of evacuees is found for the site under study, and for the reference site. The ratio of these two numbers is averaged over all accidents considered. Likewise, for each of these accidents, the exposures are found for the site under study and the reference site, and the ratio between these two numbers are averaged. The arithmetic mean of the two averages is 1 if the reference site is taken as the site under study. For any other site, the deviation of the mean from 1 is a measure of the population distribution; a number much less than unity would correspond to a relatively low population density.

The index originally postulated 500 R at 1 mile and successive distances up to 6 miles; this was changed to the 25 R limit in Part 100 by introducing a scaling factor of 20.

The method uses average rather than worst sectors and leads to the following relative weighting factors:

Distance from Reactor, miles	Weighting Factor
0.75	4600
0.75-1	2400
1-2	1250
3-4	535
5-7	100
10-15	24
20-25	7
30-35	2.3
40-45	0.6

Compared with recent studies by the Reactor Safety Study Group (WASH-1400, 1974) the SPI weighting factor described in the Ergen memo falls off very rapidly at large distances, and possibly places more emphasis on nearby population.

The Site Evaluation Subcommittee also continued to work on a possible set of general criteria or requirements for reactors to be considered for sites having population densities somewhat greater than Indian Point - Zion. At the 111th meeting, July 10-12, 1969, the full Committee decided to forward a proposed ACRS report on this subject to Mr. Price and the AEC Commissioners. The draft report, which included additional remarks by ACRS member Hanauer and was never formally sent to the AEC, is reproduced on the following page.

COPY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: LOCATION OF POWER REACTORS AT SITES OF POPULATION DENSITY GREATER THAN INDIAN POINT-ZION

Dear Dr. Seaborg:

As the use of nuclear power has grown, water-cooled power reactors of progressively larger size and reactor sites of increasingly higher population density have come to be employed. Simultaneously, progress has been made in improving the design and construction of such reactors so as to reduce the already low probability of occurrence of accidents and mitigate further any potential accident consequences. Although this progress has been reasonably commensurate with the increases in reactor size and population density, the Advisory Committee on Reactor Safeguards believes that additional steps are necessary to justify the use of sites more densely populated than the Indian Point-Zion type of site. The Committee believes that operation of large water-cooled power reactors at sites of somewhat greater population density may be appropriate if:

(a) Prior to the time of receiving a construction permit, at least one year of satisfactory operating experience has been obtained with a reactor of generally similar design, power density, and power rating;

(b) Prior to the time of starting power operation, at least ten reactor-years of satisfactory operating experience have been obtained with reactors of essentially the same design, power density, and power rating;

and if the measures described below, additional to those required for the Indian Point-Zion type of site, are adequately effected.

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1. The containment system should be designed to reduce to substantially lower levels the off-site radiation doses in the unlikely event of a major reactor accident, and should be capable of maintaining this level of protection even with substantial degradation of the system. These more stringent requirements for the containment system are appropriate for higher population density sites, because the present guidelines assume that in an emergency members of the public in the low population zone can be evacuated or otherwise protected in timely fashion. For sites of higher population density, evacuation or other effective protective measures for the close-in population are less certain to be achieved in the short times required, and it is necessary that the containment system provide a greater degree of protection.

2. Increased emphasis should be placed on detailed consideration of possible accidents leading to small or moderate releases of radioactivity to the environment, and means should be provided to reduce still further the probability of occurrence of, and the consequences of, such accidents. In particular, increased attention should be given to potential radwaste accidents. Similarly, current practices related to fuel handling, storage, and shipping should be re-evaluated and changes implemented wherever found appropriate to increased safety.

3. Further reduction in the already small releases of radioactivity from routine plant operation should be effected.

4. The number of safety issues remaining to be resolved between the start of construction and initiation of operation at power should be minimized. Where it appears that resolution of a safety issue may not be accomplished by the time of start of reactor operation, the plant design should incorporate whatever alternative features are necessary to provide adequate protection. An example of such a feature is the provision of permanent in-core instrumentation for use in the event that out-of-core instrumentation should not prove adequate.

5. Because of the small likelihood that proof of the efficacy of engineered safety systems under accident conditions will be obtained as a consequence of actual accident experience, extra margin should be provided in the design of these systems wherever

such provision is practical and will clearly improve safety. As an example of extra margin, additional emergency core flooding capability might be appropriate.

6. Steps should be taken during the design of the reactor plant to provide further protection in areas related to possible degradation of reactor vessel integrity, such as leaks and vessel wall ruptures. The protective features provided should be of such design as to preclude their jeopardizing vessel integrity.

7. Additional consideration should be given in the design of the plant to protection against damage by missiles.

8. Greater assurance of maintenance of integrity of any portions of the primary system outside the containment, and appropriate additional means for coping with possible loss of their integrity, should be provided.

The ACRS emphasizes again the vital importance of quality assurance, and the necessity for adequate consideration of diverse and independent means of protection against common failure modes in safety systems.

The Committee believes that realization of item (a) and demonstration of reasonable assurance of realization of item (b), together with adequate implementation of items 1 - 8, could provide a basis for considering applications for construction permits for large water-cooled power reactors at sites of somewhat greater population density than that of the Indian Point-Zion type of site (e.g., approximating that of the Burlington site). The Committee also believes that the additional protective features eventually resulting from these measures need not necessarily be incorporated in reactor plants either existing or yet to be constructed at sites of population density equal to or less than that of Indian Point-Zion.

Additional remarks of Dr. Stephen H. Hanauer are attached.

Sincerely yours,

/s/

Joseph M. Hendrie
Acting Chairman

COPY

ADDITIONAL REMARKS BY DR. STEPHEN H. HANAUER

In my opinion, approval of sites more densely populated than Indian Point-Zion for large water-cooled reactors should be based on verified facts, rather than reasonable assurance regarding the outcome of work not yet completed, as is appropriate for present sites, where evacuation of the surrounding population is feasible in an unforeseen emergency. For this reason, I cannot agree with the conclusion of this report, even though I concur with many of its recommendations. I do not believe that the necessary knowledge and experience are now available to support such a conclusion. It is my hope and expectation that the needed knowledge and experience will be obtained; that would be the appropriate time to consider the use of more densely populated sites, and suitable criteria for such use.

The draft letter was discussed with Mr. Price at the 112th meeting, August 7-9, 1969 and both with Mr. Price and the AEC Commissioners at the 113th meeting, September 4-6, 1969.

The summary minutes of these discussions follow below.

Summary, 112th ACRS Meeting
Meeting with The Director of Regulation

ACRS Report on Siting of Reactors at Population Densities Greater than Indian Point - Zion - Mr. Price stated that he would not be in a position to provide considered comments until the September ACRS meeting since he has not had an opportunity to discuss the subject with the Commissioners.

Some informal comments on the report were:

a. The report implies that the approval of reactor sites will be by increments rather than a step function until the Burlington type site is reached.

b. Mr. Price believes that the requirement for ten years of satisfactory operation associated with the issuance of an operating license will be difficult to implement if operation is not completely satisfactory but a large number of reactors are already built. Perhaps ten years of operation should be required for a construction permit.

c. Mr. Price stated the AEC will have to consider reducing routine release limits, a current hot issue; the AEC is vulnerable if the release limit is handled on a case-by-case basis. Since the Part 20 values are already very low, based on the best technical judgment, and the industry is already keeping releases below 1% of the permissible limit, the Regulatory Staff is considering possible alternatives. One alternative is for the plants to reduce release limits on their own, and another is to impose some design or operational restrictions on individual plants, leaving Part 20 at its current, biologically established values. It becomes complicated by the involvement of other government agencies.

d. Mr. Price believes the additional requirements for engineering safety features for metropolitan sites should not leave the impression that the safeguards for presently approved reactors are not safe. This impression could create a difficult public relations problem.

   Dr. Hanauer noted that there does appear to be a difference of philosophy between the ACRS and the AEC Staff with respect to the requirements for engineered safety features vs site features.

e. Mr. Case noted the ACRS report identified "gadgets" required but not much on operation and nothing on R & D. He believes that several significant safety research tests should be completed before approval of construction permits at more densely populated sites.

f. Dr. Morris added that industry will ask for the eight items identified in the report to be quantitative in nature rather than qualitative.

Summary, 113th ACRS Meeting Sept. 4-6, 1969
Meeting with the Director of Regulation

Proposed ACRS Report on Siting of Reactors at Population Densities Greater Than Indian Point - Zion - Mr. Price reported the status of discussion with the Commissioners on this ACRS Report. Both the Staff and the Commissioners support the intent of the letter. In fact, there is strong support for some of the requirements mentioned in the Report, and it has been suggested that the ACRS work with the Staff and nuclear industry to achieve the improvements required in these areas.

Some comments on the Report are:

a. Mr. Price believes the Report should not be issued at this time. He thinks that releasing such a report at this time would aggravate and worsen the public relations problem. He does not object to the contents of the Report as advice but does not feel it should be made public. Mr. Price does not feel that industry is placing any great pressure on the AEC to approve more densely populated sites, e.g., he does not consider the Newbold Island site as being worse than the Indian Point - Zion sites.

b. The public relations, as well as political, problem is one of convincing people near the presently approved sites that the nuclear units are adequately safe, as compared to those which could be approved at more densely populated sites if more engineering safety features, etc., are provided.

   Dr. Hanauer noted there seemed to be an apparent inconsistency in this point of view. He noted that, if industry comes forward with improved designs for more populous sites and it is required to backfit these requirements to presently approved sites, the industry will shy away from coming forth with improvements.

   Dr. Okrent noted that the principle of greater safeguards for heavily populated areas has been accepted in the location of ABM sites. Mr. Price replied that he does accept this principle, but noted that it is difficult to explain to the public.

c. The Regulatory Staff prefers meetings with the nuclear industry to have intensive discussions of the subjects which need resolution before more densely populated sites can be used for nuclear plants. From these discussions AEC criteria can be formulated for publication and public input.

Mr. Price stated, when asked, that he believes the emphasis of the meetings between AEC and industry should be on industry's input and participation to arrive at the level of design mentioned in the ACRS Report rather than on developing additional AEC criteria. Such meetings might be able to start in one month's time and would probably require a year's effort. Mr. Price stated that he felt the ACRS Report could be written after these meetings were held. The Committee decided to establish a working group to formulate a plan for the series of meetings with industry suggested by Mr. Price. (Dr. Hanauer has assigned this task to the Subcommittee on Siting Evaluation.)

d. Some Staff members believe the Report does not require enough operating experience and does not require any safety research programs (e.g., LOFT) to be completed before more populous sites are approved.

e. Some staff members have difficulties with the requirements attached to transportation of spent fuel elements. This is believed to be a separate issue from the Report.

f. Mr. Price stated that there is some difficulty related to requiring lower radioactivity levels for routine releases. The AEC is now considering this issue. Mr. Price noted he is being prodded by the Commission to develop possible alternate solutions for consideration. He added it is difficult to reduce the release limits which are already considered by ICRP and MCRP to be extremely low levels. One possible solution is to require certain hardware systems which provide better means to keep the releases at the present low levels without actually reducing Part 20 limits. Mr. Price is not sure if the Commission is willing to accept this approach as a solution.

g. Mr. Price commented that he is not sure how to resolve the Committee's statements in the Report where it mentions that "more" must be done in specific areas. He is not sure the Staff can tell when the "more" has been accomplished. This is an "open-ended" statement. He added it would be difficult to tell the public when industry has done enough to meet the added requirements for the more densely populated sites.

Meeting with Commissioners

Attendance was limited to the Commissioners (Seaborg, Ramey, Johnson, Thompson, and Larson), Mr. Price and ACRS Members (R. F. Fraley and W. B. McCool attended as observers.) The following items were discussed:

Proposed ACRS Report on Location of Power Reactors at Sites of Population Density Greater Than Indian Point - Zion - The Commissioners indicated that they would prefer not to receive this report at this time. Individual Commissioners noted that most of the items mentioned are already under active consideration by the Commission and/or the Staff and suggested that several, particularly those relating to radwaste release levels and the increased level of safety required for "city" reactors, could create difficult public relations problems.

Individual ACRS Members suggested that the Report was intended to:

a. provide guidance as requested by the AEC, JCAE, and the nuclear industry with respect to the requirements for siting reactors in more populous areas;

b. indicate that the ACRS and the regulatory procedures are not inflexible with respect to siting;

c. indicate that "backfitting" will not be required at "good sites" of the devices developed for "poorer sites". This would encourage development of improved safety devices that is now inhibited by the industry's concern of "backfitting" of new features to plants at all sites.

It was suggested that Committee representatives work with the nuclear industry and the Staff in establishing an acceptable level of safety for those items mentioned in the draft report as requiring improvement.

The minutes of these meetings and the November 5, 1969 meeting of the Siting Evaluation Subcommittee show that the draft letter received less than an enthusiastic response. Comments were received that the timing of the letter was unfortunate, and that some elements of the letter would create more of a problem now than they would have a year ago. The situation regarding timing was not likely to improve in the near future, however. There were also comments that the provisions ofthe letter were too general, and that too little operating experience was required.

The Commissioners indicated strongly that they would prefer not to receive the letter, and inquired whether the ACRS could accomplish the desired purpose of the letter by other means. It was still the Committee's option to send the letter, if it so wished. However, as noted in the minutes of the 114th meeting, October 9-11, 1969, and the November 5, 1969 Siting Subcommittee meeting, the ACRS decided to accept the suggestion of the Commissioners to set up joint meetings of the Regulatory Staff and ACRS with representatives of industry.

A brief excerpt from the minutes of the November 5, 1969 Subcommittee meeting illustrates some of the considerations involved.

> Dr. Monson stated that, if the Committee wishes to develop criteria,there is a question as to how the criteria would be made public and used. Dr. Hanauer said that the Committee had undertaken to draft a letter to provide guidance. He stated that the Committee wished to inform the nuclear industry that considerably better designs and considerable operating experience will be required before worse sites are approved. Dr. Kaufman inquired as to whether the Committee had gone on record as being opposed to construction of present type reactors at sites worse than Indian Point and Zion. Dr. Hanauer replied that it had at a JCAE Hearing. The Committee said that none of the reactors presently proposed is suitable for urban siting. Dr. Hanauar indicated that industry representatives have stated that the Indian Point and Zion sites are the worst sites that are likely to be approved for some time. Dr. Bush commented that this was on the basis of the withdrawal of the Burlington Application.
>
> Mr. Mangelsdorf thought it might be useful to meet with industry representatives regarding the Committee's letter. If the letter is issued, it might be a better letter as a result. Also, industry could feel that it had a part in the writing of the letter. Mr. Mangelsdorf said that industry has stated that the AEC does not listen to its comments.
>
> Dr. Hanauer said that, from industry point of view, it would be desirable to have a document which sets forth what has to be done at worse sites. Dr. Bush thought that criteria might bring forth a negative response from industry. Dr. Hanauer stated that at least one manufacturer, GE has already taken the position that their present day reactors are what they developed for locating at sites involving higher population densities.

Dr. Monson indicated that he would feel uncomfortable if ACRS representatives achieved an understanding with industry regarding siting in more densely populated areas without having informed the public regarding this matter. Dr. Hanauer stated that it was his understanding that the Committee was to get industry input so that whatever is necessary can be published. He indicated, however, that if the Committee should develop a negative position it is not clear what should be done to publish the fact.

Dr. Monson said that industry cooperation was dependent upon the ACRS not requiring backfitting. Dr. Kaufman stated that he did not see how the Committee could guarantee this. Dr. Monson said that the Committee's draft letter stated that backfitting would not necessarily be required. Dr. Bush thought that, if the Committee cannot say this, it should not hold the meetings with industry representatives. Dr. Bush stated that history shows that the Regulatory Staff requires improvements in safety features to be added to all plants. The question was raised regarding how one can explain providing better protection for persons located near reactors in more densely populated areas. Dr. Hanauer indicated that one reason that can be given is because of evacuability. Dr. Monson thought that it is logical and appropriate to protect a large number of people better than a small number.

Dr. Hanauer suggested that industry might be asked:

(1) Do they really want to locate reactors closer to large metropolitan centers.

(2) What industry believes is required before reactors can be located at such sites.

(3) Are any of the items suggested in response to (2) not appropriate for reactors in less densely populated areas.

During the discussions with the Regulatory Staff at the November 5, 1969 Subcommittee meeting, it was decided that Mr. Price would issue letters of invitation to various representatives of the nuclear industry to present their views on a range of matters related to metropolitan siting (the details of the letter to be mutually agreeable to the ACRS and the Regulatory Staff). This was done, and on March 4, 1970, the Site Evaluation Subcommittee and the Regulatory Staff met with representatives of the Atomic Industrial Forum and 10 utilities in the first of several such meetings. In his letter of invitation Mr. Price said

> the Commission believes that before the siting of power reactors in closer proximity to metropolitan centers can be favorably considered, further advances are needed in reactor plant design and the capability of safety systems and engineered safety features, in adapting critical components and systems to accomodate reactor inspection and testability, and in the practical demonstration of dependability of performance of vital systems.

And a list of specific items for discussion was included.

At the March 4 meeting, Dr. Monson of the ACRS said that he would like to secure the opinions of the industry representatives present, relative to what R & D and improvements they believe could be made in order to use worse sites. He stated that the ACRS was not interested in promoting nuclear power, but was interested in assuring adequate safety for whatever sites were used. He indicated that the Committee was under the impression that new techniques could be proposed, but may not be because of fear that the Regulatory Staff and the ACRS could require that any new improvements be incorporated for reactors, no matter what site was involved. ACRS member Monson said that, judging on a technical basis, developments for reactors to be located at worse sites need not necessarily be required for reactors at better sites. He thought, however, that a public policy question arises relative to whether it is advisable to provide a greater level of safety at one site than at another solely because of population density differences. An associated question was whether improvements developed for more populated sites would then be required for remote sites.

Mr. Brush of the Nuclear Reactor Safety Committee of the AIF presented a statement in which he said that there are at least two large utilities, one on the East coast and one on the West coast, where there is a pressing need to use worse sites than Indian Point - Zion. He stated that the use of metropolitan sites would alleviate the problem of obtaining right of ways for overhead transmission lines, might mitigate problems with environmentalists arising from industrial plants in non-urban areas, and would provide additional taxes in metropolitan areas where such money is badly needed. Brush said that a record of trouble-free operation over the short term would have little or no relevancy to reactor safety, per se. Brush went on to say that, although his group would readily agree that a number of outstanding issues should be resolved, there was little evidence at hand to suggest that urban siting must await the results of further research and development. Rather, he said that what they believe is needed is more comprehensive review and analysis of the data already on hand, and assessment and application of the data to revising and updating the outmoded assumptions used in such criteria as TID-14844, and a meaningful interpretation of how the AEC will permit the data to be applied to licensing problems. When Mr. Case of the Regulatory Staff asked Mr. Brush what he considered to be an acceptable level of safety, Brush replied that many in a nuclear industry believe that present plants have an acceptable risk for urban siting.

Mr. Wascher, another AIF spokesman, was quoted as stating that he felt that the industry should not find that there are new research and development requirements, after the appropriate operating experience is secured. He believed that the ACRS and the Regulatory Staff should determine what research and development is needed and undertake to obtain the necessary data.

Mr. Cahill of Consolidated Edison stated that there was a precedent for additional safety features for reactors in more densely populated areas. He pointed out that Consolidated Edison reactor sites, which involved higher population densities than others, had required the use of additional safeguards. Mr. Cahill said that he thought that plants could be adequately designed for a site like Ravenswood. He stated that industry did not design plants in terms of probability, and that the reliability of equipment was not known. Mr. Cahill said that industry could design for accidents which were considered credible, and that it was ultimately the responsibility of the Regulatory groups to make a judgment as to what is credible.

When members of the ACRS and the Regulatory Staff pointed out that there was a value of operating experience that could be obtained from the unfavorable aspects, the lessons that could be learned, and the mistakes that would be rectified in the next reactors, the general response from the industry representatives was that operation of large power reactors during the next two years would not be meaningful witn regard to safety evaluation and design.

In summary, two utilities expressed a need to use worse sites than Indian Point - Zion, Consolidated Edison and Southern California Edison. Consolidated Edison wanted to use sites having population densities equivalent to or not far different from Ravenswood, while Southern California Edison desired to use sites like Bolsa Island or Burlington.

The positions of the various industry representatives present varied widely regarding the protective features which would be necessary at various types of sites. Some believed it would be difficult not to make the same safety provisions for reactors located in the country as were provided at urban locations. Mr. Case concluded that the industry representatives seemed to be asking what was needed in order to locate reactors in more densely populated areas.

In the next months similar meetings were held with each of the light water reactor vendors, Babcock and Wilcox, Combustion Engineering, General Electric and Westinghouse. Only Mr. J. M. West of Combustion Engineering made specific recommendations for engineered safety improvements that he thought would be appropriate in connection with metropolitan siting. Some excerpts from the statement by West at the meeting of May 5, 1970 follow:

- A fracture of the reactor pressure vessel or any other part of the primary system must be accommodated.

- An additional zero leakage barrier must be imposed to prevent the escape of fission products from the containment building. This leakage barrier must be effective over long periods of time without operator attention and without power.

- Containment building leakage must be continuously monitored.

- Discharges of radioactive materials to the atmosphere or to waterways must be further reduced.

- A large heat sink must be provided such that residual and decay heat can be accommodated for many hours without a source of electrical power and without any operator action.

- The plant must be designed against any credible natural forces such as earthquakes and hurricanes, and against external missiles such as aircraft.

None of the safety criteria which I visualize for a metropolitan nuclear power plant appear to be impossible to satisfy. A fracture of the reactor pressure vessel could be accommodated by having a double pressure vessel or by having a restraining envelope outside the reactor pressure vessel. This would cost several million dollars. We have a conceptual design of a Passive Reactor Containment System (and associated auxiliaries) which would go a long way toward eliminating concern about leakage of radioactivity to the environs following an accident. This would also cost several million dollars. Even the "China Syndrome" could probably be accommodated at a finite cost. Radioactive discharges during normal operation could be reduced by a large factor at additional cost.

No problem has occurred to me which could not be handled, if it must be, in order to locate a nuclear plant in a populated area. The cost of safety features to handle these problems would be high. Utilities could not afford the cost of these features added to the already high cost disadvantage of transmission lines from remote sites. However, we must remember that several tens of millions of dollars can be spent, if necessary, on safety items in order to qualify a 1000 Mwe nuclear plant for a highly populated zone. There is at least a fair chance that safety requirements can be met at a cost which is tolerable. I recommend that we attempt to identify what will be required, develop these designs to the point where cost estimates can be made, and assess whether metropolitan siting appears to be economically and technologically feasible. If so, a strong Government industry program should be initiated.

To make any substantial progress in resolving the dilemma which now exists, we must make a clear distinction between the safety requirements for nuclear plants in metropolitan areas versus the requirements for rural locations. These requirements cannot be the same. I do not believe for one minute that after a reasonable amount of satisfactory operating experience is obtained on the present generation of light water reactors, an identical reactor will be licensed for a site in the heart of New York City or Los Angeles. If that were to occur I would feel that the safety requirements for non-populated sites had been too stringent.

Starting with satisfactory operating experience from essentially identical reactors, what additional requirements are likely to be imposed for metropolitan siting? In my opinion the answer does not lie in further ratcheting toward the items listed under Item D of this agenda (increased attention to quality assurance, further protection against common failure modes, increased margins, etc.). Instead I believe there should be a fresh look at the whole safety matter with an objective of defining what is desired and then attempting to meet the criteria for metropolitan nuclear plants by radical changes in design if necessary. Let's face it--we arrived where we are today on safety requirements for non-metropolitan plants by a torturous path which in retrospect does not appear too logical. Rather than add safety pins to the belt and suspenders already required to support the trousers, perhaps we should change the basic garment to something which is inherently more suitable--say coveralls.

One utility company has recently estimated that just in the past five years the cost of an 800 MWe nuclear power plant has increased by 20 to 30 million dollars solely due to additional licensing requirements. Even so, nobody would conclude that the plants licensed five years ago are unsafe--nor conversely that the present plants are necessarily overdesigned. We still assume, as we did then, that the fuel melts, major fractions of the fission products mix with air and steam in the containment building, that some of these fission products leak out under adverse meteorological conditions, etc.

What is a metropolitan nuclear plant likely to need that a rural plant does not have? To catalyze discussion I will mention the following things which have occurred to me:

- Engineered safeguards must be much less dependent upon correct sequential operation of "active" devices such as detectors, switches, valves and motors. Our safeguards systems should be passive to the maximum practical extent.

Mr. Storrs of Combustion Engineering discussed a core-catcher, but Mr. West stated that the core meltthrough accident was not on his list of items requiring resolution for metropolitan siting, and stated that he hoped it was not on the ACRS list. This was almost the only mention of the "China Syndrome" problem throughout the meetings with the representatives of industry.

Two conclusions drawn by the ACRS and Regulatory Staff from the series of meetings were that, except for 2 or 3 utilities, there was generally no urgent need for sites in densely populated areas and that, with the exception of Combustion Engineering, the industry had contributed no suggestions for specific design criteria for poorer sites.

In a subsequent series of Subcommittee meetings and full Committee discussions, the ACRS decided it would prepare a written report to the Atomic Energy Commission concerning the outcome of the meetings which had been held with members of industry that would review the opinions of the ACRS concerning the possible use of sites worse than Indian Point and Zion. A tentative final letter was adopted by the Committee at its 127th meeting, November 12-14, 1970 with the provision that the Comittee talk to the Commission at the December, 1970 ACRS meeting prior to transmission of the letter. There was the understanding in the Committee that the letter could be revised after the discussion, if the Committee so chose. The subject was discussed between the ACRS and the Commissioners at the 128th meeting, December 10-12, 1970. The ACRS approved the draft report after this discussion, but then, at the request of the Commission, the Committee agreed to defer dispatch of the letter pending further discussion with the Commissioners at the 129th meeting, January 7-9, 1971. After this second discussion, the Committee concluded it would not send the letter which it had adopted at the 128th meeting. A copy of the letter, which was never formally sent, is reproduced on the following page, together with the minutes of discussion on the subject at the 128th and 129th meetings.

Final
HCM/ka
11/13/70

Dr. Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON USE OF WATER-COOLED POWER REACTORS AT SITES MORE DENSELY POPULATED THAN THOSE EMPLOYED TO DATE

Dear Dr. Seaborg:

A series of meetings has been held to consider the general subject of location of water-cooled power reactors at sites more densely populated than those employed to date. The meetings were conducted jointly by representatives of the Advisory Committee on Reactor Safeguards and the AEC Regulatory Staff. Six meetings were held: one with representatives of the Atomic Industrial Forum, ten utilities, and eight architect-engineers; one each with Babcock and Wilcox, Combustion Engineering, General Electric Company, and Westinghouse Electric Company; and, one with the AEC Division of Reactor Development and Technology.

The principal items of discussion were the same for all meetings. These included: the extent of the need for use of sites of higher population density; the potential significance of operating experience to be acquired before locating power reactors at such sites; and, the areas of improvements, principally new or improved reactor plant design features, appropriate to the use of such sites.

The meetings have been concluded, and the Committee wishes to make the following comments:

1. Based on discussion at the meetings, it appears that, except in the case of a small number of utilities, a critical need for the use of sites having a significantly greater population density than those approved to date is not expected to develop before the late 1970s or early 1980s. A few utilities indicated critical need for earlier use of such sites.

2. The participants expressed a wide range of views as to the necessity for obtaining extensive operating experience with a given type of plant before locating such a plant at a more densely populated site. Some felt such experience to be of major benefit. Others questioned the relevance of operating experience to safety. It remains the Committee's opinion that valuable experience is being accumulated with operating plants, and that this experience is leading to improved reliability and greater safety. The Committee continues to believe that reactor power plants as currently designed should not be employed at sites more densely populated than those approved to date until substantial further evidence of satisfactory operation has been obtained.

3. There were noted and discussed with the participants a number of general areas in which safety provisions might potentially be improved. These are listed below.

a. More thorough implementation of quality assurance in design, construction, test, operation, and maintenance.

b. Additional improvement in containment design to effect further reduction in potential off-site doses in the unlikely event of a major reactor accident.

c. Further reduction in the already low probability of occurrence and the consequences of smaller accidental releases.

d. Additional protection against damage by missiles generated within the plant.

e. Additional protection against common failure modes in safety systems.

f. Increase in design margins of engineered safety features.

g. Minimization of the number of potentially significant safety issues left for resolution between start of construction and start of power operation.

h. Reduction in probability of occurrence and consequences of loss of integrity of the primary coolant system outside the containment.

i. Additional protection in areas related to possible degradation of reactor vessel integrity.

The Committee believes that these are the principal general areas in which improvements are required for use of sites having population densities between those of sites for which construction permits have been issued and that of the Burlington site.

4. More specific requirements related to the improvements needed in the general areas listed have not been identified, and little new information was brought to light during the subject meetings which would aid appreciably in the establishment of such specific requirements.

5. The Committee believes that the AEC should continue to pursue the delineation of specific requirements for improvements in water-cooled power reactors appropriate to the use of more densely populated sites on a time scale commensurate with the national needs for power. The Committee also believes that applications for construction permits for more densely populated sites should continue to be considered on a case-by-case basis, and anticipates that the information developed in this process will be of considerable importance in the eventual establishment of specific requirements.

Sincerely yours,

Joseph M. Hendrie
Chairman

128th ACRS Meeting Summary - Dec. 10-12, 1970
Meeting with the Commissioners
Commissioners Seaborg, Johnson, and Larson participated.

1. Use of LWRs at Sites with More Dense Population - The Commissioners were provided a draft report by the Committee on Use of Water Cooled Power Reactors at Sites More Densely Populated Than Those Employed to Date. Dr. Monson reviewed the results of meetings held between the ACRS/Regulatory Staff and representatives of the nuclear industry. (The meetings were held as a result of a suggestion by the Commission.) He noted that the ACRS has undertaken the use of a population index for site evaluation. This index is to be made available to the Regulatory Staff and the nuclear Industry. Dr. Monson added that industry was concerned that two design standards would be developed - one for presently acceptable sites and one for worse sites.

   Commissioner Johnson agreed with this industry concern. He was interested as to how the Committee arrived at practical design standards for highly populated areas, e.g., what should the requirement be for prevention of pressure vessel failure.

   Dr. Hendrie pointed out that the Committee had asked for improvement of means to mitigate the effects of reactor vessel failure at Indian Point 3, e.g., improved vessel cavity design to withstand certain failures. He expressed his personal belief that for presently acceptable sites, in which evacuation of the populace would have a reasonable possibility of success in an emergency combined with the low probability of vessel rupture, the requirements are acceptable. As more densely populated sites are selected, more consideration should be given to pressure vessel failure and its effects.

   Dr. Okrent added that he is on record as recommending that careful consideration be given to make provision for coping with a leak or split in the pressure vessel wall of the Zion units. He noted that a probability of vessel failure has been identified by the British as being in the 10^{-4} to 10^{-5} range.

   Commissioner Larsen was informed that Con Ed has stated that they are desirous of locating nuclear facilities in very highly populated areas in the relatively near future, e.g., David's Island which has eight times the Indian Point population at close-in-to-site locations.

   Dr. Beck indicated that he saw no reason why the Committee report should not be received by the AEC. He noted that parts of the last paragraph were unclear regarding review on a case basis. Dr. Hendrie commented that the intent was to have the Regulatory body consider the cases. (This section was reworded in Executive Session.) Commissioner

Johnson suggested that review of a specific case might be encouraged to provide explicit direction to the industry. He also questioned the need for including the list of areas which need improvement without a more specific indication of what is desired.

The Committee approved the draft report following the discussion with the Commissioners.

(Note: Following this meeting the Commission requested that the ACRS defer this report until it could be discussed further. The Committee agreed to this request. Discussion of this item during the January ACRS Meeting has been tentatively scheduled.)

Summary, 129th ACRS Meeting, Wash. D.C. January 7-9, 1971
Meeting with the Commissioners
Commissioners Ramey, Larson, and Johnson participated.

Siting Letter - Commissioner Ramey stated that there was some doubt in the AEC's mind if the Committee letter on the use of reactors at more densely populated sites than employed to date would serve any useful purpose at this time, e.g., the same matters are covered in the Newbold Island letter.

Commissioner Johnson added that the letter would hurt in that it would provide information to strengthen intervention regarding nuclear plants at currently acceptable sites. He saw no difficulty imposing specific requirements mentioned in the siting letter for a specific case.

Commissioner Ramey noted that it would probably be possible to prepare a transmittal letter which tells the public that the AEC/ACRS have had discussions with industry about siting of plants near dense populations.

Mr. Mangelsdorf suggested that, even though members of the ACRS have met with industry, the impact of ideas expressed by Committee members does not have the same weight as would be the case if the Committee wrote the AEC a letter. (In Executive Session the Committee decided to "file" the siting report it had drafted.)

Executive Session

Siting Report - The Committee decided to "file" the report on "Use of Water-Cooled Power Reactors at Sites More Densely Populated Than Those Employed to Date." This was based primarily on the desire of the Commission not to receive such a report.

This was the last effort by the ACRS to write a "general" letter on safety requirements for the use of sites more populated than Indian Point-Zion. The upcoming Newbold Island construction permit review first opened the door toward the use of a somewhat more populated site, and then, with the Regulatory Staff decision that, under NEPA considerations, a less populated site should be used by the applicant, a rather firm line was drawn, at least for the interim, at Indian Point-Zion.

2.13 THE CHINA SYNDROME - PART 2

As discussed in the section "China Syndrome - Part 1", at its 77th meeting, September 8-10, 1966, the ACRS met with the AEC Commissioners concerning the proposed ACRS report on "problems Arising from Primary System Rupture." The Commission proposed delay of any report; they proposed rather that the Commission would establish a task force to study and report in a few months on questions arising from the "China Syndrome" matter. The ACRS reviewed the AEC recommendation during a very difficult executive session which left the Committee very much divided. The position taken in August had been that a general letter would be written, and this position was part of the Committee treaty by which the letters on Indian Poiint 2 and Dresden 3 were agreed to. This position changed in September when it was decided to accept the AEC suggestion of a task force.

Excerpt from Summary Letter September 14, 1966

The Committee met with the Commission and discussed several items related to ACRS activities. The Committee devoted considerable attention during this meeting to problems associated with the low probability accidents involving primary system rupture followed by functional failure of the emergency core cooling system. During the discussion between the Committee and the Commission and members of the Commission staff on this subject, the Commission suggested that a technically competent task force, including personnel from the AEC staff, Commission laboratories, industry, universities, etc., be formed to gather pertinent information.

The Committee endorsed this suggestion, urged rapid convening of such a task force, and recommended that the topics to be assigned to the task force include the following:

> The degree to which core cooling systems could be augmented for additional assurance that substantial meltdown does not take place; the potential history of large molten masses of fuel following a hypothetical accident; the engineering problems associated with possible "core catcher" systems; and the build-up of excessive pressure or an explosive atmosphere in the containment.

At the meeting, the Commission suggested to the Committee that a review of existing information bearing on these problems might be available within approximately two months. The Committee has stated that early completion of such a review would be of considerable use in connection with current license applications. The Committee also suggested that the task force propose a course of action to assure development of any additional information needed. The Committee has expressed its willingness to cooperate with this task force.

Failure of the Committee to maintain the agreement taken in August remained a sore point within the ACRS for many, many months. Several members felt let down because they had agreed to the position taken in August strictly as a package. More importantly, it probably delayed development of useful and even desirable safety improvements.

Chronologically, the next development related to the "China Syndrome" came with preparation of a report on Reactor Safety Research by the ACRS at the 78th meeting, October 6-8, 1966. In its report of October 12, 1966 the ACRS attached special imprtance to several safety research areas, and identified others as also having significance. First on the Committee's list was the following:

> 1) A vigorous research program should be initiated promptly on the potential modes of interaction between sizeable masses of molten mixtures of fuel, clad and other materials with water and steam, particularly with respect to steam explosions, hydrogen generation and possible explosive atmospheres. Work should be directed to understanding the mechanisms of heat transfer connected with such molten masses of material, the kinds of layers formed at cooled surfaces, the nature and consequences of any boiling of the fuel, and the manner and forms in which fission products escape from molten fuel mixtures. Further studies should be initiated by industry to develop nuclear reactor design concepts with additional inherent safety features of new safeguards to deal with low probability accidents involving primary system rupture followed by a functional failure of the emergency core cooling system.

Second on the ACRS list of emphasis was increased work on emergency core cooling systems. Third was the development of practical methods for extensive periodic inspection of pressure vessels. And fourth was a strong program on the behavior of thick walled pressure vessels, including a thorough study of potential failure modes under pneumatic loading and the significance of neutron irradiation on pressure vessel integrity.

The Atomic Energy Commission announced establishment of a Task Force on October 27, 1966, per the press release on the following page.

Of the task force members, six came directly from the nuclear industry while five worked at national laboratories or non-profit research institutions under contract to the AEC.

Actually, the Task Force had its first meeting October 14, 1966. Member Louis Baker wrote a memorandum on this meeting dated October 17, 1966 to R. C. Vogel, his Division Director at Argonne. The letter from the AEC confirming Baker's willingness to serve on the task force first notes that the ACRS reports on Dresden 3 and Indian Point 2 contain the suggestion that the Staff and the ACRS should review the design, fabri-

AEC

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON D.C. 20545

No. IN-727
Tel. 973-3335 or
973-3446

FOR IMMEDIATE RELEASE
(Thursday, October 27, 1966)

AEC ESTABLISHES TASK FORCE TO STUDY
FUEL COOLING SYSTEMS OF NUCLEAR POWER PLANTS

The Atomic Energy Commission has established a task force of persons from the nuclear industry and the AEC's national laboratories to review power reactor emergency core cooling systems and the phenomena associated with core meltdown. Representatives of the Commission's Regulatory Staff and the AEC's Division of Reactor Development and Technology will participate in task force meetings. The Advisory Committee on Reactor Safeguards has participated in discussions with the Commission leading to the establishment of the task force.

The task force will consider the following topics:

1. The degree to which core cooling systems could be augmented, by way of design modifications and/or new design concepts, for additional assurance that a substantial meltdown is prevented.

2. The potential history of large molten masses of fuel following a hypothetical accident.

3. The possible interactions of molten fuel with materials or atmospheres in containments, and phenomena associated with such interactions.

4. The design and development problems associated with systems whose objective is to cope with large molten masses of fuel.

Members of the task force are:

Dr. Louis Baker, Jr., Argonne National Laboratory, Argonne, Illinois;

Mr. Eric Stephen Beckjord, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania;

(more)

Mr. A. Philip Bray, General Electric Company, San Jose, California;

Dr. William Krasny Ergen, Oak Ridge National Laboratory, Oak Ridge, Tennessee;

Dr. Salomon Levy, General Electric Company, San Jose, California;

Mr. I. Harry Mandil, MPR Associates, Washington, D. C.;

Mr. David L. Morrison, Battelle Memorial Institute, Columbus, Ohio;

Mr. Warren E. Nyer, Phillips Petroleum Company, Idaho Falls, Idaho;

Mr. Michael F. Valerino, Combustion Engineering, Inc., Windsor, Connecticut;

Mr. Robert E. Wascher, Babcock and Wilcox, Lynchburg, Virginia, and

Mr. T. R. Wilson, Phillips Petroleum Company, Idaho Falls, Idaho.

Dr. Ergen will serve as chairman of the task force.

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10/27/66

cation procedures, plans for in-service inspection and analyses pertaining to the ECCS "prior to irrevocable construction commitments pertaining thereto." The letter then gives the task force charter. Baker notes that, at the first meeting, work on ECCS was identified as the major subject for consideration, rather than possible means to cope with core meltdown. The task force members set up problem areas and appointed leaders, with the industry members working on LOCA prevention or the LOCA-ECCS area, not core melt.

Baker also notes that the Task Force report was to be an appraisal of the situation and a recommended course of action. The document would not state "how to design" or "how to license."

Another document of some interest is a memorandum by W. K. Ergen to the ACRS office (received September 16, 1966). Ergen, a former ACRS member, had been serving as an ACRS consultant during the Dresden 3 and Indian Point 2 reviews. The memorandum is duplicated below (the last page appears to be missing).

> Enclosed is a copy of the notes I wrote at the recent ACRS meeting representing what I believe to be the ACRS present position on primary-system rupture. No attempt has been made in the notes to edit them for general publication, and for this reason they may be more concise and easier to understand than the draft letter. Of course we can't be sure that they represent the Committee correctly unless the Committee had a chance to review them. Use them as you see fit.
>
> 1) With the increase in total power, power density and fuel exposure, the afterheat has become so large that it will violate the containment in the event that normal and emergency cooling be lost.
>
> 2) Such violation may occur
>   a) by overheating, cladding or other metal to the point of the containment by hydrogen generated and heating of the gases inside the containment.
>   b) short over-pressurization by steam or hydrogen explosions.
>   c) melting of the afterheat-generating remains of the core through the bottom of the containment. This violation will occur with certainty in the above postulated event in presently designed reactors because the afterheat is not removed sufficiently prior to such a meltthrough.

3) There are three lines of defense against this violation of the containment, which are listed below with their inherent difficulties:

a) Extreme care in design, fabrication, inspection and operation of the primary system and all components the failure of which might lead to a loss of the normal cooling.

   Difficulty: There are limits on how far one can go in this direction in a price-competitive industry. Also, the possibilities of inspection are limited and the effect of the radiation is unknown.

b) The same care with respect to the emergency cooling system. The same difficulties apply here. In addition, the presently designed emergency cooling systems have the following problems:

   (1) There is only a short time interval between the moment in which emergency coolant can be injected against the pressure in the core, and the moment when the emergency cooling has to operate.

   (2) Essential parts of the emergency cooling system have to be close to the core, which is postulated to have undergone the violent accident of loss of normal coolant. It is difficult to insure that the same accident won't impair the emergency cooling system, even if this system is redundant.

   (3) The emergency cooling system is complex, as it has to handle high-pressure, low-flow situations as well as low-pressure, high flow situations; and as the heat is removed from the core to the inside of the containment and from there, in a second step, to the containment outside.

   (4) The condition or even location of the core-remains after the loss of normal cooling are somewhat uncertain.

   (5) The test of emergency cooling systems under operating conditions and inspection of such systems in operating power reactors have not been reassuring.

c) Attempt to cope with the consequences of the event postulated above. The long and short time over-pressurization might be handled by strengthening the containment or increase in its volume or possibly filtered release to the outside. Whether the afterheat generating remains of the core could be caught short of melting through the containment is not known.

4) In the face of the above facts, the Committee has a short-term policy, to which it intends to adhere during a transition period, and a long term policy to which it intends to adhere after this period. The length of the transition period is estimated at two years, but it will be adjusted to fit developments. The short-term policy is less stringent than the long-term policy. This is admittedly inconsistent. However, it is preferable to either of the two other alternatives. One of these alternatives is to impose the conditions of the long-term policy right away. The exact details of the measures required to meet these conditions are not known at present and hence this alternative would be, in effect, a moratorium on reactors. The other alternative would be to disregard for all future time the risk inherent in the event postulated above, in spite of the fact that means to cope with the event could be developed.

   The number of reactorsto be brought before the Committee in the transition period is low compared to the number to appear later, and the Committee believes that the short term policy offers adequate protection for this relatively low number of reactors.

5) The short-term policy consists in

   a) approving reactors of the "present boiling water design" only for rural, or more remote, sites. A reactor of the "present boiling-water design" is typified by Dresden 3, and a "rural site" is typified by the Dresden site.

At the special ACRS meeting, December 2-3, 1966, Ergen presented a brief review of the status of the effort of the effort of the Task Force. Some excerpts from the minutes of this meeting are included for their insight into the way matters were developing.

One of Task Force Members, Mr. Mandil, has been urging a particular approach which has some merit, although it is not clear that it falls within the original objectives of the group. Mr. Mandil feels that, if the quality assurance requirements imposed on the pressure vessel are met by the other primary system components, their failure probabilities should be assumed to be equally low. Mr. Mandil has undertaken to draft primary system requirements which, if met, would allow this approach. He has opposed the use of high-pressure accumulators as passive emergency core cooling systems on the grounds that they are designed only to protect the core in the event of an extremely unlikely accident and themselves represent a hazard. Mr. Mandil is of the opinion that the required assurance can be achieved by upgrading the primary system.

Dr. Ergen, on the other hand, along with most Task Force members, feels that piping failures can not be completely ruled out and that the accumulators do not represent a significant hazard. It is not clear what the outcome will be as far as the Task Force is concerned.

The Task Force has reached the following tentative conclusions:

During blowdown, the core heats up rapidly to about 2200°F at which point metal-water reaction rates become rapid and energy is being released at a rate which makes the successful injection of sufficient water to recover the core doubtful.

Approximately one minute is available until this point is reached. Accumulators and other high-speed systems operate in about 20 secs., providing a safety factor of three. If water can not be injected before rapid and extensive metal-water reactions occur, it is uncertain if core cooling will be effective.

It is fairly clear that containment integrity can be assured up to the point at which the molten mass of fuel and structural material melts through the pressure vessel. After this, however, the uncertainties are so significant that the containment can not be considered a barrier. The possibility exists of violent steam explosions resulting from the blowdown. Water trapped beneath the molten material would produce a relatively minor steam explosion resulting in the dispersal of the mass throughout the water and a subsequent rapid heat input into the water, causing a second, violent steam explosion.

The Task Force will not provide answers to all of the problems. Some potential solutions have suggested themselves, however, and seem to be worth investigating. For example, venting the containment through filters or after spray-cleaning the air, could result in a much reduced accident when the containment finally is breached.

Based on the Task Force meetings so far, there is no agreement concerning the containment response or its requirement. It is not at all clear that anything will be said about containment in the final report of the Task Force. Most members are agreed that containment is required in the event of lesser accidents. However, there is a body of opinion within the Task Force which says that the containment design pressure should be reduced to reflect the reduction in probability of major accidents due to recent improvements in cooling system design. If a major accident should occur, the containment will not offer protection anyway.

A second group within the Task Force is of the opinion that, since containment can not be considered sure protection against very large accidents, one need not be concerned with assuring the very low leak rates required following such accidents.

Dr. Ergen remarked that he had been concerned that the conclusions of the Task Force might be overly influenced by its more optimistic members. He has urged that the Task Force not necessarily reach agreement in every area. If differences of opinion or major uncertainties exist, these should be made explicit in the final report.

Dr. Okrent commented that , if the Task Force were to make judgements concerning the suitability of present design approaches, the effect would be that of the industry judging itself.

Dr. Ergen replied that at the last Task Force meeting, there were some signs that such conclusions might be attempted in the final report. He felt that some effort would be required to keep such conclusions from appearing under the aegis of the Task Force.

On February 8, 1967 there was a meeting of the Engineered Safeguards/ Primary System Subcommittee of the ACRS with the AEC Task Force on Emergency Core Cooling Systems. In the executive session, one of the ACRS staff engineers noted that the task force was industry oriented and had taken the approach that reliance can be placed on a good emergency core cooling system. Several excerpts from the minutes of this meeting follow to give an idea of the points ofview that were developing among task force members and some of the technical opinions that they were adopting.

Meeting with Task Force

Dr. Ergen stated that large reactors are capable of generating afterheat such that, if normal and emergency core cooling systems are both violated, it is almost certain that containment will be violated, most likely by melting through the containment at the bottom. He indicated that there is not much hope of guaranteeing the integrity of containment, should meltdown occur.

In addition to behavior of the large molten fuel mass, Dr. Ergen noted that two additional problems were not well understood:

1. A steam explosion from the molten mass plus water could break the containment.

2. If the UO_2 heats to the melting point, a substantial fraction of the core might redistribute, vaporize, and lead to an indeterminate situation.

The Task Force therefore gave up guaranteeing that the containment will not be violated. It looked instead at two barriers: primary system integrity and the emergency core cooling system.

The Task Force confined its studies to reactors of present designs and locations. A large failure of the pressure vessel was deemed to be sufficiently unlikely that it represented an acceptable risk.

The Task Force did not propose giving up considering the consequences of a meltdown. It believes that it is probable that containment breakage can be delayed for about one hour or maybe longer. If cleanup systems can be devised to take advantage of this hour, this could be a deserving effort. Although containment integrity beyond one hour cannot be guaranteed, the Task Force does recommend a small and deliberate effort on seeing what happens in the event of a meltdown. This is not to be a "crash-like" effort, however.

Mr. Mandil stated his belief that the first line of defense should be the integrity of the primary system. He indicated that much has been done already, but the large number of plants to be built makes it prudent that even greater assurance can be provided for here on.

T. R. Wilson discussed the ability to predict the loss of coolant accident and stated that current technology was sufficient to enable prediction with reasonable assurance, although further experiments and analyses were required.

A. Philip Bray indicatged that the current interpretation of "no clad melt" in the core during a LOCA was to maintain a definable geometry and to maintain flow paths. All melting was not to be precluded. Bray believed that emergency core cooling systems could be practical. The technology was known or could be bounded.

In a Subcommittee caucus, ACRS member Harry Monson read the charter of the Task Force, then observed that the ACRS had presented the problem but the Task Force had restated it. There was discussion about the need for a harder look at those topics the Task Force was supposed to examine.

When the Subcommittee reconvened with the Task Force, there was discussion on the degree of augmenting the ECCS by new design concepts. Aside from accumulators (there was disagreement about the timing and place of their origin). Dr. Ergen said the Task Force had come up with no new ideas on this and hadn't given it much attention. He indicated that, by choice of a majority of the Task Force and "their bosses", the Task Force assignment was escalated to considerations of emergency core cooling systems, the primary system and judgement. Mr. Beckjord commented that, after a meltdown accident, the Task Force had looked and found it couldn't be sure of maintaining containment integrity. Instead of charging in many directions, it elected to try to do something about primary system integrity and ECCS.

Member Hanauer observed that people had thought they understood things like reactivity excursions before and there were some surprises. He wondered where the acknowledgement was that there may be some things we haven't thought about for real loss of coolant accidents in real, large power reactors. Mr. Mandil stated that you can always postulate things from "blue heaven" and attach a lot of safety features which are not really helpful. Dr. Hanauer expressed his personal dissent that all the problems have been identified.

The work of the Task Force took considerably longer than the two months originally proposed by the Atomic Energy Commission. In late 1977 an un-numbered report became available under the auspices of the U. S. Atomic Energy Commission entitled "Report of the Advisory Task Force on Power Reactor Emergency Cooling." The Task Force report included 12 principal conclusions and began with a summary of the conclusions which is duplicated below, together with the conclusions themselves.

Conclusions

Within the context of the above Introduction and Scope, the results of the Task Force deliberations are presented as a set of conclusions followed by supporting discussions. Additional information, where appropriate, is given in appendices.

These conclusions are quite specific in some instances and necessarily broad in others; application in all cases requires judgment.

The principal findings can be summarized as follows:

a. Core Cooling. The Task Force has concluded that within the framework of existing types of systems, sufficient reliance can be placed on emergency core cooling following the loss-of-coolant and additional steps can be taken to provide "additional assurance that substantial meltdown is prevented."

The Task Force reached the above conclusion on the basis of the findings in the report that the events associated with blowdown and core heatup are definable within existing technology (Conclusion 1); that core structural response can be evaluated within conservative bounds (Conclusion 2); that appropriate requirements can be placed on core-cooling system design (Conclusion 3); and that the phenomena associated with the currently incorporated concepts represent satisfactory approaches to emergency core cooling (Conclusion 4).

The effectiveness and adequacy of cooling systems requires that the core be maintained in place and essentially intact. Failure to preserve heat transfer area and coolant flow geometry, results in a major increase in the uncertainty of prediction of core behavior. Recommendations are made with respect to general design conditions that must be met to maintain this geometry. Additional assurance as to the adequacy of the techniques employed can be obtained by carrying out certain recommended tests.

A systematized approach to emergency core-cooling system design and evaluation (Conclusion 5) would provide further assurance that these systems will be capable of performing their intended function. The Task Force offers some suggestions for implementing such an approach. The use of reliability techniques (Conclusion 6) could well be a part of this effort.

Any assessment of emergency core cooling and core meltdown problems requires consideration of primary system integrity. The reliability of the system is a key point in our judgment of the overall reactor safety against loss-of-coolant accidents. In line with the framework of safety design and evaluation, referred to above, we consider that further improvements in primary system integrity can be achieved by upgrading components of the primary system to the same level as that of the reactor vessel (Conclusion 7). Improvements are recommended which serve to decrease the likelihood of failure in the primary systems as well as in emergency cooling systems and which provide additional assurance that these systems will function properly (Conclusions 7 and 8).

The above conclusions and recommendations relative to emergency core-cooling systems follow the practice of assuming the initiating event in the loss-of-coolant accident to be a rupture in the primary system.

b. Core Meltdown. The Task Force reviewed the potential history of large molten masses of fuel following a hypothetical accident in which all emergency cooling was assumed to have failed. The role of containment in such an accident was also considered.

It was concluded that the description of the events that could take place subsequent to a postulated meltdown of large portions of a core is at present indeterminate and quite speculative (Conclusion 10). Reliable and practical methods of containing the large molten masses of fuel that would probably result from such a meltdown do not exist today (Conclusion 12). Accordingly, it is not considered possible to assure the integrity of the containment if meltdown of large portions of the core were to occur. Thus, the containment with respect to its objective and relation to other safeguards should not be viewed as an independent barrier, but it still represents a substantial safeguard system (Conclusions 9 and 10).

While with present technology the integrity of the containment cannot be assured in the event of a postulated core meltdown, there is likelihood that a length of time will elapse before breachment of the containment might occur. It may be possible to develop preventive measures which are effective during this period and which could significantly reduce the hazards resulting from subsequent failure of the containment (Conclusion 11).

The desirability of utilizing such systems and the merits of requiring containments to be designed to assure such time availability should be evaluated after the effectiveness of these systems has been established through necessary development work. The use of such safeguards will depend on weighing their merits with those of other safety features to obtain the desired objectives in overall reactor safety.

The Task Force considered also the design and development problems associated with systems whose objective is to cope with the consequences of core meltdown, such as large molten masses of fuel, and releases of energy and fission products.

We recommend for the near future a small-scale, tempered effort on these problems. The reason for this are as follows:

(a) if such systems could be developed and their reliability established, they would have certain advantages. They would probably not have to be connected to the primary system. In that case, the likelihood that they would be incapacitated coincident with the primary-system break would be still smaller. Any increase in confidence obtained from these systems could be used to reduce emphasis on other safety related features.

(b) to produce effective designs, if indeed feasible, might require both considerable fundamental research and practical engineering application. Both laboratory investigations and large-scale meltdown tests might be required, as scale-model tests or single-fuel-assembly tests might not be adequate. Core meltdown tests would require a large expenditure of funds, manpower, and an extended schedule to complete the design, fabrication, testing and evaluation. Important aspects which could be included in the scope of such a basic development program are discussed in this report, but before any large effort is started, the necessary contents of the program would have to be defined.

(c) for the time being, assurance can be placed on existing types of reactor safeguards, principally emergency core-cooling.

The purpose of the small-scale effort would be an improved understanding of the related phenomena and possibly a definition of the content of a larger program. A larger program should be undertaken only if it can be shown to have adequate prospect of success.

CONCLUSION 1 - PHENOMENA ASSOCIATED WITH LOSS-OF-COOLANT

Current technology is sufficient to enable predicting with reasonable assurance the key phenomena associated with the loss-of-coolant; for quantitative understanding of the accident, the analysis of such an event requires that the core be maintained in place and essentially intact to preserve the heat-transfer area and coolant-flow geometry. Without preservation of heat-transfer area and coolant-flow geometry, fuel-element melting and core disassembly would be expected. With the start of core disassembly there would be a major increase in the uncertainty of prediction of core behavior, and degeneration of the core to a meltdown situation could not be ruled out.

Although basic analytical techniques are available to adequately predict the complex behavior characteristics of a loss-of-coolant, further assurance of the understanding of the event would result from additional experimental and analytical information. Hence, experiments in geometries representative of reactor coolant systems should be conducted, and more precise, analytical representations should also be developed.

CONCLUSION 2 - STRUCTURAL RESPONSE REQUIREMENTS TO BLOWDOWN

The mechanical or structural response to blowdown of key primary-system components must be such that the extent of deformations which could occur do not interfere with effective cooling of the core, do not preclude reactor shutdown, and do not cause further consequential primary system damage. The structural integrity of emergency core-cooling systems themselves must also be such that emergency core cooling can be accomplished.

As discussed in Conclusion 1, it is within the state of technology to predict, within conservative bounds, the hydraulic forces associated with blowdown. Methods are available for predicting the structural response to these forces, including prediction in the region of limited plastic deformation[1,2]. The magnitude of these forces is within the range that can be handled with practicable engineering design. Designs involving more extensive plastic deformation should not a priori be excluded, but the extend of deformation is currently difficult to predict.

CONCLUSION 3 - REQUIREMENTS OF EMERGENCY CORE-COOLING SYSTEM

The design requirements for the emergency core-cooling system must be:

a. First, to terminate in a loss-of-coolant accident core-temperature transients which could otherwise result in the loss of a definable core heat-transfer and coolant-flow geometry;

b. Then, to reduce the core to emergency core-coolant temperatures; and,

c. Finally, to maintain the core in this condition until full recovery from the loss-of-coolant accident is achieved.

It is important to recognize that fulfillment of the first requirement necessitates the prevention of bulk melting of the clad. At the present time and in the context of present peaking factors, a conservative interpretation of this requirement would be that the emergency core-cooling system be designed to prevent clad melt. Currently, the accepted procedure for fulfillment of the above requirement is to analytically demonstrate by means of a conservatively bounded evaluation that the core cladding in its normal geometry does not melt. This procedure is considered to be sufficient. However, it must be emphasized that this interpretation of "no clad melt" is not a requirement in itself since it may be possible to demonstrate that temperature transients can still be terminated in the presence of some clad melting; and that therefore, the overall objective for emergency core cooling would be satisfied.

CONCLUSION 4 - TECHNOLOGY OF EMERGENCY CORE COOLING

Sufficient test data are available to indicate that the phenomena of spray cooling and flooding represent satisfactory approaches to emergency core cooling. The implementation of these phenomena as cooling techniques is amenable to experimental verification. While there has been considerable effort expended in such experimental verification of core-cooling techniques, further testing at higher temperatures and degenerated conditions as well as general evaluations should be conducted. These are identified in the following discussion.

CONCLUSION 5 - PRACTICABILITY OF EMERGENCY CORE-COOLING SYSTEMS

The requirements for emergency core cooling are such that it is practical to design adequate emergency core-cooling systems within the current engineering technology.

The determination that the emergency core-cooling system used on a particular plant will be adequate required detailed systems engineering evaluation. It is suggested that the elements of this evaluation be developed into a standardized procedure to insure that the evaluation is complete in all cases.

CONCLUSION 6 - RELIABILITY ANALYSIS

The concept of reliability analysis has proven a useful and effective tool for systems evaluation in other industries. It is concluded that this concept can likewise be used to similar advantage in the assessment of emergency core-cooling systems. It would be of particular use in the relative comparison of systems and would also serve to aid in the identification of areas within a system network which are critical to its reliability. It is, therefore, recommended that the necessary reliability discipline and techniques be established within the nuclear industry and that this be placed on a formal basis to facilitate its implementation.

CONCLUSION 7 - PRIMARY SYSTEM INTEGRITY

A main line of defense against the possibility of a core meltdown is the integrity of the primary system boundary. Much has been done already to assure an acceptable level of integrity; however, the large number of plants now being constructed and planned for the future makes it prudent that even greater assurance be provided henceforth. Accordingly, we recommend that improvements, of the types suggested below, be made both from a short-range and long-range standpoint.

<u>Short Range</u>

a. As a minimum, those parts of the primary system whose failure could lead to large breaks should be designed, manufactured, and inspected to the high degree of reliability comparable to that presently used for reactor vessels, and to the additional requirements enumerated below. The present efforts on preparation of nuclear piping and nuclear valve and pump codes should be expedited and these codes put into effect without delay to reflect these high standards. These standards should also be applied to those components critical to emergency core cooling. Thorough reviews of the design of each component and subsystem making up the entire primary coolant system should be made by a qualified group separate from the one that has responsibility for the design. This separate group could be within or without the same organization. These design reviews should also include systems and components other than the primary system which are critical to the problem of core cooling.

b. Adequate allowance should be made in the design and operation of components and systems for the effects on materials resulting from neutron irradiation, such as the shifts which occur in the nil-ductility transition temperature[1]. In addition, reactor vessel material, weldment, and heat-affected zone samples, should be included in the reactor vessel for periodically monitoring changes in reactor-vessel-material and weldment properties during the life of the vessel. These considerations should be included in an appropriate standard or code. It should be noted that safety limits and conditions to assure that a plant is operated within approved design limits have to be specified in Plant Technical Specifications as required for obtaining AEC operating licenses.

c. Further emphasis should be placed on using overlapping inspection techniques, on greater quality control, and on the training of inspectors and test personnel. Areas suggested for consideration include:

   (1) Apply more than one nondestructive-test method in order to increase the assurance of flaw detection where special considerations such as geometry, accessibility, or variation in test technique warrant. This overlapping in inspection could include, for example, the ultrasonic testing of weld joints as well as their radiography. In this connection it is urged that standards and procedures be established to further the use of ultrasonic testing in the inspection of primary components.

   (2) Establish qualification standards for all nondestructive-test inspectors and test personnel. (It is understood that the ASME Boiler and Pressure Vessel Committee is presently working on establishing such standards.) Such personnel should be required to formally pass these standards before they can be used to inspect any primary coolant component or system. Further, the personnel should be re-examined periodically (every two years) to assure that they are fully knowledgeable and up-to-date with all latest testing techniques and requirements.

   (3) Have a formal quality-assurance plan, prepared by the primary-component manufacturer and approved by the organization responsible for the plant design, which delineates the quality control that will be used in the manufacture of the component.

   (4) Establish a separate monitoring system to assure that all phases of the quality-assurance program for the manufacture of each component are fully implemented.

d. Review and upgrading of Section III of the ASME Code, other appropriate codes, and inspection standards should be performed frequently to keep pace with improvements in technology, design techniques, inspection methods, and test equipment. Require that such codes and standards be used by all fabricators of primary coolant components and systems. (Ultrasonic testing of plates and forgings is an example where the development of tighter inspection standards is underway.)

e. Prepare and keep on file accurate manufacturing and inspection records of primary system components signed by a responsible company representative.

f. Require a leak detection system (such as air-activity detectors) external to the primary system and not connected to it so as to provide early warning if a leak develops in the primary system. (Experience as summarized in Appendix 3 indicates that leaks occurring in the primary systems are small and any propagation would be very gradual.)

Long Range

In addition to the relatively short-range action outlined above, effort should proceed toward the development of reliable and repeatable in-service techniques and associated standards for detecting flaws in primary system components, especially reactor vessels, during plant shutdowns. It should be noted that effective utilization of such shutdown inspections will require a reference inspection before the component is placed in service. The purpose of the periodic inspections is to determine whether any change has occurred since the previous inspection. It is understood that a program on this subject is being initiated by the Pressure Vessel Research Committee together with fundamental work on pressure vessel materials.

CONCLUSION 8 - BREAK SIZE FOR EMERGENCY CORE COOLING DESIGN

a. We consider it unnecessary to assume that large and rapid failures will occur in any component or system which is designed, manufactured, inspected, protected against missiles, and operated in accordance with the requirements given in Conclusion 7 or their equivalent.

b. Because the record of conventional as well as nuclear plant performance to date clearly indicates that small leaks from a pressurized system can occur, we consider it necessary that back-up means be provided for introducing water into the primary system to assure continued core cooling.

c. In addition to a. and b., the emergency core-cooling system should also be capable of handling a large and rapid failure of those components and systems which are not designed, fabricated, inspected, protected against missiles, and operated in accordance wtih Conclusion 7 or its equivalent.

d. We expect that, as recommend herein, more and more elements of the primary system will be designed, manufactured, and inspected to the same degree of high standards as required by Section III of the ASME Code, its revisions in process, and additional requirements such as those recommended in this report, to give the same reliability as reactor vessels. This evolution, which will further assure primary system integrity, should make it possible to design emergency core-cooling systems for reduced break sizes, because large and rapid failure of components meeting the recommended standards will not have to be considered. Eventually, a minimum in the reduced break size would still have to be specified as an acceptable basis for designing emergency core-cooling systems. In establishing such a minimum, a prudent safety factor based on engineering experience and judgment should be used. We consider that even with this safety factor the minimum acceptable break size eventually will be considerably smaller than the current design basis.

CONCLUSION 9 - SAFEGUARDS ROLE OF CONTAINMENT

The present concepts of containment, with their cooling systems, can provide an adequate barrier to the release of fission products to the environs when emergency core-cooling systems fulfill their design objectives. Both energy release and fission product release can be effectively contained.

Since the performance of the containment as a safeguard system is related to the performance of the other safeguard systems, we recommend that its design basis be chosen accordingly. Containment design should be based upon the energy released by the coolant, decay heat, and metal-water reactions consistent with functioning of emergency core-cooling systems and a prudent safety margin.

CONCLUSION 10 - CORE MELTDOWN

If emergency core-cooling systems do not function and meltdown of a substantial part of an irradiated core occurs, the current state of knowledge regarding the sequence of events and the consequences of the meltdown is insufficient to conclude with certainty that integrity of containments of present designs, with their cooling systems, will be maintained.

CONCLUSION 11 - COUNTERMEASURES PRIOR TO LOSS OF CONTAINMENT INTEGRITY

Although containment integrity cannot be assured in the event of a postulated core meltdown, a significant period of time may elapse before breachment of the containment occurs. It may be possible to develop preventive measures which are effective during this period and which could reduce the hazards resulting from subsequent failure of the containment. The desirability of utilizing such systems and the merits of requiring containments to be designed to assure such time availability should be evaluated after the effectiveness of these systems has been established through necessary development work. The use of such safeguards will depend on weighing their merits with those of other safety features to obtain the desired objectives in overall reactor safety.

CONCLUSION 12 - HANDLING OF LARGE MOLTEN MASSES

Reliable and practical methods of handling large molten masses of fuel for long periods of time do not exist today. The desirability of seeking such methods in order to improve the independence of the containment as an engineered safeguard should be considered in the light of primary system integrity and emergency core cooling effectiveness. It should be recognized that effective means of holding the molten core are not in themselves adequate to prevent containment violation from overpressure.

A report dated December 4, 1967 from the Regulatory Staff to the ACRS provided the Staff conclusions concerning the Task Force report. Reproduced below is the summary from the Regulatory Staff report.

SUMMARY OF REGULATORY STAFF CONCLUSIONS CONCERNING ADVISORY TASK FORCE REPORT ON EMERGENCY CORE COOLING

Conclusions 1-6 - Emergency Core Cooling

We are in general agreement with the Task Force conclusions on the technology and practicability of presently designed emergency core cooling systems. We are, however, encouraging efforts directed to developing improved concepts for these systems. We believe that the present R&D effort and rate of development of analytical techniques are generally adequate to resolve the questions remaining in this area. We will continue working with R&D and industry to insure that the safety research projects continually reflect our needs in this respect.

The Task Force recommendation on design requirements of emergency core cooling in general terms spells out a reasonable set of objectives for these systems. We believe, however, that an additional requirement should be to reduce the core temperature quickly enough to prevent excessive metal-water reactions. We further believe the concept of providing a greater margin for smaller, higher probability breaks should be included in the requirements of these systems.

We agree that the use of reliability analysis techniques, in conjunction with engineering judgment, may prove to be a useful tool in the design and evaluation process. We have supported effort under the AEC's safety research and development program to develop data and techniques applicable to these methods and are working within the staff to increase our understanding of this approach. We believe, however, that at the present time the development of these techniques and the availability of necessary input data is inadequate to rely solely upon this approach.

Conclusion 7 and 8 - System Integrity

We believe that the recommendations made by the Task Force to improve reactor coolant and emergency core system integrity are worthwhile. For the most part these recommendations have been incorporated in existing codes by way of revisions or are at advanced levels of consideration by code committees for future revisions; or either have been developed by the regulatory staff or are under preparation by the regulatory staff in the form of supplementary criteria.

The Task Force concluded that it should not be necessary to design emergency core cooling systems with the capability to handle large and rapid failures of those reactor coolant system components which meet the requirements to achieve increased integrity recommended by the Task Force. Hence, the Task Force argues that it will be possible in the future to design emergency core cooling systems for a reduced reactor coolant system break size. In considering the arguments presented with this conclusion, we note that (1) the recommendations for upgrading reactor coolant system components presented in the Report do not represent a major difference from that already proposed or planned; in short, there are no suggestions which are expected to have a major unexpected impact upon the integrity of these systems, and (2) the logic presented in support of a reduced break size is based upon the assumption that piping, pumps, and valves and other equipment of primary systems can achieve the same level of integrity of pressure vessels by applying similar methods of design, manufacture and inspection. Actually, from certain postulated external causes, as, for example, seismic loads, it may be that the less massive piping systems are inherently more vulnerable to damage than the reactor vessel. Our conclusion, therefore, is that although it may eventually be possible for piping systems to achieve equivalent levels of integrity as that of the reactor vessel, until substantial operating and testing experience and the results of in-service inspection have been accrued, the design basis for the emergency core cooling system should continue to be the same as now, even at the upgraded integrity levels for the reactor coolant system.

The Task Force recommended a design basis for containment which would be based upon the energy released by the coolant, decay heat, and metal-water reactions consistent with functioning of emergency core cooling systems and a "prudent" safety margin. The present staff approach to containment design includes the elements suggested in the Task Force Report but adds (in what might be considered a safety margin) an energy and mass input at a prescribed rate and for a prescribed time. These inputs can be related to metal-water reactions (assuming no emergency core cooling) or to other sources as, for example, a steam generator break. The staff feels that the basis for containment design should continue to reflect the need for accommodation of large energy and mass inputs, in addition to those recommended by the Task Force, until the time that experience can verify the confidence reflected in the Report that a loss-of-coolant accident will not progress to the point at which these allowances would be needed.

Conclusions 9-12 - Containment

In connection with core meltdown problem, the Task Force recommends that although a long-range basic research effort should be considered, the level and scope should be defined only after a more detailed study of the applicable research techniques. The Report lists several phenomena recommended for study. We agree that the suggested phenomena are worthy of some exploration and are planning to discuss with RDT personnel, the initiation of new projects within the safety research program to incorporate the topics suggested. We believe, however, that such experiments would be better directed and more productive if conducted in parallel with a design study effort aimed at developing, in concept at least, reliable and practical methods of containing large heat generating molten masses. We are not as pessimistic as the Task Force with regard to the possibilities of developing such methods, since this problem has not yet received careful and considered study.

In another approach to this problem, the Task Force notes that, since the radioactive fission products and not the molten mass constitute the hazard, it might be possible, in the time expected to be available prior to a melt-through situation, to remove fission products from the molten mass and thereby reduce the hazards resulting from the subsequent failure of the containment. Some of the preventative methods suggested with this recommendation may be usefully applied in conjunction with developing methods for handling large molten masses. It is our opinion, however, that this approach does not constitute a satisfactory substitute for developing methods for handling large molten masses and should be resorted to as a primary effort only if the development of such methods is not successful.

At the 94th meeting, February 8-10, 1968, the ACRS completed a letter report to the AEC Chairman Seaborg on the report of the Task Force; the letter is also duplicated on the following page.

Of particular interest from the point of view of the "China Syndrome" are the last three paragraphs of the ACRS report, in which the Committee strongly recommends that a positive approach be adopted towards studying the workability of protective measures to cope with core meltdown. In the next to the last paragraph, the Committee notes that, while the Task Force report presents considerable information on phenomena associated with large scale core meltdown, there is little examination or discussion of the degree to which the efficacy of core cooling systems might be augmented by way of design modifications, and similarly the report does not provide recommendations on design approaches to cope with large molten masses of fuel, or on the particular research and development problems related to these approaches. And, finally, the Committee recommends that additional design and development effort be aimed at means of providing protection against the extremely low probability type of loss-of-coolant accident in which emergency core cooling systems of current design may not be effective.

To be more blunt, the ACRS was saying that the Task Force had not provided any answers to the issue of ameliorating the effects of core meltdown or coping with core meltdown. Also the Task Force had not defined a research and development program which could provide answers in this regard, and the Committee was reiterating its previous stated recommendations that this be done.

In effect, eighteen months had passed with no new effort toward coping with core meltdown; and, in effect, there had been a negation of the recommendation in the draft ACRS letter of August/September, 1966, namely

> that still greater protection of the public by some independent means be provided, particularly for reactor sites near the population centers--that progress for this objective would require an evolutionary process of design and a vigorous program of research, both of which should begin immediately and should be aimed at reaching a high state of development in approximately 2 years.

What is perhaps unfortunate is that lack of study of the problem left the nuclear industry, the regulatory groups, and society in general, in a poor position to judge whether significant improvements in safety were feasible, and hence, in no position to make an educated decision on whether such additional measures should be considered. On the other hand, there was considerable school of thought that LWR's were already adequately safe, if not more safe than necessary when compared to other existing societal risks.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

February 26, 1968

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT OF ADVISORY TASK FORCE ON POWER REACTOR EMERGENCY COOLING

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards offers the following comments on the recently issued Report of the Advisory Task Force on Power Reactor Emergency Cooling.

The Committee believes that the Task Force has performed a valuable service by assembling in a single document discussions covering many of the problems associated with postulated loss-of-coolant accidents and the phenomena important to proper functioning of emergency core cooling systems. Also, the Task Force has reviewed in a useful manner the many phenomena involved in the course of a postulated large-scale core meltdown.

Certain of the report's conclusions and recommendations appear to constitute expressions of judgment as to the adequacy or sufficiency of particular reactor safety provisions in respect to their capability for providing assurance against undue risk to the health and safety of the public. No attempt is made to comment on these. There are, however, a number of other conclusions in the report concerning which the Committee wishes to recommend emphasis, supplementation, or a differing viewpoint. Comments on these are given below.

In Conclusion 1, the report states in connection with the loss-of-coolant accident: "... for quantitative understanding of the accident, the analysis of such an event requires that the core be maintained in place and essentially intact to preserve the heat-transfer area and coolant-flow geometry. Without preservation of heat-transfer area and coolant-flow geometry, fuel-element melting and core disassembly would be expected. With the start of core disassembly there would be a major increase in the uncertainty of prediction of core behavior, and degeneration of the core to a meltdown situation could not be ruled out." The ACRS is in substantial agreement with this observation.

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With respect to assuring that the core remains essentially intact during a loss-of-coolant accident, the report emphasizes the importance of properly assessing and designing for the hydraulic effects incurred, and lists several important specific aspects of the problem that must be recognized and dealt with in designing to cope with such effects. The ACRS agrees with this emphasis.

The possibility of fuel element failure from high internal pressure and high clad temperature during a loss-of-coolant accident is mentioned. In this connection, the Committee notes that present license applications show that a large fraction of fuel rods may fail in such accidents even though the emergency core cooling system works as designed. The Committee believes that, in addition to the work proposed by the Task Force, further research is needed to ascertain the modes of fuel rod failure and to determine that failures will not propagate or tend to block coolant flow excessively.

Conclusion 2 discusses further the importance of controlled and acceptable structural deformation during reactor blowdown in a loss-of-coolant accident. The ACRS agrees with this and calls attention to the need for considering deterioration during the life of the reactor and the role that periodic inspection could play in alleviating this potential difficulty. Also, more conservatism in design and fabrication may be needed where structural member response to accident-induced hydraulic forces is not testable. Further, the Committee continues to be concerned with the possibility of thermal shock effects on the pressure vessel, or other parts of the primary system, as a consequence of the rapid introduction of emergency cooling water in the unlikely event of a large primary system rupture.

The Committee endorses Conclusion 4 which recommends further testing of emergency core cooling at higher temperatures and for degenerated conditions such as core distortion.

The systematized approach to the design and evaluation of emergency core cooling systems described in Conclusion 5 appears potentially useful. However, deliberate allowance should be made for the possibility of aggravated accident conditions introduced by possible design errors, by weaknesses common to redundant components, or by other unexpected conditions, and full attention should be given to the potential advantage of diverse approaches to the design of emergency core cooling subsystems. It should be recognized, also, that new design features may introduce new potential safety issues in specific reactor designs.

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The Committee endorses the recommendation of Conclusion 7 for improvements in primary system integrity to reduce still further the already low probability of primary system boundary failure.

The ACRS agrees with the statement of Conclusion 10 that: "If emergency core-cooling systems do not function ... the current state of knowledge regarding the sequence of events and the consequences of the meltdown is insufficient to conclude with certainty that integrity of containments of present designs, with their cooling systems, will be maintained." Recognizing that absolute certainty cannot exist concerning any facet of safety, the Committee strongly recommends that a positive approach be adopted toward studying the workability of protective measures to cope with core meltdown. Basic safety research experiments would provide valuable insight and, possibly, direct attention to potentially profitable avenues of design which eventually could lead to substantial additional protection in this area. The proposal in Conclusion 11 for study of preventive measures to be made effective prior to loss of containment integrity to minimize the ultimate hazard is a helpful step in this direction.

In summary, the Task Force Report presents considerable information of interest on primary system integrity, key phenomena effective during loss of coolant and core heatup, functional considerations for emergency core cooling systems, and phenomena and effects associated with large-scale core meltdown. However, there is provided little examination or discussion of the degree to which the efficacy of core cooling systems might be augmented by way of design modifications or new design concepts. Similarly, the report does not provide recommendations on design approaches to cope with large molten masses of fuel, or on the particular research and development problems related to these appraoches.

The ACRS endorses the Task Force recommendations for improvement in primary system boundary integrity and for additional research and development work on emergency core cooling systems. The Committee further recommends, as it did in its 1966 report on safety research, that a vigorous program be aimed at gaining better understanding of the phenomena and mechanisms important to the course of large-scale core meltdown. It also recommends that additional design and development effort be aimed at means for providing protection against the extremely low probability type of loss-of-coolant accident in which emergency core cooling systems of current design may not be effective. The ACRS urges that these matters be pursued vigorously by manufacturers of nuclear equipment, the electric utilities, and the AEC, as appropriate.

Sincerely yours,

/s/

Carroll W. Zabel
Chairman

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The report by the Task Force on Power Reactor Emergency Cooling was used for policy decisions by the Atomic Energy Commission during the ensuing years in that the AEC placed its emphasis on improvements in quality control and improvements in emergency core cooling systems,* both of which had been recommended by the ACRS in its letters of August 16, 1966 on Dresden 3 and Indian Point 2.

That the judgments reached by the task force were subject to technical flaws ws already clear by early 1968. For example, the Task Force had missed the fact that the clad might embrittle at temperatures far below its melting point, thus requiring peak clad temperatures far below the melting point. With time other flaws developed. For example, consider Task Force conclusion 1 which states that current technology is sufficient to enable predicting with reasonable assurance the key phenomena associated with the loss of coolant, to provide quantitative understanding of the accident, etc. Actually, our knowledge of LOCA-ECCS in 1967 did not include the very important effects associated with steam binding and ECCS water bypass in PWR's, and had important omissions

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*On April 24, 1968, in a memorandum from Beck to Shaw the Regulatory Staff re-iterated strong support (perhaps for the last time) of an R&D program on means of handling large masses of molten fuel. The relevant paragraph is excerpted below.

> As a parallel effort, we again recommend that the handling of molten fuel be investigated. As indicated by the Emergency Core Cooling Task Force Report, methods are not currently available for handling the large masses of molten fuel which might result from an extensive core meltdown; however, we believe that with careful study and a vigorous engineering approach such methods can be developed. As we indicated in our report to the ACRS on this subject,* in view of the gaps that still exist in verifying the effectiveness of emergency core cooling systems and the absolute dependence currently placed on these systems, we believe work to develop such methods should be initiated. We also note that the ACRS in its letter on the Advisory Task Force Report** stated the Committee strongly recommends that a positive approach be adopted toward studying the workability of protective measures to cope with core meltdown. We are most willing to work with you closely on this matter, including providing additional information and assisting you in developing your study effort and any associated safety research programs.

On the other hand, in a report from the Director of Regulation, Mr. Price to the Commissioners dated February 20, 1969, it is specifically noted "there are no current plans to study events following large-scale core meltdown".

with regard to the behavior of ECCS for BWR's. Similarly, problems with dynamic forces, both in PWR sub-compartments and in pressure suppression systems for BWR's were not foreseen by the Task Force. Nevertheless, this report was called on to support policy decisions which led to little or no work on the examination of possible measures for ameliorating or coping with core meltdown, from whatever cause.

In various ways, over the next few years the ACRS attempted to get additional information developed on the phenomena associated with large scale meltdown and the possible design measures that could be taken to ameliorate large core meltdown. For example, in a letter dated April 14, 1967 to Mr. Milton Shaw of the AEC from ACRS Chairman Palladino concerning the water reactor safety program summary description, the Committee says

> It is not clear that substantial early effort will be devoted to gaining an understanding of the various mechanisms of potential importance in describing the course of events following large-scale core melting, including steam explosions and hydrogen generation. Information should be gained which would provide a better foundation for assessing the possibilities of coping with large-scale core melting.

In a report dated March 20, 1969 from Acting Chairman Hendrie to AEC Chairman Seaborg, the Committee forwarded comments on the water reactor safety research program. On page 6 of these comments, which were also sent to AEC General Manager Hollingsworth, the ACRS says

> With regard to containment of molten cores, no AEC work is currently planned. Interest in this work continues, however, and the problem may be a more critical concern for larger reactors and much more populated locations than are used at present. Some work in this area in the nature of scoping studies and possible solutions is appropriate. Also, research aimed at providing a better understanding of the more important phenomena involved should be undertaken.

In a letter dated November 12, 1969, from ACRS Chairman Hendrie to AEC Chairman Seaborg, the Committee reiterates its statement made in the previous letter of March 20, 1969, and notes that the ACRS had strongly recommended safety research of this kind several times during the last 3 years. The Regulatory Staff had also strongly supported such work, however, only smaller modest efforts had been initiated at that point. The ACRS goes on in the letter of November 12, 1969 to say

The Committee further recommends in its comments of March 20, 1969 that consideration be given to research and development specifically aimed to improving the potential for siting of large water reactors in more populated areas than currently being utilized. For example, studies should be undertaken to develop reactor design concepts providing additional inherent safety or possibly new safety features to deal with very low probability accidents involving primary system rupture followed by a functional failure of the emergency core cooling system. It appears that because of funding limitations and for other reasons the recommendations of the ACRS will not be implemented at this time.

In 1970 a draft report, concerning the various phenomena important in consideration of core meltdown and the possible design of features to prevent loss of containment integrity in the presence of core meltdown, was prepared by Battelle Memorial Institute, Columbus, under the auspices of the Atomic Energy Commission. This report represented a very good collection ofthe information available up to that time. It did not include much in the way of design approaches, although there were some considerations of that sort. Shortly after this draft became available, the ACRS arranged, through the AEC Division of Production, to have a short study done on its behalf by members of the Savannah River Research Laboratory, which was operated by the du Pont Company. On September 16, 1970, a presentation entitled "Concepts for Mitigation of Postulated Power Reactor Core Meldown Accident," was given by representatives of du Pont at a meeting of the Reactor Research Safety Subcommittee of the ACRS. The summary of this meeting says that

> Dupont believes it is feasible to control the situation involving molten uranium. All of the tentative concepts du Pont has devised depend upon active components and require both power and a cooling water supply. One system considered by du Pont includes the possible use of steel troughs having a 4" width, a depth of 2', and a length of 4'; the troughs will be under water and provide a large heat transfer area.

The summary goes on to say

> du Pont believes that further design studies would require an intimate knowledge of modern power reactors. Because of this and a reduction in engineering staff, du Pont appeared reluctant to perform additional studies themselves but recommended that as the next (or second) step, three years of research and development be carried on 1) properties and molten material, 2) reaction of molten material with water and 3) methods of dose reduction with vented gases. Du Pont believes that these studies would cost one to two million dollars. They recommend that, if a third step were to be carried out later, it should be an engineering effort to provide a specific design concept for a specific reactor. They believe that this would cost a factor of ten more than the above research and development effort.

It's not completely clear whether, by the performance of this study, friction was generated within the Atomic Energy Commission. The Director of the Division of Reactor Development and Technology, Milton Shaw, was positively against any such efforts; and this, in effect had been a short end run around his position. In any event there was great difficulty in getting any such further work performed in this way for the ACRS.

The ACRS met with Shaw and members of his staff, and with representatives of the Battelle Memorial Institute and du Pont during the 129th meeting, January 7-9, 1971, to discuss the matter of postulated core meltdown accidents. Following this meeting, in a memorandum from R. F. Fraley, the ACRS Executive Secretary, to Shaw, dated January 11, 1971, the ACRS presented its recommendations. The memorandum is duplicated on the following page.

In summary, the ACRS found that both Battelle and du Pont had separately concluded that it appears technically feasible to mitigate the consequences of a core meltdown accident. The Committee stated that it believes that even though a core retention system may not be effective for all causes and modes of core meltdown, it could, as an independent backup, decrease the probability of an untenable fission product release to the environment by at least an order of magnitude, a result that becomes increasingly difficult to achieve by refinement of systems designed to preserve core integrity within the reactor vessel.

Shaw responded a year later in a letter dated February 3, 1972 to Fraley. He effectively refused to follow the recommendations of the ACRS with regard to studies concerning core meltdown. A copy of this rather lengthy letter is included, since it rather clearly indicates another point of view concerning this situation. And perhaps it gives some insight into why it was essentially impossible to get any effort by the AEC on this problem over a number of years.

The ACRS chose not to respond directly to this letter from Mr. Shaw, but in a letter dated February 10, 1972 from ACRS Chairman Seiss to AEC Chairman James Schlesinger, the Committee notes that "although the ACRS has recommended that research and design studies be undertaken on systems which might be capable of coping with a largely molten core, little such work appears to be underway." There appears to have been no change in the reactor safety program with regard to this matter under AEC Chairman Schlesinger.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 11, 1971

Milton Shaw, Director
Division of Reactor Development
and Technology

ACRS COMMENTS ON A CORE RETENTION SYSTEM TO MITIGATE THE CONSEQUENCES OF A CORE MELTDOWN

The Advisory Committee on Reactor Safeguards appreciates the meeting with you, members of your staff, and representatives of Battelle Memorial Institute and E. I. du Pont de Nemours and Company, on January 8, 1971, to discuss the matter of postulated core meltdown accidents. As you know the Committee has had a continuing interest in this matter.

Following its meeting with you, the ACRS further reviewed the usefulness and feasibility of a core retention system to mitigate the consequences of a core meltdown. The Committee agrees with you that quality assurance, including assurance of proper functional performance, of present systems having safety functions is of primary importance. The Committee believes that the probability of meltdown with present systems is very low, and that more stringent application of principles of quality assurance will make the probability still lower. However, improvements to systems and system quality cannot lead to continued significant increase of safety without limit; external phenomena, unforeseen events of very low probability, common mode failures, and human error will set a practical limit to system reliability.

The Committee believes that even though a core retention system may not be effective for all causes and modes of core meltdown, it could, as an independent backup, decrease the probability of an untenable fission-product release to the environment by at least an order of magnitude, a result that becomes increasingly difficult to achieve by refinement of systems designed to preserve core integrity within the reactor vessel.

The Committee has found the work by Battelle and du Pont to be very helpful in its considerations. Both groups have separately concluded that it appears technically feasible to mitigate the consequences of a core meltdown accident. Both groups have recommended that, if work in this area is to be continued, the logical next step is to choose one or possibly two design approaches which appear to have the best potential of success, to evaluate this design in greater depth, and to pursue an associated research and development program organized so as to obtain information vital to the success or failure of the particular design approach. The ACRS believes that the results of the studies thus far are encouraging.

The Committee recognizes that physical and physicochemical properties of the molten fuel and structural materials would be required before a good conceptual design could be made, but it is believed that present knowledge of these properties may be sufficient to establish basic feasibility. However, there appear to be other major uncertainties that do affect basic feasibility. For example, sudden admission of a stream of molten fuel into water (even hot water), especially in a manner that can trap water under the fuel, as in steel retaining channels, could lead to a steam explosion of such violence as to make the entire scheme impractical. The Committee believes that, early in the program, such problems should be explored qualitatively; for example, with material that can be readily melted in conventional furnaces, but using quantities that are large enough to give confidence in the results.

The Committee believes that a program of conceptual design studies and analyses for a core retention system, coupled with the kind of exploratory experiments cited above, should be undertaken. One to two million dollars over a period of two to three years might be a reasonable estimate of the effort and time scale to accomplish this step. The Committee recommends that a program of this type be undertaken with a completion goal of 1973-1974 for this phase. The Committee believes it important that the group undertaking the task have considerable background and resources in practical engineering and metallurgy, as well as a strong research and development capability.

R. F. Fraley

R. F. Fraley
Executive Secretary

cc: R. E. Hollingsworth, GM
H. L. Price, Director, REG

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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Advisory Committee on Reactor Safeguards
U.S. A.E.C.

Mr. Raymond F. Fraley
Executive Secretary
Advisory Committee on Reactor Safeguards
to the U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Fraley:

We appreciate the comments and recommendations of the Advisory Committee on Reactor Safeguards contained in your letter of January 11, 1971 on core retention systems to mitigate the consequences of a core meltdown. We also appreciated the opportunity to discuss our thoughts on this matter with the Committee. As we indicated, this reply is intended to summarize our comments during those discussions and in subsequent discussions with the Safety Research Subcommittee. These discussions have been helpful and I regret that we have been so late in replying officially to your letter.

We agree with the Committee that occurrences which might be mitigated by an additional engineered safety feature, such as a core catcher, are of very low probability. We also tend to agree with the Committee's further observation that there is a practical limit on how far it is possible to go in providing protection against very serious and unlikely large scale accidents, but note that this agreement pertains primarily to special add-on safety systems to cover various types of postulated events, and not necessarily to other means of providing meaningful protection such as by basic plant design changes, strengthening standards and criteria and their implementation, along with strong administrative procedures to provide additional assurances of obtaining the full benefits of practices imposed by Levels One and Two* design and safety measures. Thus, we cannot conclude that such a limit has necessarily been reached in the

*In previous discussions with the Committee and others, Three Levels of safety have been defined; this and subsequent references are consistent with the definitions of these previous discussions.

light water reactor industry, based on the current record of design, construction and early operation of light water reactor power plants, on the current and potential types of errors, flaws, discrepancies and other problems and concerns, and on the positive steps that can be taken to improve such protection if an undesirable limit is being reached. Moreover, if concerns or evidence available to the Committee indicate that an undesirable limit is being reached for any specific plant or group of plants, we urge that the Committee make such detailed information known and also that the Committee, in that event, recommend more stringent design, test, or administrative requirements be placed on specific applicants so as to reduce further the probability of a series of events contributing to the postulated large-scale accidents of concern. Most specifically, we cannot agree with the Committee's belief that this concern can be compensated for, nor that protection can be significantly increased, by the AEC's undertaking a research program on molten core retention. Our views in support of these positions are still essentially those that were discussed with the Committee previously and are summarized below.

The Ergen Task Force on Emergency Core Cooling concluded that the first line of defense against major accidents in water reactors is the reliability and integrity of the plant primary system, achieved through assurance of high quality in design, construction and operation. We agree with this conclusion and note that relative to Level Three design and surveillance uncertainties and difficulties it is straightforward to assure provisions for system integrity monitoring and failure detection and to require effective corrective actions when defects occur. In fact, routine operation of the primary system with concurrent assessment of integrity by direct or instrumented observations of leaks is a time-tested method of assurance of the integrity of systems of the specific high pressure coolant type with which we are dealing. Further, the causes and effects of primary system boundary failure, while still under study, are fairly well understood, particularly for those materials which have been used for these systems for many years, and have been demonstrated to be susceptible to satisfactory handling by known design, fabrication, operation and inspection techniques, and related disciplined quality assurance practices. The record of industry in this regard must be considered. To the extent that the causes and effects of such failures may not be sufficiently well understood increased efforts should be identified and given sufficient priority to promptly provide and assure the use of the information required for this understanding. To the extent that the requirements are not being implemented properly for a given reactor plant, or for a group of plants, increased efforts should be focused directly on these deficiencies and concerns.

Most importantly, the backup systems, particularly those which use water as a coolant, are likely subject to similar types of quality assurance uncertainties or failures to which the primary system is subject. Further, some of them are not as well understood considering the abnormal or hypothetical conditions being postulated for their design and operation. Moreover, they tend to detract in a number of ways from the probability of predictable performance of other interrelated and more vital plant systems. At the same time, such backup systems generally have the basic limitations of assurance of timely availability and the failure detection uncertainties normally associated with quiescent systems which are infrequently called upon to operate even though they, or parts thereof, may be tested routinely.

In agreement with the Ergen report conclusions, the AEC and the industry have mounted large and costly programs to provide technology, standards and emphasis related to obtaining initial and continuing assurance of primary system integrity. While encouraging responses from designers and suppliers of key reactor systems and components indicate that the industry is moving in the right direction, the much-too frequent difficulties encountered in obtaining the quality specified and in getting plants tested and on the line clearly show that further improvements in the achievement of adequate quality are still needed and can and must be obtained. Since there is no comparable substitute for such quality, efforts must continue, and are continuing, to provide guidance, encouragement and support in a substantive way to assure that adequately reliable systems are being built and operated. Moreover, where such efforts may not yet be satisfactory, there are many compelling reasons to insist on more management attention, more effort, special administrative and test procedures, or design changes to assure emphasis on these basic requirements rather than attempting to encourage reliance upon special new safety systems. In time, these will also have to be exposed to the same type of questioning regarding probability of predictable performance for various postulated events.

On the basic matter of primary system integrity, which everyone agrees is the first and most basic element of this concern, a major effort has been underway for several years to develop understanding of the potential failure mechanisms in heavy section piping and components used in light water reactor plants. Significant progress is being made in establishing process and inspection criteria to increase assurance that the higher probability small failures can be identified by monitoring and testing techniques, and major failures can be avoided. Furthermore, considerable additional understanding has been gained as to design, manufacture and operation of piping systems containing tees, valves, pump volute castings and the like to detect defects and to avoid severe failure. Full application of the additional insights thus gained can, we believe, reduce significantly the already low probability that these severe types of accidents, resulting in the potential concern for core meltdown, can occur.

As we have frequently noted, we would appreciate any comments on current development or implementation programs related to primary system integrity which would serve to improve or expedite them. Such comments would permit us and others to realign our resources to better close in on this important problem area and attain the additional confidence in system integrity to provide the necessary level of assurance of plant performance and public safety.

Because questions on assurance of primary system integrity will continue to arise, and in recognition of the Committee's point that it is not possible to eliminate all deficiencies in material and human factors even after disciplined monitoring and testing programs have been established, we recognize that it is prudent to provide backup safety features which will detect and arrest the course of accidents at a tolerable level of consequence and/or mitigate serious consequences. Also, it is essential to obtain administrative procedures, including independent audits, that provide assurance of such coverage. Here again, we have mounted a large effort related principally to understanding the severe loss of coolant accidents and emergency core cooling and containment response as well as encouraging the adoption of proven test and administrative procedures. These programs, like the ones related to system reliability and integrity, are continuing. Technical deficiencies and uncertainties in these areas should, in practice, continue to be compensated for by conservatism in ratings and pessimism in the analysis of the response of the reactor plant to fault conditions, or by appropriate administrative procedures. It should be noted that recent concerns about, and emphasis on, the assurance that ECC systems will function as desired and will not adversely interact with the surety of performance of other systems and components also important to plant safety illustrate the difficulties of reaching and sustaining definitive conclusions in such technical areas. These depend on continuing research and development programs with their analytical models based on small scale experiments to attempt to understand the course of highly complex and improbable events in real and individual reactor systems, over the life of each plant.

It is clear that a reduction of between one and two orders of magnitude in the probability of occurrence of a large release of fission products to the environment, as noted by the Committee, is a general goal which we could agree would represent additional assurance of public safety. However, there appears to be more than one way to achieve this goal particularly when faced with the increasing difficulty of obtaining agreement on defining the worth of the improvements now being made, and the extent to which others of this type are still required. Moreover, in view of this uncertainty, there also appears to be significant differences in how to estimate the potential value of additional efforts on achieving such reductions in each plant as well as the potential penalties associated with such approaches. As the Committee has noted, an additional engineering safety feature such as a melted core retention system might be developed. On the other hand, it is likely that substantially higher degrees of assurance of system integrity and efficacy of present engineered safety systems can be developed more easily and with greater confidence by increasing the probability that designs perform as planned. The principal factors potentially affecting the options appear as follows:

a. Proceeding along both paths at a necessarily reduced rate would undercut the programs related to increasing integrity and emergency core cooling. In addition, we are concerned that the considerable effort that would be required for a program to develop a realistic melted core retention system, even if it could be accommodated within available resources, would detract attention from recently intensified efforts to resolve the remaining questions and uncertainties related to preventing or arresting the large scale loss of coolant accident at an earlier and more predictable stage. Since these are the existing areas of apparent and identifiable weaknesses directly contributing to the concern, we cannot agree with an approach that detracts from efforts to eliminate these weaknesses.

b. The melted core retainer approach requires significant technical efforts in an area of much less well established technology than primary system integrity or emergency core cooling. Thus, efforts related to melted core retention are less likely to be successful and accepted in a timely fashion than those involving present core retention systems, i.e., primary system integrity and emergency core cooling.

c. The melted core retainer concept would, in all likelihood, be based on an analysis supported only by small scale experiments and could probably not be tested for adequacy of design nor for adequacy of performance in each plant after installation. Moreover, this concept would probably require the reliable functioning of active systems to approximately the same extent as emergency core cooling systems, and therefore would be subject to similar reservations related to predictable and successful performance under many postulated conditions similar to the type that are now apparently stimulating the search for an additional measure of safety.

d. The two contractors who recently have looked most closely at core retention have estimated the R&D program to cost between 15 and 20 million dollars exclusive of integrated tests. Based on past history with this type of estimating, these dollar estimates would most likely prove to be extremely low. The Committee's estimate of one to two million dollars would simply represent a down payment on an indefinite major commitment. Similar remarks apply to the projected schedule. Our current budgets do not provide for anywhere near this type of additional funding, even giving consideration to possible augmentation and industry support.

As a consequence of these considerations, we see no alternative but to continue with intensive efforts to address questions related to primary system integrity and emergency core cooling as the priority needs as identified in the Water Reactor Safety Program Plan and in supplemental more detailed documents. We suggest that the Committee could provide significant assistance in this work through renewed and increased emphasis on all of those aspects of design and construction which address these areas, particularly through emphasis on quality assurance provisions and the provisions for the proper discharge of the applicant's management responsibilities for quality assurance in design, construction, test, operation, inspection and maintenance activities.

We will be more than pleased to meet with the representatives of the ACRS, REG and the manufacturers to pursue these matters further, recognizing that the detailed designs and related quality assurance practices are really a key to the resolution of this matter, although noting that some very useful modifications and reorientation of ongoing R&D programs may be derived from such a systematic review. Pending such a review, we are continuing to strengthen the relevant R&D programs and encouraging and guiding others to address these concerns.

In addition to our discussions of these matters related to core retention systems for light water reactors, we have had numerous discussions with the ACRS and Regulatory staff on core retention systems for the LMFBR. Recently, we received the ACRS Report on the Fast Flux Test Facility dated 1/13/72. In agreeing with proceeding with construction, the ACRS recommended an intensive development program on an ex-vessel, post-accident core retention and cooling system for the FFTF, should the system be needed. As has been presented, our major effort on the breeder is to concentrate our limited resources on more basic and realistic approaches to achieve safety, and to be in a position to provide a backup capability, if needed, to assure through a rather extensive development program that the postulated debris from a hypothetical core disruptive accident can be cooled and retained inside the FFTF reactor vessel. As so indicated during our discussions, we believe that reasonable assurance can be provided over the next two year period that such efforts will enable us to achieve satisfactory resolution of these matters prior to FFTF startup, should such a backup capability be needed. In any event, we consider the matter of molten core retention in FFTF as a separate subject from the above response on light water reactor core retention and will address separate correspondence to this matter as required in the future.

Sincerely,

Milton Shaw, Director
Division of Reactor Development
and Technology

cc: L. Manning Muntzing, Dir. of Reg.
R. E. Hollingsworth, Gen. Mgr.
E. G. Case, Dir., DRS
J. J. Flaherty, AGMED
Chairman (2)
Comm. Ramey
Comm. Johnson
Comm. Larson
Comm. Doub

Actually, a meeting of the Reactor Safety Research Subcommittee was held on April 5, 1972, part of which was devoted to discussion of the February 3, 1972 letter from Mr. Shaw, although the principal topic of discussion was the manner in which the AEC reactor safety research program had been organized, administered and managed, and whether the ACRS should recommend changes.

Examples from the minutes of the April 5, 1972 Subcommittee meeting give some insight into Committee member thinking concerning the letter from Shaw.

> Dr. Monson referenced the February 3, 1972 letter from DRD&T which responded to ACRS comments relative to the development of a molten core retention system. Dr. Monson indicated he believed that two major issues should be addressed by the Committee. The first of these is to what extent DRD&T should decide Regulatory needs regarding reactor safety research. He thought the AEC Commissioners should make the decision as to what safety research should be performed by the AEC. He said that, if the Committee decides that not proceeding with work regarding a core catcher could lead to a failure of reactors to be acceptable to the Committee in the future, the ACRS should inform the Commission regarding this. Dr. Monson stated that the above DRD&T letter indicates that the Committee has said that a limit has been reached regarding improvement in primary system integrity but that the Committee has not done so. The DRD&T letter states that research should not be undertaken regarding the core catcher because reliance would have to be placed on analysis and on small experiments. On the previous page, the letter indicates that the problems of ECCS are being resolved by analysis and small experiments. In one place, it is stated that industry's record indicates that a large break can always be detected prior to rupture, and later in the letter there is discussion of how much is being done to increase assurance that small failures can be detected sufficiently early.
>
> Dr. Monson said that a second issue is whether the ACRS still believes that research and development should be performed on a core catcher. He indicated that approximately 75 reactors have been approved without such a device. The Committee might, however, say that larger or higher power density reactors or reactors located closer to metropolitan areas cannot be built without core catchers installed.

Dr. Monson said that the February 3, 1972 letter from DRD&T indicates that the ACRS is not an advocate of quality assurance. He said that the Committee has been worrying about quality assurance and the effectiveness of ECCS for a long time. Dr. Monson thought that the basic difference between the DRD&T position and that of the Committee is that DRD&T believes that as much as practicable should be done regarding improving primary system integrity, and, if that is not enough, a very good job should be done regarding the ECCS, and then nothing additional would be needed. Dr. Monson said that the Committee agrees with this position except for the last part. The Committee has never suggested that a reduction be made in the effort to improve QA or the ECCS, but the Committee wanted additional money spent regarding the development of a molten core retention system.

In a letter dated November 20, 1974 from ACRS Chairman Stratton to Honorable Dixie Lee Ray, the new Chairman of the Atomic Energy Commission, the Committee reiterated "its previous recommendations for research into phenomena involved in core melt down, including the mechanisms, rate and magnitude of radioactive releases, and the study of means of retaining molten cores or ameliorating the consequences. In this connection more knowledge of the possibility and extent of steam explosions in the presence of large quantities of molten fuel and steel is of particular importance." By the time this report had been issued, the draft version of WASH-1400, Reactor Safety Study (AEC, 1974) was available with its rather interesting analyses of the various possible paths, in addition to LOCA, that could lead to core melt, and also its estimates of the average consequences of accidents in light water reactors.

Finally, as a last bit of chronology, in a letter dated June 12, 1975 from ACRS Chairman William Kerr, to Mr. Ralph V. Carlone, Assistant Director of the U. S. General Accounting Office, the Committee commented once again on its recommendations for research into phenomena involving core meltdown as follows on the next page.

In summary, what we see is that there was roughly a decade between the time that very strong recommendations were made for not only research into means to understand what goes on with regard to an accident involving the full scale meltdown of the large core in the light water reactor but the actual recommendation that, if possible, a means to ameliorate or cope with this event be developed. Nearly a decade went by with essentially no response from the nuclear industry or the Atomic Energy Commission, and almost no response from the Nuclear Regulatory Commission until the very latter part of this era, when segments of research into phenomena were initiated, but no conceptual design studies were included. We will come back to this subject in connection with a look at the Reactor Safety Study and related matters.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 12, 1975

Mr. Ralph V. Carlone
Assistant Director
Resources and Economic Development Division
United States General Accounting Office
Washington, D.C. 20548

Dear Mr. Carlone:

This letter is in response to your letters of May 7, 1975 and June 3, 1975 which transmitted certain questions on which you sought the views of the Advisory Committee on Reactor Safeguards. The Committee completed its response to your questions at its 182nd Meeting, June 5-7, 1975. The Committee had the benefit of a Subcommittee Meeting in Los Angeles on May 30, 1975, at which representatives of the NRC Staff and of Aerojet Nuclear Corporation (ANC), the contractor responsible for LOFT construction, experiments and analysis, discussed the status of LOFT and other aspects of light water reactor safety research. Representatives of ERDA, EPRI, Westinghouse and Holifield National Laboratory were also present at the Subcommittee Meeting.

The Committee response to your questions is attached.

Sincerely,

WKerr

William Kerr
Chairman

Attachment:
ACRS Response to Questions

cc: Honorable William A. Anders,
Chairman, NRC

Question 1

"In its November 20, 1974, letter report on Water Reactor Safety Research, the Committee recommended more work on emergency core cooling systems, including conceptual design work, analytical studies, investigations of ways ECCS performance might be optimized, and assessments of the overall reliability of ECCS. Within the same report, the Committee reiterated previous recommendations for research into phenomena involved in core meltdown, including the mechanisms, rate and magnitude of radioactive releases, ways to retain molten cores, or ameliorating their consequences, and the possibility and extent of steam explosions in the presence of large quantities of molten fuel and steel.

"What was the basis of the Committee's recommendation in its November 20, 1974, report as to the importance of conducting research into the phenomena involved in a core meltdown?

"Which of these two areas, ECCS and core meltdown, is more important, and why? Should LOFT be used for research on the core meltdown phenomena described above?"

Answer

In its August 16, 1966, reports on the Indian Point 2 and Dresden 3 reactors, the ACRS first recommended major improvements in emergency core cooling systems, and strong measures to reduce the probability of loss-of-coolant accidents including improved primary system quality, expanded inservice inspection, and improved leak detection. In its October 12, 1966, report on the reactor safety research program the ACRS first recommended a vigorous research program on potential modes of interaction between sizeable masses of molten mixtures of fuel, clad and other materials with water and steam, on the mechanisms of heat transfer connected with such molten masses, and other related mechanisms and phenomena. The ACRS also recommended that studies be initiated to develop reactor concepts with new safeguards to deal with low probability accidents involving primary system rupture followed by a functional failure of the emergency core cooling system.

The second major recommendation of the October 12, 1966, report related to the need for improved understanding of the loss-of-coolant accident (LOCA) and the phenomena important to a proper functioning of ECCS. The third and fourth recommendations related to methods of better assuring pressure vessel integrity.

The ACRS has reiterated its recommendations for safety research in these areas on many occasions, most recently in its November 20, 1974, report on Water Reactor Safety Research.

In effect, these actions by the ACRS represented an effort 1) to reduce the probability of occurrence of an accident 2) to assure the presence of reliable, conservatively designed ECCS to keep core temperatures within acceptable limits, should a LOCA occur and 3) to obtain knowledge concerning core meltdown and possible design steps to cope with or ameliorate the consequences of this unlikely event as a possible means of providing still greater protection of the public health and safety. The ACRS believes that the importance of the items is in the order stated; that is, first in importance is the prevention of accidents, and second in importance is the mitigation of consequences, should an accident occur.

Core meltdown can arise from a variety of initiating events; hence, if measures could be developed to cope with core melt, accidents other than LOCA might also be mitigated.

The ACRS view has been that the ECCS must be designed to cope with a complete spectrum of postulated pipe breaks, including sudden gross rupture of the largest pipe. To attain assurance in this regard, the Committee has persistently given the matter much attention, both in licensing reviews and in recommendations for safety research. The expected performance of currently designed systems satisfies existing criteria; the ACRS, nevertheless, has urged that still more reliable and capable ECCS be developed (see ACRS reports on Interim Acceptance Criteria of January 7, 1972, and on Acceptance Criteria of September 10, 1973).

With regard to safety research on core meltdown for LWR's, very little has been done in the ensuing years since 1966. The absence of adequate knowledge of relevant phenomena and of any serious design efforts on plant changes to cope with or mitigate core meltdown has handicapped evaluation of the true potential for enhancing protection of the public health and safety in this regard.

The ACRS has been advised that the Reactor Safety Research Division (RSR) of NRC will initiate a new program on core meltdown phenomena at a funding level of $500,000 in FY-76. The ACRS believes that research on both ECCS and core meltdown is important, and that the effort on the latter should be expanded.

2.14 1967 IAEA SYMPOSIUM ON CONTAINMENT AND SITING

Several papers discussed engineered safeguards for LWR's from various points of view. Levy (IAEA, 1967) argued for a systems approach to containment design. He recognized that containment is only part of a total system, which prevents and limits the consequences of a release of fission products. Levy argued that containment performance requirements should be determined by looking at the overall system, rather than by making arbitrary assumptions about the source of the accident and the effectiveness of some of the provided features.

Levy called attention to the previously oversimplified interpretation of the role of containment as a last-ditch barrier. Interestingly, he suggested consideration of a controlled venting capability (through a containment fission product removal system) to prevent overpressurization, an idea which was re-emphasized by the American Physical Society Study Group Report on Light Water Reactor (Lewis, 1975) and is receiving considerable attention currently.

Levy's estimate of 0.99999 as the reliability of the ECCS was perhaps overly optimistic by today's standards.

Levy did not explicitly mention the "China Syndrome" but, as a lead engineer for General Electric and as a member of the Ergen Task Force on Emergency Core Cooling, he was very knowledgeable about the subject. The paper by Vinck and Maurer (Vinck, 1967), on the other hand, said

> The present stage of development is such - and it looks like this may last for some time to come - that one has to live with the potential of core meltdown, be it partial or complete. Therefore, the emphasis which is placed on secondary containment against fission product release and on the demonstration that it will be maintained under all circumstances is still justified.

Kellerman and Seipel (IAEA, 1967), presented a paper concerning containment reliability, and identified a large number of ways in which containment integrity might be lost in a LOCA, but they allowed the possibility of maintaining containment integrity, given a LOCA and a failure of the ECCS (and hence a molten core).

These papers are an illustration that at least at the time that the manuscripts were prepared, the inevitable loss of containment integrity with core melt (the loss of the "last-ditch independent barrier") was not universally recognized.

P. A. Morris and R. L. Waterfield of the USAEC Regulatory Staff presented a paper entitled, "Site Evaluation and Diffusion Calculation Procedures in the USA." (IAEA, 1967) It gave the most recent assumptions and typical results obtained in the application of 10 CFR Part 100. It discussed dose reduction factors required for various sites using the MCA release to an intact containment, never mentioning the "China Syndrome" problem.

Davis and Robb (IAEA, 1967) included a look at siting practice in the U. S. and decided that population cednter distance and population distribution have had little to do with nuclear plant siting; at least outside metropolitan areas. Table 1 from their paper is reproduced on the following page.

Three very interesting papers from the United Kingdom dealt with siting. Charlesworth and Gronow (IAEA, 1967) of the British Inspectorate for Nuclear Installations, provided "A Summary of Experience in the Practical Application of Siting Policy in the United Kingdom." They described the system of site classification, which compared the effects of postulated iodine release in terms of the most populated 10° sector and of the entire surroungind population density, using weighting factors for the expected dispersion with distance. (The United Kingdom was maintaining the requirements of a zone having limited population close to the plant, but was tending at that time to move from a policy of relatively remote siting (for the U.K.) to more populated sites).

Charlesworth and Gronow had the following conclusions:

> The time has not yet come when it can be pretended that decisions on the safety of a nuclear plant can be based more firmly on objective assessment than on judgement....
>
> When the consequences of a fault or accident cannot be securely limited or contained, then the chance of the fault or accident occurring must be sufficiently remote to justify taking the risk. To attempt to quantify this statement, it might be stated that the probability of such an accident must be such as to give an acceptable margin of safety over it occurring during the lifetime of the plant, and of course the margin must be wide enough to take into account the uncertainties of the probability figure. Before any decision can be taken as to the wisdom of expressing policy in such terms due regard must be given to the possible consequences which follow from enshrining figures of this nature often to the detriment of the original intention. One point is that if a certain probability figure is approved, any improvement required will be seen as illogical. Over the

Table 1

UNITED STATES NUCLEAR POWER PLANT SUMMARY

Station	Licensed Station Capacity MWe Net	Exclusion Distance Miles	Population Center Distance Miles	Population 0 to 5 Miles	Population 0 to 10 Miles
Turkey Point Units No. 3 & 4	1302	0.85	20	0	42,000
Palisades	700	0.44	16	4,500	15,000
Browns Ferry Units No. 1 & 2	2130	0.76	10	2,800	30,600
Peach Bottom Units No. 1, 2, & 3	2170	0.48	21	6,145	23,550
Point Beach Units No. 1 & 2	910	0.7	13	1,241	20,845
Diablo Canyon	1060	0.5	10	10	
Monticello	471	0.3	22	3,942	
Oconee Units No. 1 & 2	1678	1.0	21	2,163	36,334
H. B. Robinson Unit No. 2	663	0.26	56	10,800	
Burlington	993	0.23	11	119,370	17 mi.-Phil.
Fort St. Vrain	330	0.40	14	1,951	8,420
Vermont Yankee	514	0.17	25	7,400	29,200
Quad-Cities Units No. 1 & 2	1430	0.23	5	5,369	
Indian Point Units No. 1 & 2	1128	0.32	1	53,040	155,510
Dresden Units No. 1, 2, & 3	1630	0.5	14	2,500	23,000
Connecticut Yankee	462	0.32	9	10,000	49,500
Oyster Creek	515	0.25	25	38,500	106,500
Nine Mile Point	500	0.75	6	1,900	30,900
Millstone Point	549	0.4	3	60,000	96,000
R. E. Ginna	412	0.29	12	1,500	8,000
Malibu	462	0.20	10	6,000	11,900
La Crosse	50	0.21	15	1,000	7,500
San Onofre	430	2.0	10	8,800	22,000
Shippingport	90	0.4	7	20,000	22 mi.-Pitts.
Yankee	175	0.5	21	2,000	30,000
Big Rock Point	72.8	0.5	135	5,000	9,000
Elk River	23	0.23	20	8,000	30 mi.-Minn.
Carolinas-Virginia	17	0.5	25	2,000	8,000
Enrico Fermi	60.9	0.75	7	9,000	61,000
Humboldt Bay Unit No. 3	70	0.25	3	35,000	40,000
Piqua	11.4	0.14	27	21,000	42,000
Pathfinder	58.5	0.5	3		

whole range of releases, effort must still be made to reduce the uncertainties which must enshroud the figures, while in the range of possible releases where results are very serious, and too serious to be allowed to occur, broader logic demands that improvements should continue to be made in the light of an assessment of what is worthwhile in terms of cost and advantage. Another point is that a number of aspects of design are not amenable to a statistical assessment of failure, and due regard must be paid to these, otherwise attention will be focused on matters which can more readily be treated statistically to the detriment of a balanced judgement.

CONCLUSIONS

The analysis of a number of existing and prospective sites for nuclear power reactors in terms of potential population risk reveals that there is no unique classification of the sites in order of merit. Orders of merit will vary depending on whether the criterion is the risk to the population as a whole or only that part resident in the most densely populated sector. If the exposure of individuals living close to the site is considered then a further merit order would be obtained. Some sites can be distinguished that are low in the merit order on all counts and clearly a special case would have to be made to justify their use in the immediate future. Moreover, there are too few practical sites high in the merit order on all three counts to form an adequate basis for the substantial development programme which is envisaged for nuclear energy in the UK.

An examination of the practical sites demonstrates that siting alone is an inadequate means of providing proper safeguards for the public. Therefore greater emphasis must be placed on the safety aspects of the design, construction and operation of a nuclear plant. Criteria proposed for doing this have been mentioned in this paper and are further discussed elsewhere in the Symposium. At present no conclusions have been reached although work is in hand to define standards which will meet the immediately foreseeable siting requirements.

Adams and Stone of the Central Electricity Generating Board (IAEA, 1967) proposed that the parameter determining acceptable siting be taken as individual risk and presented arguments why a constant, incremental, annual risk of 10^{-5} would be acceptable. In particular, they calculated an average decrement in life of only 3 to 6 days from such a risk (although, of course, a few would suffer much larger decrements and almost all others none).

Adams and Stone arrived at a siting policy which requires an exclusion area; a controlled area, where development that would prevent emergency action would not be allowed; and then an area of unrestricted population.

This paper drew considerable comment. For example, Ilari suggested that "community risk" must also be considered. Adams countered in two way. First, he argued it would be difficult to tell thousands of people they were being exposed to a risk that would not be permitted, if they were a much large group. Perhaps more importantly, he argued that if a criterion is based on the total potential number of casualties, the uncertainty in that number due to conditions and magnitude of the radioactive release is far greater than the difference that the choice of any site could make.

Adams did concede that quantitative risk assessment would not be possible when construction of a plant began, and would be subject to large uncertainties in any event.

F. R. Farmer presented a much-to-be quoted paper, "Siting Criteria - A New Approach," wherein he proposed that a probabilistic approach be employed in reactor safety assessment, and wherein he proposed a risk acceptance criterion (or limit line) in which the acceptable frequency of occurrence of an accident fell off as the consequences increased (with a rate such that the expected contribution to risk (frequency times consequences) was less for very large accidents than for smaller ones (a negative slope of -1.5 on a log-log plot). The famous Farmer limit line plot is reprodced below.

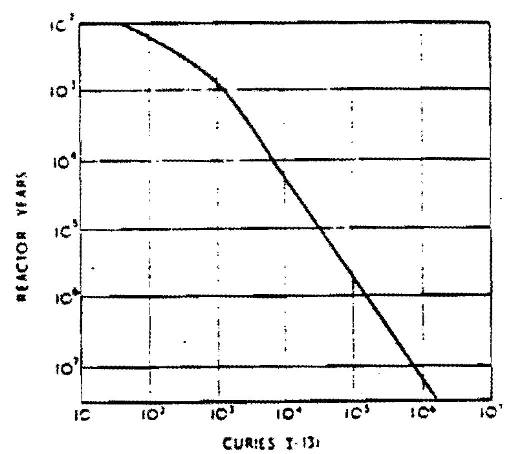


FIG. 11. Proposed release criterion

Farmer proposed this line as dividing acceptable and non-acceptable events, and suggested that only a relatively few events would be near the line for a reactor; these would lead to the principal contribution to risk, for which he suggested a criterion of less than 0.01 premature deaths per reactor year.

In later papers by British workers (Beattie, Bell and Edwards, 1969), the Farmer line was usually given a slope of unity; and risk assessments were made with an arbitrary assumption that one event would lie in each decade of frequency or consequences.

Additional insight into Farmer's thinking is available from the following summary by an ACRS staff engineer of some remarks he made to the Risk Assessment Review Panel (a group established by the USNRC, in November, 1977, and chaired by Professor H. Lewis, to review aspects of WASH-1400).

PRESENTATION BY MR. F.R. FARMER TO THE RISK ASSESSMENT REVIEW GROUP. NOVEMBER 10, 1977.

Mr. Farmer, former director of the Safety and Reliability Directorate, UKAEA, made a short but very interesting and informative presentation to the Risk Assessment Review Group. He emphasized several times that his remarks were personal viewpoints and were not necessarily the official position of the UKAEA.

Initially, in the UK it was hoped that absolute safety could be achieved by insuring that a system could tolerate the worst accident that could be devised, or by insuring that any transient proceeding toward a dangerous state would be self correcting. Unfortunately, those ideas were too idealistic since new accidents were always found which could defeat the safety design.

When it was realized that a maximum credible accident could not be usefully defined and that a 100% safe plant could not be designed, the use of simple criteria was investigated. Eventually, these simple criteria, such as ability to withstand two coincident faults, were superceded by requirements for redundancy and diversity.

During the early 1960's an ambitious attempt was made to develop codes of practice. However, Mr. Farmer feels that this period was too early to standardize, and the standards as developed were not useful during development stages of nuclear plants.

By the mid 1960's, general procedure was for the principal (groups) involved to make a joint conclusion as to the safety of new designs. At this time Mr. Farmer became concerned that these decisions were based too strongly on a case-by-case judgment. He felt that he, at least, needed some yardstick or target on which to base his opinion on safety of proposals brought to him.

As a result the following proposal was made, "that within the foreseeable program of gas cooled reactors, the chance of an accidental release of fission products causing one death to the public should be less than one." Mr. Farmer indicated that this target objective was set years ago (1967) and it will take several years to determine if it can be met.

The foreseeable program was assumed to be about 1000 reactor years (30 reactors operating for 30 years). It was further assumed that the release of fission products which might cause one casualty on a typical reactor site was about 10^4 curies of ^{131}I. A target objective that the release of 10^3 curies of ^{131}I should have a probability of occurrence of less than 10^{-3} per reactor year was set. It was further proposed that an accident having ten times the consequences (10^4 Ci ^{131}I) should have a lower probability of occurrence by at least a factor of ten. This proposal was first published at an IAEA conference in 1967. Although the proposal relates consequences in terms of ^{131}I, Mr. Farmer said that the basis is equally applicable to chemical plants. It was recognized at the time (and is still true) that there was insufficient data available, and the chemical industry was enlisted to obtain data.

Mr. Farmer went on to say that he thinks a probability of occurrence of 10^{-4} (of an accident leading to serious casualties) is the borderline of acceptability; he didn't appear to have much confidence that 10^{-4} is being met (universally). He stated that chemical plants are not within that value now, but efforts to improve are ongoing.

2.15 SOME SPEECHES 1966 -1969

An except from the trade journal Nuclear Industry, April, 1966, reports a meeting which gives a rather clear picture of the then current AEC Regulatory Staff position on the siting of LWR's in big cities.

AEC RESTATES ITS POSITION: NO CITY SITES FOR BIG UNITS

As an educational venture, the Symposium on Locating Nuclear Power Plants in Cities", held March 22 by the metropolitan New York section of the American Nuclear Society, was an unquestioned success.

A maximum of 60 persons was expected; 122 actually attended, including Norman Cousins, editor of the Saturday Review, who has been appointed by New York's Mayor John V. Lindsay to head his Task Force on Air Pollution.

For all,the sophisticated as well as the uninitiated, the meeting was significant for it served as the occasion for what was perhaps the clearest statement of AEC policy on urban siting that anyone has made so far. At the same time, it gave industry representatives a chance which they did not neglect to state their case.

The spokesman for the AEC's regulatory staff was Clifford K. Beck, deputy director of regulation. What he had to say left little question about the AEC's basic position on the building of big power reactors in urban centers. Barring a drastic change of attitudes and policies, which does not seem to be in the cards, there is little chance that the AEC will soon permit any utility to build a big nuclear plant in a big city.

Essentially, Beck said, what is needed is experience in designing, building and operating big plants, a very high degree of assurance that adequate engineered safeguards can be designed, and equal assurance that all crucial systems, including engineered safeguards, will perform with great reliability. Deciding when these conditions have been satisfied will be "a matter of judgment," but "up to the present and as of now, taking into account the status of all these matters, it appears that 'adequate experience' has not as yet been accumulated."

Beck identified three well-recognized trends in nuclear power planning and construction:

- The growing size of plants and the tendency to go to ever-higher burnups, which means that in-reactor fission product inventories are large.

- The tendency to standardize design, which he found hopeful. He stressed, however, that so far this is "a paper trend only," for none of the big 'standardized' plants has been operated.

• The increasingly strong incentives to build nuclear plants close to major load centers. The effects of this tendency, he said, "could be to add a disproportionately large increase in the potential hazard to people."

Elaborating, he stated a point of view which few in industry accept, at least in the sense that the AEC seems to have accepted it:

"If a given reactor, presently operating, should be moved closer to people than it now is with all other factors remaining unchanged, there would be an increase, first in the risk per individual...and, second, in the risk to the population as a whole.... If the reactor should be increased in size at the same time, the increment in risk would be still larger. Thus, just to maintain the level or risk to individuals and to the population where it now is, a movement of reactors closer to population centers would require improvements in the safety status of the facility."

Listing the improvements that will be necessary, he mentioned:

• The need to set reactor design, construction and operating standards "at the high quality level required." The problem, he said, is that "for many systems and components, the quality standards are better defined and more clearly satisfactory than for others."

• The need to resolve "any residual technical areas of uncertainty." Under this heading he mentioned metal-water reactions, and "certain problems" concerning the "design, codes, construction practices and testing of steel pressure vessels." Specifically, he said, "the relationship between technology available and actual practice in construction of vessels, the factors affecting rate of defect growth...and feasible methods for periodic inspection or otherwise verifying continued acceptability of the vessel are among the problems requiring further clarification.

• The need to establish the adequacy of accident-prevention and consequences-limiting safeguards. Here he gave three criteria: each system must be capable of performing as it is designed to perform at any time under any conditions; each must be proven highly reliable, and appropriate techniques must be developed for testing each system after it has been installed. Meeting these conditions would, he noted, sometimes call for "imagination and clever design and engineering."

• The need for sufficient experience with large reactors "to assure a high level of confidence." Noting that while "experience with the specific type of reactors" proposed for urban siting is highly desirable, he maintained that "if such experience consists of steady, uneventful operation... this would have quite limited usefulness in indicating the effectiveness of protective safeguard systems." Hence, the need for experimental work and, beyond that, for "observation and periodic in-situ testing of safeguards systems. Such a program, carried out over a period of time on a full-size, full power prototype would give valuable indications of the reliability and readiness of such systems."

Thus, by implication, Beck told his audience that the approval of metropolitan sites for big reactors is years off. Moreover, he made it clear that he was not talking only of the biggest plants in the largest cities:

"Before large power reactors are moved into areas where substantial increases in magnitude of potential hazard would result, and as a means of assuring that the present low levels of risks to individuals and the overall population at least do not increase, these matters must be established at the *highest possible level of confidence* [italics added].

Speaking for the Consolidated Edison Co., W. Donham Crawford stressed that the utility plans to continue to press for the approval of city sites, not only because it is convinced that they will be safe but because it must. Describing Con Edison's problem, he pointed out that nuclear power offers the answer to air pollution problems--an issue of increasing sensitivity in New York, and that usually it is not practical to build plants, nuclear or conventional, outside of the city because of resistance to constructing overhead transmission lines and the high cost of installing underground lines.

Crawford told his audience that before committing itself to build its second nuclear plant at its Indian Point site, Con Edison had discussed the possibility of a site in the city with the AEC and had concluded that "approval for a metropolitan site to permit 1969 operation was quite unlikely."

Looking ahead to the ultimate solution, Crawford foresaw "that a large part of the answer may be simply the passage of time, during which the public can observe the successful operation of more remote nuclear plants and the coming and going of nuclear powered vessels in and out of busy harbors."

From what Beck said, it will take considerably more than this to satisfy the AEC.

Beck's basic line of reasoning was indirectly challenged by J. C. Rengel, general manager of Westinghouse's Atomic Power Division. Repeatedly, he stressed that reactor manufacturers must build reliable, safe plants at any sites and he asserted that there has been too much emphasis on engineered safeguards designed to limit the consequences of an accident. As he put it, to the manufacturer "they are only last-resort systems." Making the point another way, he stressed that "basically, safety is no less important in a remotely located plant than in the city," and he devoted most of his talk to describing procedures for designing and building plants that can be counted on to operate safely and reliably anywhere.

Acknowledging that an urban site does involve some "special considerations," he outlined safety criteria Westinghouse has adopted for plants designed to be built in cities, and he left no doubt that the company believes it can meet those criteria now.

After listening to Beck, Crawford and Rengel, and to S. A. Szawlewicz, who reviewed the AEC's nuclear safety R&D program, Cousins said he was convinced that all that it is humanly possible to do to make nuclear power plants safe is being done. "But," he asked, "is all that is humanly possible enough?" He concluded that on the siting question the positions of the AEC's regulatory staff, as expressed by Beck, and the city's Task Force on Air Pollution were essentially the same.

Also in 1966, Congressman Aspinall included some remarks on reactor safety in an address to the American Nuclear Society in Denver, Colorado. These are reproduced below:

Need for Continued Safety Emphasis in Nuclear Power Industry

At the same time, however, I do feel that a few words of caution would not be inappropriate. Lest the successes that nuclear power is now enjoying should turn to euphoria, I would echo some remarks delivered a fortnight ago by my colleague Chet Holifield, Chairman of the Joint Committee on Atomic Energy. In speaking before the Edison Electric Institute he reminded the utility companies that the large, so-called "conventional" reactors being sold today involve extrapolations from operational reactors that are three to five times smaller in size, and that these larger reactors are new and in fact pioneering efforts. It is therefore necessary that we ramain ever vigilant of the safety aspects of nuclear plants. If we should relax our efforts to maintain the nuclear industry's remarkable safety record and a significant nuclear incident were to result, the consequences to the public--not to mention the industry itself--could be most unfortunate.

I therefore say, "Make haste slowly." If the Atomic Energy Commission's regulatory program is marked by "unparalleled conservatism," as some pundits have observed, then I say: So be it. Another commentator has remarked, perhaps with some exaggeration, that the safety features required by the Commission to be built into a reactor are somewhat akin to a man wearing three belts and two pairs of suspenders. My retort to this comment is simply this: Caution is the parent of safety.

> Safety consciousness is and should remain the number one criterion in the atomic energy field. Therefore, I want the AEC to continue to rigorously apply its high safety standards to the construction and operation of nuclear reactors.
>
> I want the Advisory Committee on Reactor Safeguards to continue to insist that multiple engineered safeguards be built into each reactor.
>
> And I want the reactor manufacturers, utility companies, and others in the nuclear industry to continue to take as seriously as they have their obligation to guard the safety of the public,. If they do-- and I can assure you no stone will be left unturned by the Joint Committee to see that its will will be done in this regard-- I think we can look forward to a thriving atomic power program whose record for safety will not only ensure its public acceptance but serve as a model for other industries.

On the other hand, Congressman Craig Hosmer, who for many years was the ranking Republican member of the Joint Committee on Atomic Energy, made three speeches in 1968-69 which included a different type of reference to the ACRS.

At the Conference on "Nuclear Fuel-Exploration to Power Reactors" in Oklahoma City, May 23, 1968, Representative Hosmer spoke on "The Nuclear Industry Today: The Way I See It". There follows a short excerpt from this speech:

> This distortion of nuclear plant prices-- first low, then high -- obscures a key element in assessing the true economics of nuclear power generation. Its present high side of the price scale threatens to discourage some future nuclear plant sales. It even may kill off some plants already announced. Specifically, I have in mind the Bolsa Island Nuclear Power and Desalting project in my own California Congressional district. Planning costs have skyrocketed from $444 million in 1965 to $765 million today. Some of that cost increase responds to unrelated circumstances, but a large portion comes from a boost in installed nuclear capacity costs from $132 per kilowatt to $250 per kilowatt, and a jump in the cost of power delivered at the load center from 6.2 to 7.9 mills per kilowatt hour. A 'go or no go' decision must be made by project partners before June 30th, and I am very pessimistic about it -- very pessimistic indeed."
>
> <u>"Practical Value" and the ACRS</u>
>
> It is things like this that lead me to believe that we can forget for a long time about the AEC making any "practical value"

determination on Light Water Reactors. Possibly, the Joint Committee ought to fulfill the commission's not-too-secret wish that its responsibilities for such a finding be eliminated.

And if the Joint Committee moves in that direction, it might also take a good hard look at the proposition of eliminating the Advisory Committee on Reactor Safeguards. AEC already has asked relief from the mandatory ACRS review of every license application. That could provide a good opportunity to evaluate the organization's future.

ACRS has no responsibility for the economics of the nuclear business and apparently could care less. The reactor manufacturers are afraid to approach it with many new safety improvements, particularly in the area of an integrated, systems approach to safety. With good reason, they are afraid ACRS will act as an Advisory Committee on Reactor Redundancy, and simply order them to add the new features to the existing safeguards, thus increasing costs further. I cannot help but wonder if ACRS had outlived its usefulness -- if it now serves less as a protective boon than it does as an anachronistic burden."

At the Joint Winter Meeting of the American Nuclear Society and Atomic Industrial Forum, Congressman Hosmer gave a speech which included the following comments:

Licensing and Regulation

Now, at just about this point in one of these intramural exercises I usually strike out with forked tongue at the 'deplorable regulatory mess,' whatever that is. I'll refrain tonight simply because in the past it has proved largely fruitless to do so. Not entirely, however. Six months ago I called for burning the Advisory Committee on Reactor Safeguards at the stake. Since then it has moved slightly off cloud #9 and we have had a couple of interesting off-the-record chats. So tonight I'm going to recommend that we temporarily forego the burning, but instead of letting ACRS itself pick and choose what it wants to create fear and trepidation about, that in the future the Commission designate to it the matters which AEC believes important enough for specific ACRS review. After all, ACRS is an advisory set up. So it seems appropriate that the Commission tell it what it wants to be advised about instead of ACRS telling AEC what advice it is going to get.

Frankly, I am concerned that if ACRS cannot somehow speed up doing its homework at the operating license stage, this country may be in for some serious brownouts. Some 72,000 megawatts of nuclear electricity is scheduled to come on the line in the period 1971-1973. It is required for the utilities' basic load growth. If ACRS cannot do its job with considerable dispatch as to the operating licenses involved, the delay will run us

> into drastic power shortages, and severe public censure of the whole idea of nuclear kilowatts would be inevitable.

Finally, Representative Hosmer addressed the Nuclear Safety Program Information Meeting at ORNL, February 17, 1969. Several excerpts from this speech follow:

> Reactor safety is a deep concern and a serious obligation to all of us here tonight. It is only one element of our overall nuclear safety program, but it is by far the largest in terms of its potential impact on the public at large. I imagine that collectively you devote more of your working hours to this one topic than to all other areas of reactor design and operation put together. And that is how it should be.
>
> Safety always has been -- and always will be--the Number One item on the nuclear power priority list. Without this concern on the part of the reactor manufacturers, the operators, the Atomic Energy Commission and the Joint Committee, there probably wouldn't be a nuclear power program today. And without a nuclear power program today, there might be many dark cities in 10 or 20 years.
>
> The Fact & Fiction of Reactor Safety
>
> Because of this before-the-fact emphasis on reactor safety, the nuclear power industry has largely solved the public acceptance dilemma it faced just a few short years ago. We're still not technologically ready for 1000 megawatt nuclear plants in uptown Manhattan--or even in Queens--but I believe that most people today are confident that the peaceful uses of atomic energy are being developed with their best interests in mind, and that their personal health and safety are being skillfully protected by highly trained and competent individuals.
>
> Achieving this high level of public confidence has not been easy. There are, of course, still some critics who like to get their names in the paper by making muckraking charges or fabricating half-truths about careless atoms and the alleged dangers of atomic anything. We're obliquely accused of plotting the grossest kinds of sins-spewing death into the atmosphere, fouling rivers and streams irradiating mothers' milk, poisoning babies and so on. Lately, these kinds of things have been spiced and laced with emotion packed allegations regarding massive fish-kills and other ecological retrogressions, despoiling the esthetics of landscapes and even architectural immorality in the design of power stations. To hear these charges made, you would think that the entire atomic energy program is some sort of sinister plot against mankind in general and the Sierra Club's shibboleths in particular. But for the life of me, I can't think what possible motive the United States government would have for the callous disregard for human welfare of which we are accused. I still believe our objective is simply to get

kilowatts to the people, safely and reliably, in adequate amounts and at the lowest possible cost.

All this is reminiscent of the AC/DC furor in New York City back in the 1880's. The city was in the process of adopting a revolutionary form of electric power called alternating current. The fear-mongers, ignorant of the facts and deaf to the explanations of the experts, shrieked that the whole concept was a death trap and that thousands would be killed. Fortunately, George Westinghouse persevered with his plan, and showed up his critics for what they were.

Some of our critics today like to point with alarm to certain of the technical problems we have had in some of our first reactors, but they constantly confuse reliability with safety. They overlook the fact that these were demonstration plants whose economics and operational characteristics were open to question from the outset. That's why we built them. I'll be among the first to admit that the reliability of the first-generation nuclear power plants has been nothing to write home about. But we have every confidence that additional experience in design and quality control during fabrication will improve that record.

Now, when a reactor shuts down safety during an emergency or malfunction, this is evidence that the reactor has been properly designed to safely take care of such eventualities. The reason for the shutdown may indicate some sort of design deficiency relating to reliability for normal operation, but the matter of ultimate safety can hardly be questioned as long as the shutdown procedure is effective. From my vantagepoint, the safety record of the civilian nuclear power industry still looks good because no member of the public has ever been injured because of the operation of a civilian nuclear power plant.

This is a record in which we all can take pride, but it is a special tribute to those who make reactor safety a profession. Here, I'm thinking about the reactor designers and operators, and the men on the regulatory side--those in Harold Price's operation at the AEC, the licensing boards and even the beloved Advisory Committee on Reactor Safeguards. They have done a remarkable job under trying circumstances, for which all Americans can be grateful.

A Review of Reactor Red-Tape

Now, despite all this intramural back-slapping, and if, indeed, the safety record of the civilian nuclear power industry is so good, why, then, is there so much pressure for changing the regulatory groundrules? The answer often given is that while present reactor licensing procedures have proved satisfactory in the past for a low volume of licensing business, there is reason for concern that they may not prove adequate for the future as we move toward annual licensing of large numbers of nuclear power stations, or at least larger size stations.

The handwriting is on the wall. Already in a couple of instances, utilities seem to have backed away from nuclear plants and gone back to fossil fuel in order to avoid the delay and uncertainty involved in getting a plant on the line. Much of this delay has no connection to the licensing process; but much of it does. Further, the utilities obviously can see such obstacles as esthetics, scenic preservation and anti-trust problems on the nuclear road. Logically, these are as pertinent to conventional plants as nuclear ones, but, in fact, the existing machinery of bureaucracy is more efficient in raising them in the nuclear case.

Things like this bother me, just as they bother other members of the Joint Committee. The development of the peaceful uses of atomic energy has been a national objective for almost two decades now. If we are going to sustain the level of progress we have achieved in the past few years, and if we are going to make sure that America's electric power supply keeps up with her demand, we are going to have to make sure that our power industry does not strangle on red tape. We have too much at stake as a nation to do otherwise.

Reactor Safety and Public Policy

Another apparent reason for all the discussion about reactor licensing is philosophical. Safety, after all, is relative. It is not an absolute. There is no such thing as absolute safety any more than there is such a thing as partial pregnancy. Right now, there is an infinitesimal chance that the roof will fall on our heads. Now, whenever you deal with something like this, you are going to wind up with different estimates of the degree of safety to be required--the old question of how safe is safe enough?

To most of you in this audience, reactor safety is a technical problem, and you'll discuss it in those terms for the next two days. But it also is a public policy question--a political problem, to use a more familiar term. Traditionally, it has been up to the Atomic Energy Commission and the Congress--specifically the Joint Committee--to answer the "how safe is safe" question. Put another way, it is a determination of what level of public risk is acceptable in order to achieve the benefits of cheap nuclear electricity in large loads.

This has been done. A balance was struck which fully protected the health and safety of the public, but still permitted the orderly, economic development of atomic power as a national objective. However, today many people rightfully or wrongly feel that the balance is becoming lopsided, not by deliberate reassessment, but simply by the sheer inertia of regulatory bureaucracy.

By way of extreme contrast, the automobile industry developed almost totally without formal decision of the degree of risk acceptable to society. We have, in fact, acceeded to a price of some 50,000 highway deaths annually for the convenience of the automobile. A few efforts are being made by the government at various levels to reduce this price, but it is in no way comparable to the constant, comprehensive attention given by the government to nuclear safety.

How Safe is Safe Enough

I do not suggest that we relax by one iota the necessary safety requirements which have been added to plant designs or otherwise in recent years. But what I am suggesting is that we recognize in practice that everything done in the interest of safety doesn't necessarily make things safe. We must consider the whole system, and avoid rushing every new safety device into production just because someone calls it a safety device. Our objective is added safety, and I believe that is best achieved by a systems approach based on a realistic estimate of the possibility of an accident, rather than carelessly adding new control and protection devices on top of existing safeguards.

It also is necessary to recognize that the licensing and regulation functions are not aimed at safety alone. Safety perhaps is only incidental. The actual focus is on economics.

Before you accuse me of a careless disregard for human life, let us see just what our traditional answer in the United States has been to the question of "how safe is safe enough?"

Studies recently made by Dr. Chauncey Starr, Dean of Engineering at UCLA, show that in the design and operation of conventional --coal, gas and oil-fired--electric power stations, we have come to accept a public risk limit of about one death per year per? one million installed kilowatts of generating capacity.

In the nuclear power area, technical studies of statistical but unlikely catastrophes which would spread radioactivity in populated areas, indicate that about 10 lethal cancers per million population might result.

On this basis, if the conventional power plant risk limit were applied to nuclear plants, each 1,000 megawatts statistically could be allowed one such accident every 10 years.

However, such a catastrophe would completely destroy the nuclear site of the plant and require complete dismantling and years of costly reconstruction. Because power companies expect nuclear generating stations to last 30 years, the economic consequences of such a catastrophe every 10 years would be totally unacceptable. In fact, even one such failure either on a statistical or actual basis during the nuclear plant life would be unacceptable.

Therefore, to meet economic performance requirements of the power companies the catastrophe rate of less than one in over 100 plant years would be needed, or 1/10 the rate currently associated with conventional plants. All this simply means that nuclear plants are 100 times as safe as conventional plants, because we design and build them ten times more conservative than that.

The Sources of the Problem

Now, it is apparent that no one is particularly enamoured with the way the reactor licensing process is working today. We all recognize that things are changing--the rapid acceptance of nuclear power, bigger sizes, new designs, changes in old designs, new companies, and so forth. These changes obviously raise new questions requiring new answers, which may range from very conservative to very liberal. The Commission recognizes this as much as either the utilities or the manufacturers. Evidence of this is the AEC's internal review of its licensing mechanism which is currently underway. The report will be completed sometime in the Spring and I am sure that it will address itself concisely to some of the problems at hand.

Recently, I conducted a review among some of the knowledgeable people in this field. I wanted their views on the problems in reactor licensing and what they thought we might do about them. And if I could summarize their comments, I would say they are concerned primarily about the lead-time, in the name of safety and licensing, that it takes to receive a construction permit. But that is by no means the only problem.

They are also very concerned about the continual changing of regulatory requirements. Maybe this is necessary in a rapidly changing field, but every application seems to bring new requirements, and more often than not, the requirements seem to change during the processing of a particular application. This, they contend, makes it impossible to file a satisfactory construction permit application, and the long lead-times make orderly load-growth planning unreasonably difficult.

Additionally, many of those polled--again, rightly or wrongly--complained about what they regard as overly strict requirements for design detail at the construction permit stage of the licensing process. This results, they say, in a premature freeze on design, and a trade-off of progress for safety objectives which they believe may be ill-defined or even non-existent.

Three Proposals for Improving the System

In this polling effort to find out how others view the problems, I also asked for and received suggestions for improving the reactor licensing process--some good, others not so good.

There were three suggestions most frequently made. One is to revamp the functions of the ACRS. The second is a proposal that the manufacturers be licensed as well as the utilities; what has become known as the Good Housekeeping Seal concept. And the third, of course, is that the Regulatory Staff should be spun off and organized as an independent government agency. I would like to treat each of these subjects briefly and give you some idea of my thinking.

First, the ACRS. Half in jest, I suggested last summer during a speech out in Oklahoma City that the ACRS be abolished and got a standing ovation. I certainly don't expect that kind of a reaction here tonight. I didn't even expect it then. My purpose simply was to get some discussion going about ACRS roles and missions, and it seems to have succeeded.

There is no question in my mind that these men perform a valuable service to the nation by providing some independent expertise on reactor safety. The questions I have are two. One is whether or not ACRS members worry more about keeping their personal records clean against all contingencies by striving for absolute safety rather than the high relative safety society and progress require. My second concern is whether the functions we have presently assigned to the committee are the proper ones.

As you are probably aware, a bill to relieve ACRS of its responsibility for reviewing each and every construction permit application was introduced in Congress last year at AEC's request, although no action was taken. I am hopeful that when the Joint Committee gets around to taking up this matter, we might broaden our vision and consider some alternatives.

One particular frequent suggestion has been to re-establish ACRS as a general advisory committee on reactor safety. I have a feeling that such an approach would be more consistent with the original objectives when ACRS was established. The idea here, obviously, is to remove ACRS from the regulatory mainstream and let it devote its time to the major safety questions affecting all reactors. The nuts-and-bolts review of specific applications would then be left to the Regulatory Staff with the continued safeguard of a public hearing. It seems to me that the important question is whether today's reactors would be any less safe without a specific ACRS review. I'm not prepared to answer that tonight. I only suggest that we take a good look at it.

2.16 MIDLAND

On October 30, 1968, the Consumers Power Co. submitted a Preliminary Safety Analysis Report for two PWR's, each having a power rating of 2552 MWt to be located on the southern boundary of Midland, Michigan. According to the report to the ACRS from the Regulatory Staff dated January 23, 1969,

"The applicant does not now propose to include any iodine removal system (i.e., containment reagent spray or charcoal filters) which would help to decrese offsite doses from accidents. Furthermore, a containment vessel leakage rate of 0.2% per day is proposed."

"The applicant proposed to include the Dow Chemical Company complex (in which a maximum of 10,175 people work during the day shift) within the exclusion area. In addition, it proposes to include the City of Midland (population of 27,779) within the low population zone."

The Regulatory Staff report states that the population at the Midland site exceeds that of Zion and Indian Point to distances of about 5.5 miles. Beyond that distance, Midland is more typical of a site in an agricultural community.

The Staff report estimated that thyroid doses on the site and in Midland, using the assumptions of Part 100, would exceed the guideline limit of 300 rem, and that rapid evacuation and/or dose reduction features will be required.

An early review of the acceptability of the site was requested.

The Midland Subcommittee met on February 4, 1969. A large portion of the minutes of the Executive Session are reproduced below to provide insight.

<u>Summary</u>

This meeting was held with the Midland applicant and the Regulatory Staff to discuss population related questions in an attempt to evaluate the proposed site. Conclusions of the meeting are as follows:

1. Post-accident doses, when evaluated by the Ergen-Monson method, are no worse than for the reference site.

2. All Regulatory people and consultants appear to disagree with the Applicant's meteorological assumptions.

3. Evacuation plans for the site and Midland require further discussion.

4. Addition of iodine removing safety features and reducing the containment leak rate will reduce the post-accident doses.

5. The applicant is willing to have further discourse on the subject of adding or improving the plant safety features.

6. The Exclusion Area and Low Population Zone size can probably be reduced considerably by adding safety features.

7. The cooling pond for heat dissipation and steam from reactor steam generators used in the Dow processes represent new features.

8. The Midland project should be reviewed at the February 1969 ACRS meeting.

Executive Session

Dr. Monson opened the meeting by stating that Dr. Thompson is on board to give continuity of thinking with regard to siting policies. The purpose of the day's review is to determine acceptability of the Midland population distribution. Mr. Price is to give DRL conclusions orally at full ACRS as has been more and more his current philosophy.

The site is different from most for the following reasons: The exclusion area radius is 1170 m. and contains 353 people and the Low Population Zone radius is 3 miles and contains 28,000-48,000 people. The 28,000 number is the permanent population plus 1/4 of the temporary population, and the second number includes all temporary people. There was some question about the validity of the 48,000 number. The Population Center Distance is about 700 feet.

According to Dr. Monson, all definitions of exclusion area, etc., can be altered to meet the intent and requirements of 10 CFR 100. This can be done by reducing these distances. At Indian Point, for example, the exclusion area and LPZ radii are 0.22 and 0.67 miles and the Population Center Distance is 0.63 miles. This is with a 3025 MWt reactor. Zion has an LPZ of about 1 mile.

If Midland would improve the containment leak rate and iodine scavenging, they might get around the legal problems. So, the question becomes; does this site meet the intent of 10 CFR 100? This city boundary itself is not too meaningful here since the city includes the entire Dow plant.

Mr. Squires asked the following question; if the Dow plant is not considered in determining PCD, what happens? The PCD becomes approximately 1 1/2 miles in this case and, in Squires' opinion, the Committee may want to decide whether or not to include this population. The words in 10 CFR 100 refer to city boundary but the intent is something different.

Population tables were handed out (Attachment 1) which showed comparisons with Indian Point, Zion, Trap Rock, and Burlington. The "maximum" numbers on page 1 include people which may be counted twice, both as a resident and working at the plant. However, there are people who work at the plant who live outside the 3-mile radius. A concentration of approximately 4000 people is located in the corporate offices in the NW part of the Dow plant. Looking at the (P + 1/4 T)* column in Attachment 1, Midland looks acceptable. In the "maximum" column,

Midland

0 - 1 mile - No apparent problem
0 - 2 miles - Less than Zion, no apparent problem
0 - 3 miles - A little worse than other sites
0 - 4 miles - A little worse than other sites
0 - 5 miles - Same as accepted sites

So, even if one includes 100% of people who might be there, the site compares favorably in terms of total population.

Discussion of Worst 22½° Sector table, page 2, Attachment 1. Looking at "maximum" column,

Midland is

0 - 1 mile - Better than Trap Rock
0 - 2 miles - Worse than any previously approved sites
0 - 3 miles - Similar to Indian Point
0 - 4 miles - Similar to Indian Point

Dr. Gifford (a former ACRS member) asked what kind of population criteria are being considered in view of the amount of population data generated. This led to a discussion of page 3 of Attachment 1; comparisons of deviations from the reference site. The bases for these were reviewed for the benefit of Dr. Thompson. The Midland index looks better than the reference site with regard to injuries as shown but not as favorable as the reference site on fatalities. Using the "maximum" numbers, Dr. Monson's conclusion is that ther is no question that this site is acceptable when talking about the large accident. Six

*Permanent population plus 1/4 the temporary population.

accidents have been calculated for the worst 22½° sector and are listed on page 3 of Attachment 1. In this, Midland appears to be better than the reference site. So, it seems to Dr. Monson, looking at this population data, that the site is acceptable even with the maximum people possible present.

Dr. Monson then read excerpts from ACRS reports on:

Reactor Siting Criteria - 12/13/61 - 10 CFR 100 should be a guide, and TID 14844 and 10 CRF 100 should be flexible and changeable. One purpose of the guides is to limit exposures to large populations.

Letter of 12/13/60 - refers to using reactors of proven design. This letter refers to limiting the number of people killed to less than a catastrophic number. The reactor should be sited such that if the incredible accident occurs with no safeguards, people in a city > 15-25,000 people should not receive > 300 Rem.

October 22, 1960 - Another reference is included here on catastrophic numbers of fatalities. There is also a discussion of exclusion area, LPZ, and PCD. PCD doese would be limited to 200 Rem with the incredible accident. This concept has already been violated in some approved sites.

With these examples, it is clear that reactor operating experience was lacking, that the incredible type accident was assumed credible and, if it did happen, the number of people killed would be minimized. Dr. Monson observed that population matters are more important with the larger reactors and, if nothing else is done, the reactor must be moved farther away. With adequate improvements, the larger reactor might be moved closer.

Dr. Zabel asked the old hands if adequate operating experience had been gained since 1960-61. Dr. Thompson felt that the experience is running behind the rate of size increase. The amount of operating time is not significant and, at the same time, no serious accidents have occurred. But, the absence of bad experience may be considered good experience, per Dr. O'Kelly. Dr. Monson noted that the people far from the site become more important with larger reactors and the bad accidents.

Dr. Gifford felt that some discussion of safety improvements of the last few years would be useful. Dr. Thompson stated that there was considerable emphasis on PV's at the expense of other things such as post-accident cooling in the 1960 era. There have been some changes in philosophy such as Section III which are hard to evaluate from the safety viewpoint. According to Dr. Thompson, depending on which side you look at, you can paint a dark or light picture. Dr. Monson felt that there is not an abundance of experience to justify significant changes in criteria now; e.g., the recent Swiss reactor meltdown of the past few weeks is a bit of bad experience.

Dr. Gifford brought up the interrelation of plant design aspects with the population aspects. This plant design has certain changes from previous designs; e.g., steam system. Dr. Monson observed that there is a danger of ruling out site reviews if the plant design must be reviewed, too.

Dr. Thompson's comments on 10 CFR 100 pointed out that there were two considerations: the close-in population which iodine was the main concern and the far-out people for whom man-rem was the consideration. People then were looking at sites farther away from cities so that very large populations wouldn't be badly affected. The basic intent of 10CFR 100 was to limit doses in the event of severe accidents.

At the 106th meeting, February 6-8, 1969, the ACRS completed its preliminary site review of the application. The minutes of the meeting, which are reproduced below, succinctly summarize the Committee's action.

Specific Projects - Midland Plant

The Committee completed its preliminary site review of the application by the Consumers Power Company for authorization to construct the Midland Plant. The Committee concluded and reported orally to the applicant that: "The Committee has reviewed the proposed Midland site primarily from the standpoint of population and population density. The following remarks therefore refelct Committee conclusions based primarily on that one aspect of site evaluation.

"The Committee considers the site proposed to be unacceptable for use with reactor plants designed and analyzed as presently described in the PSAR. However, it believes that the site may be acceptable for use with reactor plants of the proposed power rating if: (1) The facility is equipped with adequate engineered safety features and protective systems; (2) the facility is analyzed sufficiently conservatively - particularly in respect to: determination of exclusion area and low population zone; assurance of low potential doses at short distances from the reactor in the unlikely event of a serious accident; evaluation of the number and location of people who could be safely and quickly evacuated in such an event; and, use of assumptions, for example those related to meterology, in dose calculations; (3) the facility is designed, constructed, and utilized sufficiently conservatively; and (4) the facility is provided with thoroughly structured, effective emergency plans, including evacuation plans."

Significant factors in the Committee's considerations were the high population within four miles of the site, the minimal engineered safeguards proposed in the application, and the use of less than conservative assumptions in the dose calculations.

ACRS review of the construction permit application for Midland continued in 1970. This was the first Babcock and Wilcox plant in a "populated" site, and the review was conducted in considerably greater depth and breadth than usual, involving several Subcommittee and full Committee meetings. Resolution of several generic safety items was required or the provision of flexibility to accommodate future requirements was sought. The Committee asked that the Regulatory Staff review detailed criteria for the protection and emergency power systems prior to installation of these sytems, rather than rely on a commitment to meet the appropriate general design criteria, etc.

In a sense, the very considerable length of the ACRS report on Midland, which is reproduced on the following pages, was an indication that the site was "a more populated one."

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

June 18, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON MIDLAND PLANT UNITS 1 & 2

Dear Dr. Seaborg:

During its 122nd meeting, June 11-13, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by the Consumers Power Company for a permit to construct the Midland Plant Units 1 and 2. During this review, the project also was considered at Subcommittee meetings held on January 22, 1969, at the plant site, on April 24, 1970, at Chicago, Illinois, on February 4, 1969, March 24, 1970, and June 10, 1970, at Washington, D. C. and at the ACRS meetings of February 6, 1969, April 9, and May 8, 1970, in Washington, D. C. In the course of these meetings, the Committee had the benefit of discussions with representatives and consultants of the Consumers Power Company, Babcock and Wilcox Company, Bechtel Corporation, Dow Chemical Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Midland Plant site is on the south bank of the Tittabawassee River adjacent to the southern city limits of Midland, Michigan. The main industrial complex of the Dow Chemical Company lies within the city limits directly across the river from the site and provides an area of controlled access about two miles wide between the reactor site and the Midland business and residential districts. The exclusion area of the plant site has a radius of 0.31 miles and includes a small segment of the Dow plant; no Dow employees are permanently assigned in this segment, and the applicant has the right to remove any persons from this segment if conditions warrant. The low population zone has a radius of 1.0 miles and contains 38 permanent residents and about 2,000 industrial workers, mainly employees of Dow Chemical Company. The number of permanent residents within five miles of the plant site was estimated to be 41,000 in 1968, mainly in the city of Midland and its environs.

The applicant has established criteria for, and has begun the formulation of a comprehensive emergency evacuation plan. This plan is being coordinated with the well-established plan of the Dow Chemical Company for emergency evacuation of the Midland chemical plant and portions of the City of Midland in case of major emergencies at the chemical plant. Close coordination with appropriate municipal and state authorities is also being established.

The Midland units will each include a two-loop pressurized water reactor designed for initial core power levels up to 2452 MWt. The nuclear steam supply systems and the emergency core cooling systems of these units are essentially identical with those for the previously reviewed Oconee Units 1, 2 and 3 and Rancho Seco Unit 1 (ACRS reports of July 11, 1967 and July 19, 1968, respectively). The combined electrical output of the two units will be 1300 MW. In addition, 4,050,000 lbs per hour of secondary steam will be exported to the adjacent Dow plant to supply thermal energy for chemical processing operations.

The prestressed, post-tensioned concrete reactor containment buildings are similar to those approved for the Oconee Units 1, 2 and 3. The design will include penetrations, which can be pressurized, and isolation valve seal water systems to reduce leakage. Channels will be welded over the seam welds of the containment liner plates to permit leak testing of the seam welds.

Cooling water for the Midland reactors is supplied from a diked pond with a capacity of 12,600 acre-feet. Make-up water is taken from the Tittabawassee River. The cooling water supply is sufficient for 100 days of full power operation without make-up during periods of low river flow. In the unlikely event of a gross leak through the dikes of the cooling pond, a supplemental source of water will be available. The supplemental source is provided within the main pond by excavating a 24 acre area to a depth of six feet below the bottom of the main pond. This source can supply shut-down cooling capability for 30 days without make-up.

The applicant will conduct an on-site meteorological monitoring program to verify the applicability of the meteorological models used for accident evaluation and routine release limits as well as to determine any meteorological effect of the cooling pond. This program should be completed during construction.

Midland is the first dual purpose reactor plant to be licensed for construction. The export steam originates from the secondary side of the steam generators and may contain traces of radioactive leakage from the primary system. The demineralized condensate from 60 to 75 percent of the export steam is returned by Dow to the feed water supply of the reactor plant. The condensate from the remaining steam is either chemically contaminated or cannot practically be returned to the nuclear plant. It is collected in the Dow waste treatment system for dilution and processing with other streams before eventual discharge to the river. Thus, the unreturned portion of the condensate represents an effluent from the reactor plant to which the requirements of 10 CFR Part 20 must apply.

This matter may be considered in two parts: (1) the steps taken by the applicant to ensure that any radioactivity in the export steam is within the limits set by 10 CFR Part 20 and as low as practicable and (2) the measures taken by the Dow Chemical Company to ensure that the export steam can be used in chemical operations without product contamination and that the unreturned steam condensate is properly managed for safe disposal. In connection with item (1), the applicant proposes to monitor and control radioactivity in the export steam. A representative, continuous sample of the export steam will be condensed for monitoring and laboratory analysis. The gamma activity of this flowing sample will be continuously monitored by on-line analyzers and an alarm actuated if the activity exceeds an appropriate limiting value. The alarm will serve to indicate any change in the integrity of the steam generators or fuel cladding. Samples of this condensate stream will be analyzed at appropriate intervals by sensitive low-level beta counting for determination of gross beta activity and concentration of selected radionuclides. The applicant agrees to limit, by maintaining high integrity of the steam generators and fuel cladding, the yearly average gross beta activity in the export steam to one-tenth or less of the limits specified by 10 CFR Part 20 for the selected radionuclides. The yearly average will include any periods of short duration when the concentrations may approach but not exceed the 10 CFR Part 20 limits. The applicant states that in his judgement it is practical to operate the plant within these limits. If these limits are exceeded, corrective measures will be taken in the plant or the delivery of export steam to Dow will be terminated. He also agrees to demonstrate the analytical equipment and procedures in development programs to be carried forward and completed during construction of the Midland Plant. In connection with item (2), Dow has stated that they will apply for a 10 CFR Part 30 Materials License to receive, possess, and use the export (secondary) steam as a source of thermal and mechanical energy. No export steam or condensate will be intentionally introduced into any product. Isolation of the export steam from contact with products will be accomplished by the use of heat exchange devices which will provide suitable physical barriers. Programs will be established to provide for detection of leaks in the heat exchange devices by analyses, monitors, and other means; for repair of leaks when detected; and for appropriate administrative control of the programs.

Dow has stated that accumulation of radioactivity from the export steam and release of radioactive materials in the effluent will be in accordance with 10 CFR Part 20. The unreturned condensate will represent less than 10% of the total liquid effluent disposed of through the Dow waste treatment plant and the annual average concentration in the total effluent is expected to be less than 1% of the 10 CFR Part 20 limits.

The Committee believes that the criteria proposed by the applicant and Dow for the control of radioactivity in the export steam are necessary and adequate. The detailed procedures for implementation should be developed during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

To minimize the likelihood of subsidence at the site, the applicant and Dow have agreed to prohibit future salt mining operations within one-half mile from the center of the reactor plant. No new wells will be drilled within this distance and all existing wells will be abandoned and plugged. The Committee believes these arrangements are satisfactory.

A large volume of liquid chlorine is maintained in a refrigerated storage vessel about one mile from the Midland plant control room. The applicant is continuing his study of the consequences of a major accidental release of chlorine from this vessel. He has included in his criteria for the design of the control room the objective of finding a practical method of maintaining the concentration of chlorine in the control room atmosphere below the eight hour threshold limiting value (TLV) of 1 ppm for the most serious conceivable chlorine accident. The Committee believes that adequate air purification facilities should be provided in the control room ventilation system to reduce chlorine concentration to the eight hour TLV of 1 ppm so that operators can work without respiratory equipment during an extended chlorine emergency. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff.

The reactor vessel cavity will be designed to withstand mechanical forces and pressure transients comparable to those considered in the design of the Zion and Indian Point-3 plants.

The applicant has stated that he will provide additional evidence obtained by improved multi-node analytical techniques to assure that the emergency core cooling system is capable of limiting core temperatures to the limits established at present. He will also make appropriate plant changes if the further analysis demonstrates that such changes are required. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The safety injection system for the Midland plant is actuated by either low reactor pressure or high containment pressure signals. However, of these two, the reactor is tripped only by the low reactor pressure signal. The Committee believes that provision also should be made to trip the reactor by the high containment pressure signal.

The applicant plans to develop more detailed criteria for the installation of protection and emergency power systems together with appropriate procedures to maintain the physical and electrical independence of the redundant portions of these systems. The Committee believes that these criteria and procedures should be reviewed and approved by the Staff prior to actual installation.

The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly because of flow starvation, gross enrichment error, or from other causes to be remote. However, the resulting effects in terms of local high temperature or pressure and possible initiation of failure in adjacent fuel elements are not well known. Appropriate studies should be made to show that such an incident will not lead to unacceptable conditions.

The Committee believes that consideration should be given to the utilization of instrumentation for prompt detection of gross failure of a fuel element.

The Committee has commented in previous reports on the development of systems to control the buildup of hydrogen in the containment which might follow in the unlikely event of a major accident. The applicant proposes to make use of a technique of purging through filters after a suitable time delay subsequent to the accident. However, the Committee recommends that the primary protection in this regard should utilize a hydrogen control method which keeps the hydrogen concentration within safe limits by means other than purging. The capability for purging should also be provided. The hydrogen control system and provisions for containment atmosphere mixing and sampling should have redundancy and instrumentation suitable for an engineered safety feature. The Committee wishes to be kept informed of the resolution of this matter.

The Committee recommends that the applicant accelerate the study of means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant stated that the engineering design would maintain flexibility with regard to relief capacity of the primary system and to a diverse means of reducing reactivity. This matter should be resolved in a manner satisfactory to the Regulatory Staff during construction. The Committee wishes to be kept informed.

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to the Midland Plant Units 1 & 2.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, the nuclear units proposed for the Midland Plant can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References

1) Amendments 1 - 12 to License Application

2.17 NEWBOLD ISLAND

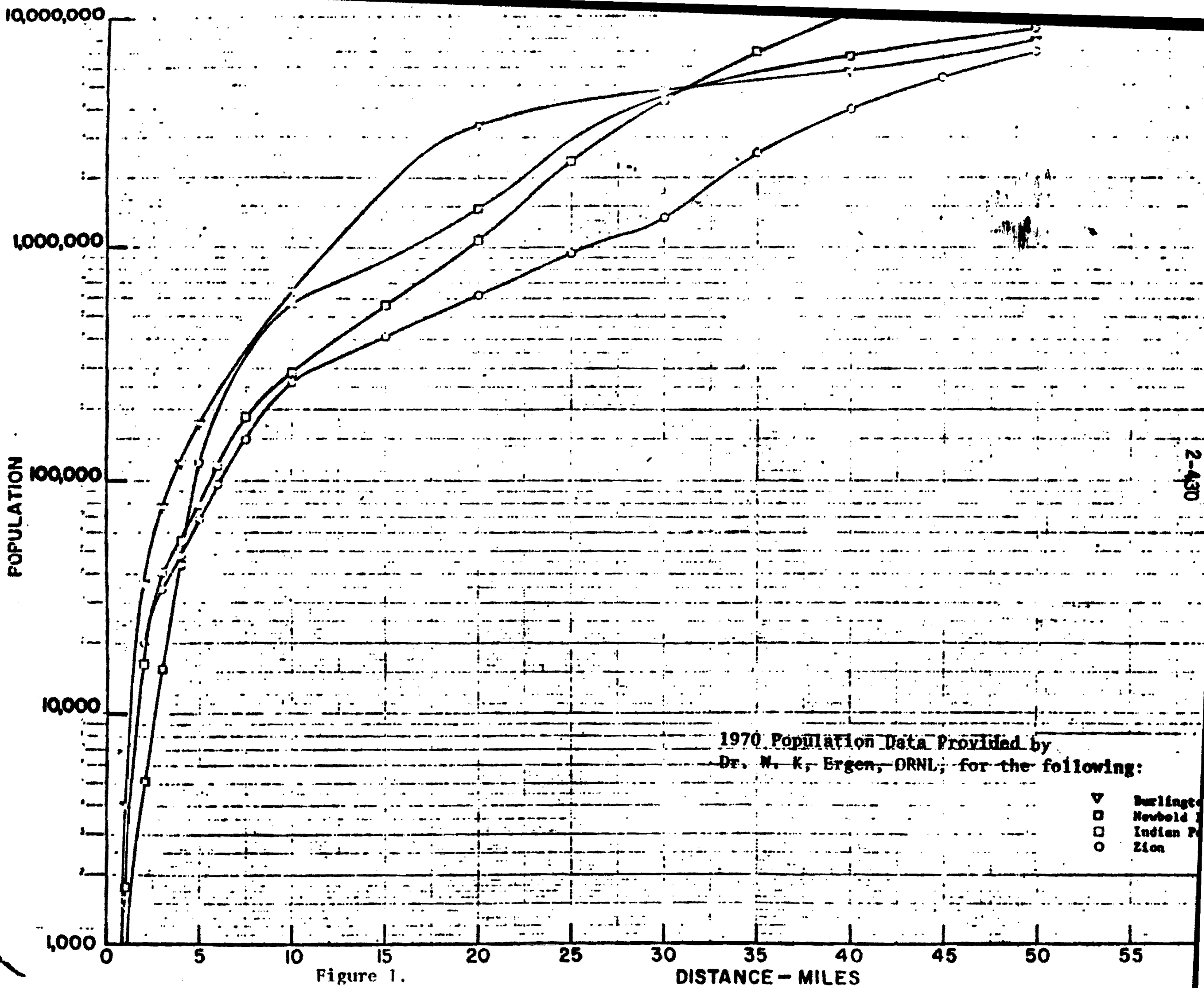
By letter dated May 6, 1969, the Public Service Electric and Gas Company of New Jersey requested an informal review of their proposed Newbold Island site by the Regulatory Staff and the ACRS. The proposed Newbold Island site consisted of 530 acres located on Newbold Island on the Delaware River in Mansfield Township, New Jersey. It was 4 1/2 miles south of Trenton, New Jersey and 11 miles northeast of Philadelphia, Pa. Cumulative population within the first 4 miles of Newbold Island was significantly lower than the cumulative population of Zion, Indian Point, or Burlington. At 4 miles the effect of Levittown and Trenton became important, and the projected cumulative population about Newbold Island exceeded that at Indian Point and Zion beyond 4 and 3/4 miles. The cumulative population about Newbold Island significantly exceeded that of Indian Point or Zion from 5 to 33 miles. Beyond 34 miles, the projected cumulative population for Indian Point exceeded that of Newbold Island.

The ACRS Newbold Island Subcommittee met on July 1, 1969. The population distribution was compared with that of Indian Point, Zion, Trap Rock and Burlington using the rather simple, arbitrary Site Population Index (SPI) described above. The minutes of the July 1 meeting show that the Subcommittee members concluded that Newbold Island had a similar SPI index to Trap Rock, which the Committee had decided was "not unacceptable" at its 100th meeting in August 1968. One of the Subcommittee members was Dr. Hanauer, who dissented from the draft ACRS letter of July 1969 on "Location of Power Reactors at Sites of Population Density Greater than Indian Point - Zion", which reinforces the idea that the Subcommittee felt Newbold Island was similar.

In a written report to the ACRS dated July 22, 1969, the Regulatory Staff gave no conclusions concerning the acceptability of the Newbold Island site. At the 112th meeting, August 7-9, 1969 and the 113th meeting, September 4-6, 1969 the ACRS considered the Newbold Island site proposal.

As shown in the ACRS report of September 10, 1969, which is reproduced as follows, the Committee decided the site was not unacceptable. However, the ACRS called attention to several matters as requiring attention, including measures to cope with pressure vessel leaks and rupture, as practical. In effect, many of the comments in the Newbold Island list were similar to those previously included in the draft letter of July 1969 on the use of sites worse than Indian Point - Zion (which was never sent (see Section 2.12). The ACRS consciously withheld approval of the proposed Newbold Island containment at the site review stage.

For purposes of information, the population distributions for Burlington, Indian Point, Zion and Newbold Island are compared in Figures 1 and 2.



POPULATION
10,000,000
1,000,000
100,000
10,000
1,000
0
5
10
15
20
25
30
35
40
45
50
55
DISTANCE – MILES
1970 Population Data Provided by
Dr. M. K. Ergen, ORNL, for the following:
Burlingt
Newbold
Indian P
Zion

Figure 1.

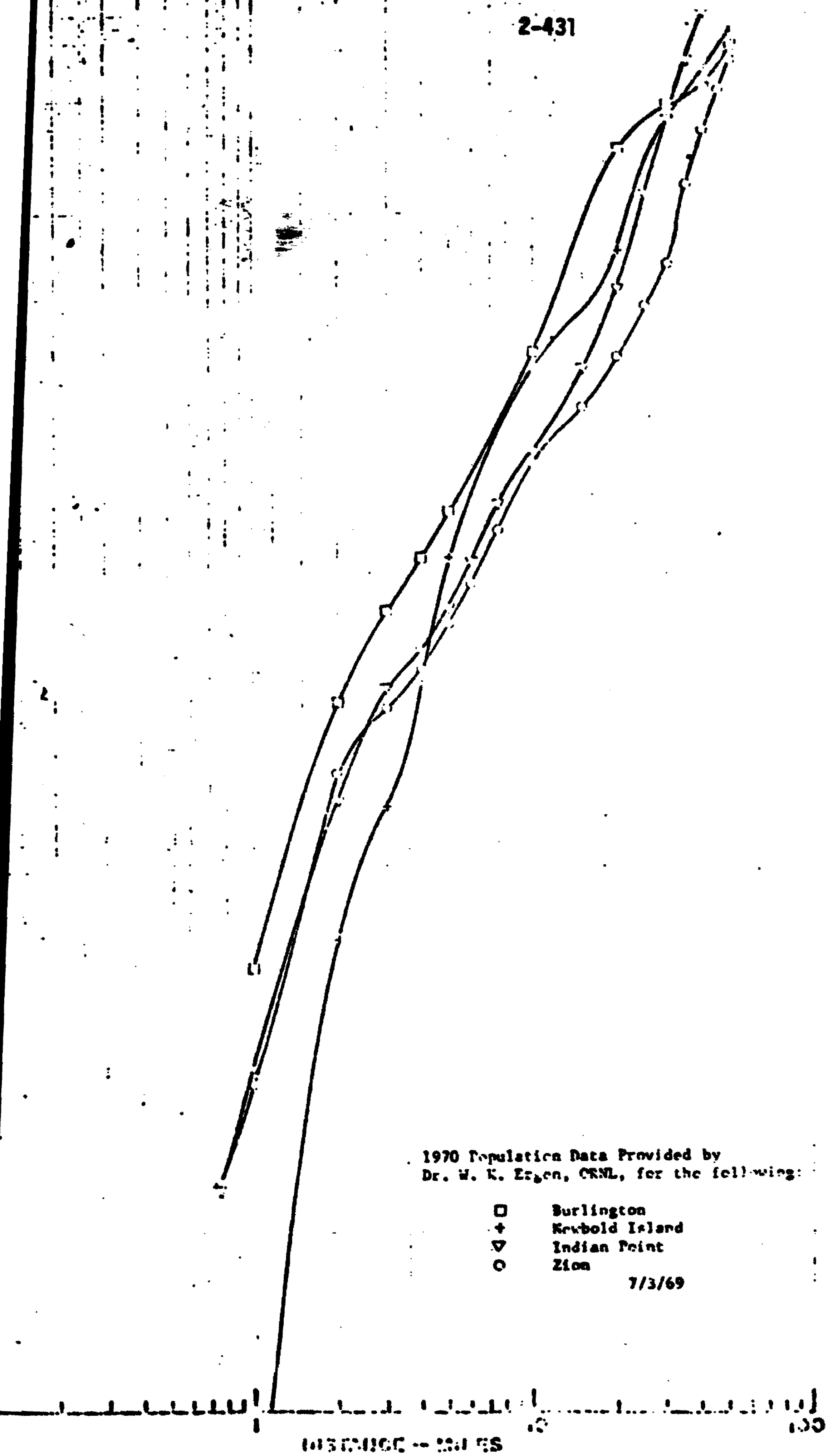

1970 Population Data Provided by
Dr. W. K. Ergen, ORNL, for the following:
□ Burlington
+ Newbold Island
▽ Indian Point
○ Zion
7/3/69
1
100

Figure 2.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 10, 1969

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: PUBLIC SERVICE ELECTRIC AND GAS COMPANY - NEWBOLD ISLAND SITE

Dear Dr. Seaborg:

At its 112th meeting, August 7-9, 1969, and its 113th meeting, September 4-6, 1969, the Advisory Committee on Reactor Safeguards considered the Newbold Island site, which the Public Service Electric and Gas Company proposes as the location for a nuclear power plant including two boiling water reactors of approximately 3400 MW(t) each. The site consists of approximately 500 acres located on Newbold Island in the Delaware River. A relatively high population density is associated with this site; it is 4-1/2 miles south of Trenton, New Jersey (1960 population - 114,000) and 11 miles northeast of Philadelphia, Pennsylvania (1960 population - 2,000,000). The nearest population center is a grouping of suburbs in Bucks County, Pennsylvania, known collectively as Levittown (1960 population - 70,000), with its nearest boundary 3.4 miles from the site. An ACRS subcommittee visited the site on July 1, 1969. During its review, the Committee had the benefit of discussions with representatives of Public Service Electric and Gas Company and their consultants, and the AEC Regulatory Staff, and of the documents listed below.

Preliminary studies of the geology, seismology, hydrology, and meteorology of the site have been made and have revealed no significant problems. Natural draft cooling towers will be used in the plant.

The conventional dry-well and suppression-chamber containment system will be enclosed in a low-leakage reactor building with air recirculation and filtration to reduce further the releases of radioactivity in the unlikely event of an accident. The Committee believes that the proposed containment system is a useful approach, but cannot comment at this time on its adequacy.

Special attention will be required with regard to the integrity of any portions of the primary system outside the containment and to the steamline isolation valves. Appropriate additional means for coping with possible valve leakage or a loss of integrity outside the containment should be provided.

Public Service Electric and Gas Company described procedures involving additional holdup of off-gas releases during routine plant operation. The Committee believes that special attention should be given to the control of liquid waste releases and to the prevention of radwaste accidents, as additional means of keeping radiological releases at a very low level.

The Committee believes that, for this site, additional study of the problems related to possible degradation of reactor vessel integrity, such as leaks and vessel wall ruptures, is needed. Measures that will ameliorate these problems should be implemented to the extent that they are practical and significant to public safety. The features provided should be of such design as to prevent their interference with other engineered safety features.

Other matters noted in previous ACRS letters pertaining to large water reactors should receive appropriately greater attention in the design of the plant. The Committee believes a more conservative approach is appropriate in the design of a plant at this site, with regard to the margins in the engineered safety systems, protection against possible internally-generated missiles, and the number of items to be resolved after the construction permit review.

The Committee emphasizes again the vital importance of quality assurance, and the necessity for adequate consideration of diverse and independent means of protection against common failure modes in safety systems.

The conclusion reached by the Committee regarding this site has been influenced in part by its expectation that some satisfactory experience will have been obtained with reactors of this general type by the time a construction permit is issued, and some satisfactory experience will have been obtained with reactors of this type having the same power and power density as those proposed for this site by the time an operating license is issued.

The Advisory Committee on Reactor Safeguards believes that, subject to the above comments, the Newbold Island site is not unacceptable with respect to the health and safety of the public, for a plant having the general characteristics described above and designed with due attention to the other matters discussed.

Sincerely yours,

Stephen H. Hanauer
Chairman

References attached.

On February 27, 1970 Public Service Electric and Gas Company filed a formal application for the licenses required for the contruction and operation of the proposed Newbold Island Nuclear Generating Station. The applicant proposed certain additional features for these plants compared to those being incorporated in similar boiling water reactors in much less populated sites. These included an additional main steam line stop valve; enclosure of the main steam line and isolation valves within a tunnel chamber which is evacuated to the reactor building; charcoal adsorbers in the gaseous radwaste system for holdup of noble gases; and a system to reduce turbine seal releases of radioactivity. The proposed secondary containment was a low-leakage building (10% per day), held at negative pressure to provide controlled and filtered venting and having a 2 psig design pressure capability. These additional features were aimed at reducing releases from routine operation or related to the existing design basis accidents for BWR's.

On May 15, 1970, the AEC Regulatory Staff issued a preliminary report to the ACRS which indicated no major problems with Newbold Island.

The ACRS began its review of Newbold Island very early, rather than waiting for the Staff to have essentially completed its own review. A first Subcommittee meeting was held June 3, 1970, and discussion was initiated on the topics listed in the September 10, 1969 site letter, plus others. The meeting minutes note that General Electric was very reluctant to discuss capability to withstand postulated pressure vessel ruptures of various sizes, including rupture of a large nozzle, but agreed to discuss these matters under pressure by the Subcommittee. This topic was discussed at considerable length in ensuing Subcommittee and full Committee meetings. General Electric concluded that the vessel internals and the ECCS were such that a break area up to four square feet anywhere in the vessel could be withstood and the core cooled acceptably, and that the concrete shield wall around the vessel could be strengthened so that it would maintain its basic integrity and not generate unacceptably large missiles.

A matter that was forced to a partial decision on Newbold Island was the issue of anticipated transients without scram (ATWS). General Electric had previously reported in generic studies and during the review of other projects that an automatic pump trip would resolve short term considerations in ATWS (leaving long-term cooling still to be reviewed) but had never taken positive steps to include the feature. The Regulatory Staff had taken no position on what was required or acceptable, if anything. The ACRS advised General Electric that, for Newbold Island, a commitment on the pump trip (or some other method for dealing with ATWS) would be required prior to completion of the construction permit review. The Committee took a similar position with regard to the need for resolution of hydrogen control following a LOCA.

There was much discussion of the adequacy of the ECCS and the potential for increasing its functional capability or reliability. General Electric maintained that the system function was well understood and that the proposed ECCS was adequate (General Electric said it had an 800°F margin for the design basis pipe break) and could not be significantly improved from a functional or reliability viewpoint. The Regulatory Staff officially remained aloof from such discussions. At a Subcommittee meeting on April 26, 1971, they indicated they were divided concerning the desirability of improved ECCS for the Newbold Island site. The Staff favored improvements in consequence-limiting devices as they arose from the Part 100 approach.

There was also considerable discussion on the relative advantages of the proposed secondary containment versus the higher pressure (10-15 psig) building in the preliminary proposal by Consolidated Edison for Trap Rock.

The memorandum of March 15, 1971, as follows, from ACRS member Okrent to Subcommittee Chairman Siess, illustrates some of the thinking which entered into the Newbold Island review.

Actually at about the same time that application was made for a construction permit for Newbold Island, Philadelphia Electric applied for a construction permit for a BWR at the Limerick site west of Philadelphia. This was also a highly populated site. According to the rough site population index used by the ACRS, Newbold Island was worse than the average Indian Point-Zion site, while Limerick was somwhat better. This index did not include meteorology.

However, the local meteorology for Limberick was worse, and the prevailing winds were generally easterly, that is towards Philadelphia from Limerick. Also, the population around Limerick exceeded that at Newbold Island for the first four miles. In the eyes of the Regulatory Staff this made the "Limerick site equivalent to Newbold Island, or worse." The Regulatory Staff stated that whatever requirements were to be made for Newbold Island should also apply to Limerick. However, in practice, they did not require strengthening of the shield wall around the reactor vessel, as was done at Newbold Island. What the Staff did consider were features to cope with the MCA or lesser accidents. They did not consider Class 9 accidents, or accidents not previously reviewed for BWR's at less populated sites.

One small sidelight of the review of the Newbold Island reactor was that the Regulatory Staff was asked by the ACRS to take a new independent look at the possible sources and possible effects of fire. However, the ACRS Subcommittee minutes indicate that the Staff did no additonal review in this regard, and the matter was not pursued. This was in 1971, roughly 4 years before the Browns Ferry fire. Also, the minutes of the April 26, 1971 Subcommittee meeting record that the Regulatory Staff thought that there would be no

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

March 15, 1971

C. P. Siess, Chairman
Newbold Island Subcommittee

RESPONSE TO YOUR MEMO OF FEBRUARY 8, 1971 REGARDING NEWBOLD ISLAND

Attached is a list of questions and suggestions regarding the Newbold Island review. I suggest that the Subcommittee explore these areas with the Staff and applicant in an appropriate manner.

J. E. Hard
for D. Okrent

Attachment

List of Questions and Suggestions Re: Newbold Island

cc: Other ACRS Members

Reactor Building Comparison - Inleakage Versus Higher Design Pressure

A study comparing the advantages and disadvangates of both types should be done and done well.

Reactor Vessel Integrity

The applicant has been reviewing the capability of his original design to withstand losses in reactor vessel integrity. This should be continued so that the capability of the entire system is studied on a more uniform basis as to what is the limit of capability and where weak spots first arise (the uses of ASME Section III allowable stresses for one function, 90% yield for a second, and ultimate for a third makes this difficult).

However, the Committee's siting letter was broader. "The Committee believes that, for this site, additional study of the problems related to possible degradation of reactor vessel integrity, such as leaks and vessel wall ruptures, is needed. Measures that will ameliorate these problems should be implemented to the extent that they are practical and significant to public safety."

These additional studies were not to be limited by the original design proposal and its capabilities. Hence, the applicant should go beyond his current studies and meet the intent of the letter in examining possible measures.

Engineered Safety System Margin

This is not limited to ECCS but includes all engineered safety systems such as containment, RHR, etc. The Committee should request examination of what additional margins in safety can be provided. This might be accomplished by improvements in design capability, in capacity, in redundancy and diversity, or by combinations of these steps.

Missiles

In line with the Committee's letter, a systematic re-evaluation of potential missiles should be made, with acceptable bases provided for excluding those which continue to be excluded.

Items to be Resolved During Construction

Which other old or new generic items remain to be resolved? Which will not be resolved prior to ACRS action on a construction permit? Why? How is fully satisfactory resolution to be assured, so that these items are not incomplete or inadequate at operating license review time?

Sabotage

What positive steps have been taken to review sabotage considerations in design?

Primary System Lines Outside Containment

Is there more that the Staff believes can be done with regard to quality - or to control of radioactivity in the event of rupture of these lines?

Common Mode Failures: Electrical and Mechanical

Is there more that the Staff thinks can be done to accomplish significant improvements in reliability?

Con Ed Memo of December 3, 1969 from Boyd to Morris

This listed many items which Regulatory Staff members thought warranted reconsideration in connection with a BWR at a site like Newbold Island. We should have the Staff's thinking on these matters. Among those of interest are Item Nos. 2, 39, 49, 56, 57, 68, 85, 86 and 91 in the referenced memo, but others should not be ignored.

Reliability of Containment Isolation

Has detailed consideration of required reliability been given by Staff to this function? Is it acceptable?

Seismic

Should we ask Philbrick, Page and Wilson for an independent review of seismic design factors?

Torus

Should we accept the invulnerability of the torus and its appendages? If so, have we examined forces, etc. adequately? Are the current margins adequate?

Reactor Shutdown

Should we examine possible advantages (and disadvantages) of a faster (high pressure) liquid poison system? How much assurance must we have concerning the new generic items on possible control rod binding?

D. Okrent
02/22/71

diagnostic instrumentation available without the AC vital bus. ACRS member Monson thought that the DC system should be such that there would be assurance of safe shutdown if all AC power was lost. (This was to become an active, generic safety concern several years later.)

At the June 3, 1971 Subcommittee meeting there was an inconclusive discussion of the proposed seismic design basis. ACRS consultant Philbrick had suggested that a much larger earthquake could occur, and ACRS consultant Wilson had estimated that the recurrence interval for the proposed safe shutdown earthquake (0.2g) was about 2500 years. The Regulatory Staff had no response concerning an acceptable recurrence interval. The Applicant stated he had no probabilistic basis in proposing the seismic design basis.

At the June 3, 1971 Subcommittee, there were also some inconclusive discussions about the reliability of various engineered safety systems. When asked how the Applicant knew that this plant had the necessary safety-related reliability, Mr. Schneider replied that it was based on experience and on making logical improvements. He also noted that if the AEC goes much further in requiring the expenditure of time and money for extremely conservative designs, it may stop the construciton of nuclear power plants.

In its report to the ACRS of June 30, 1971 the Regulatory Staff concluded that "the Newbold Island facility can be built and operated at the proposed location without undue risk to the health and safety of the public." At the 136th meeting, August 5-7, 1971, the ACRS completed its review and reported favorably concerning the proposed construction permit application. However, the recommendation was not unanimous; there were additional comments by three members of the ACRS, Messrs. Monson, Okrent and Palladino, to the effect that a construction permit should not be issued unless there were a major change in the containment design, and the high pressure coolant injection system (which was intended to deal with small breaks in the primary system) was made redundant.

The Newbold Island letter is reproduced on the following pages together with the ACRS report on Limerick, which was completed at the same meeting.

The ACRS again reported favorably on Newbold Island on July 17, 1973 following a re-review of the increased close-in population estimates.

After the favorable ACRS letter, the construction permit application for Newbold Island proceeded to a hearing before an Atomic Safety and Licensing Board (ASLB). The case had become a very controversial one, with some members of state government and some Congressmen from the region expressing concern or outright opposition.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

AUG 10 1971

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON NEWBOLD ISLAND NUCLEAR GENERATING STATION UNITS NOS. 1 AND 2

Dear Dr. Seaborg:

At its 136th meeting, August 5-7, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application by the Public Service Electric and Gas Company for a permit to construct the dual-unit Newbold Island Nuclear Generating Station. This project was also considered at the 130th, 133rd, 134th, and 135th meetings of the Committee on February 4-6, May 6-8, June 10-12, and July 8-10, 1971, respectively; and at Subcommittee meetings on June 3, 1970 at Argonne National Laboratory, and on February 3, March 29, April 26, June 3, July 7, and August 4, 1971 in Washington, D. C. During its review the Committee had the benefit of discussions with representatives and consultants of the applicant, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below. The Committee reported the results of its pre-application site review to you in a letter dated September 10, 1969.

The station will be located in New Jersey on 530-acre Newbold Island which is near the east bank of the Delaware River about 4-1/2 miles south of Trenton, New Jersey (1970 population - 105,000) and 11 miles northeast of Philadelphia, Pennsylvania (1970 population - 2,000,000). The nearest population center is a grouping of suburbs in Bucks County, Pennsylvania, known collectively as Levittown (1970 population - 72,000), with its nearest boundary 3.4 miles from the site. The applicant has specified a radius of one mile for the low population zone, which had in 1969 a transient population associated with industry of approximately 1200, and a small resident population which is expected to be about 100 by 1985. The minimum exclusion distance is 700 meters, which extends to the west bank of the Delaware River. As pointed out in the Committee's report of September 10, 1969, a relatively high population density is associated with this site.

Each unit includes a boiling water reactor to be operated at 3293 MWt. With respect to core design, power level, and other features of the nuclear steam supply system, the Newbold Island units are essentially duplicates of the Browns Ferry Units 1, 2 and 3, and Peach Bottom Units 2 and 3. Waste heat from the station will be rejected to the atmosphere by natural draft cooling towers.

In its report of September 10, 1969, the Committee listed several matters which it believed warranted special attention in the design of a plant for the Newbold Island site. In response to these recommendations, the applicant has included in the Newbold Island design several features, in addition to those normally provided for boiling water reactor units, to reduce still further the potential for release of radioactivity to the environment. The principal additional features are described below:

Reactor Building. For each unit, the conventional steel drywell and suppression chamber primary containment, the fuel handling area and spent fuel pool, and the principal components of the engineered safety features are contained in an unlined reinforced concrete building of cylindrical shape with a domed roof. This building is designed to Class I seismic standards and to resist the standard tornado, and missiles from this or other sources. The building can resist an internal pressure of 2 psig, and inleakage at a differential pressure of 1/4-inch of water will be limited to 10 percent of the building volume per day. A filtration, recirculation, and ventilation system (FRVS) is provided to recirculate and filter the reactor building atmosphere and maintain the building at a negative pressure relative to the outside environment.

Main Steam Lines. A low-leakage, slow-acting, stop valve has been added downstream of the two fast-acting valves in each main steam line, and a seal air system has been provided to further reduce leakage of radioactivity after main steam line isolation. The portion of the main steam lines containing the isolation valves is enclosed in a Seismic Class I tunnel chamber connected to the reactor building so that any out-leakage following the unlikely event of a design basis loss-of-coolant accident will be treated by the reactor building FRVS before release to the atmosphere. The entire length of the main steam lines up to and including the turbine stop valve will be designed to Class I seismic standards. The main steam lines from the third isolation valve to the turbine stop valve will be designed and fabricated in substantial accordance with the requirements for AEC quality assurance Classification Group B. In addition, selective inspection of critical areas of this piping will be performed during refueling outages.

Radioactive Waste Disposal. The radioactive waste disposal systems include several features beyond those normally provided in boiling water reactor plants. The liquid waste system permits the recycling of equipment and floor drain wastes and the evaporation of chemical and laundry wastes before discharge to the environment. The gaseous waste system provides for the recombining of hydrogen and oxygen, condensing the vapor, hold-up for decay of short-lived isotopes, and cryogenic separation of the noble gases. Krypton and xenon may be stored for periods sufficiently long that krypton-85 becomes the only significant remaining radioisotope. Provisions will be made to utilize non-radioactive steam in the turbine gland seals and to process containment purge gases when deinerting. The Committee believes that these waste management systems are capable of limiting releases of radioactivity to the environment to levels that are as low as practicable.

Reactor Vessel Integrity. The applicant has described improvements in the design and fabrication of the reactor vessel. These include redesign of the large nozzles to reduce stress concentrations; redesign of the bottom head to reduce the number of welds and improve the capability for in-service inspection; and improved procedures and standards for inspection during fabrication. The applicant has studied the problems related to possible degradation of reactor vessel integrity and has concluded that a nozzle failure or a small break would not impair the integrity of the biological shield, the primary containment, or the reactor internals, and would not affect the ability to cool the core. In addition, the biological shield has been redesigned to increase substantially its ability to withstand internal pressures, jet forces, or missiles.

Emergency Core Cooling System. The emergency core cooling system (ECCS) has been modified in two ways. The high-pressure coolant injection (HPCI) system has been changed to inject water directly to the core through the core spray sparger rather than into the downcomer region via the feedwater sparger. In addition, the applicant has stated that the steam-turbine driven HPCI pump will be modified to the extent feasible to increase the volume of water delivered to the core. The low-pressure coolant injection (LPCI) system has been changed to inject water inside the core shroud through four separate vessel penetrations, rather than through the recirculation lines. The applicant has stated that these changes provide increased reliability of these systems and reductions in the peak clad temperatures attained in the unlikely event of a loss-of-coolant accident.

The Committee believes that the design changes described above are suitably responsive to the concerns stated in its letter of September 10, 1969 regarding additional matters which should be considered for a plant at the Newbold Island site.

In the event of an unisolable break of an instrument line or a process line, reactor coolant will be discharged to the reactor building. Since the instrument lines will contain a 3/8-inch flow-restricting orifice inside the primary containment, failure of as many as eight such lines will not lead to pressures inside the reactor building greater than the 2 psig at which it relieves to the environment. However, failure of a process line, if not isolated in a very short time, could lead to pressures in excess of this relief pressure and significant amounts of reactor coolant would be discharged to the environment. Although the off-site doses from such an accident would be well within the 10 CFR Part 100 guidelines, they would be comparable to or greater than the doses calculated for other less probable accidents. The Committee believes, therefore, that the applicant should make design provisions for reducing the quantity of reactor coolant discharged to the reactor building in the event of a process line break.

The applicant has studied design features to make tolerable the consequences of failure to scram during anticipated transients, and has concluded that automatic tripping of the recirculation pumps and injection of boron could provide a suitable backup to the control rod system for this type of event. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for the Newbold Island reactors. However, further evaluation of the sufficiency of this approach and the specific means of implementing the proposed pump trip should be made. This matter should be resolved in a manner satisfactory to the AEC Regulatory Staff and the ACRS during construction of the plant.

The applicant has stated that a system will be provided to control the concentration of hydrogen in the primary containment that might follow in the unlikely event of a loss-of-coolant accident. The proposed system is not capable of coping with hydrogen generation rates in accordance with current AEC criteria unless the primary containment is inerted. Therefore, the Committee believes that the containment should be inerted and that the hydrogen control system should be designed to maintain the hydrogen concentration within acceptable limits using the assumptions listed in AEC Safety Guide 7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident."

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to the Newbold Island Station.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the Newbold Island Nuclear Generating Station Units Nos. 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Additional comments by Dr. H. O. Monson, Dr. D. Okrent and Dean N. J. Palladino are attached.

Sincerely yours,

Original Signed by
Spencer H. Bush

Spencer H. Bush
Chairman

References - Newbold Island Nuclear Generating Station Units Nos. 1 and 2

1. Public Service Electric and Gas Company letter dated February 27, 1970; License Application; Preliminary Safety Analysis Report (PSAR), Volumes 1 through 5

2. Amendments Nos. 1 through 5 and Nos. 7 through 9 to PSAR

ADDITIONAL COMMENTS BY DR. H. O. MONSON,
DR. D. OKRENT AND DEAN N. J. PALLADINO

Although the large, low pressure, high in-leakage secondary reactor building proposed by the applicant for Newbold Island Units 1 and 2 represents an improvement over reactor buildings currently employed for BWRs at sites with lower surrounding population densities, we believe that further improvement is appropriate. The relatively small volume of the steel pressure-suppression type primary containment introduces some crowding of equipment and some attendant problems in the simultaneous accomplishment of full protection against violation of primary containment by possible missiles, jet forces, and pipe whip, and accomplishment of full access for in-service inspection. Some further protection would be provided against extremely low-probability accidents involving a concurrent loss of primary system integrity and a limited violation of primary containment by the use of a large, relatively high-pressure (of the order of 10 psi, as has been proposed for a BWR at another site having a comparable surrounding population density), low-leakage, secondary containment building. Such a high-pressure, secondary containment, coupled with a pressure-suppression primary containment, provides a combination which can tolerate a fairly substantial violation of primary containment arising from the same event which caused a loss of coolant, as well as further protection against unforeseen events. We believe that this improvement in safety capability is warranted for a more densely populated site like Newbold Island, and recommend that the issuance of a construction permit be contingent on the use of a high-pressure, low-leakage secondary containment.

For postulated loss-of-coolant accidents involving small break sizes, the high-pressure coolant injection system (HPCI) arranged so as to inject into one of the core spray loops is predicted by the applicant to be highly effective in limiting peak clad temperatures to moderate levels. We believe that for a high power, high-power-density reactor at a site as densely populated as Newbold Island, the applicant should give further consideration to the use of an HPCI system on the second core spray loop. The purpose would be to provide redundancy of this means of protection in the event that the single HPCI system became ineffective because of failure of an HPCI component or because the accident arose from rupture of the core spray line into which the HPCI injects. The automatic depressurization system which together with the low-pressure emergency cooling systems constitutes an alternate means for coping with small breaks, albeit by introducing a larger opening, would continue to serve as a backup.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

August 10, 1971

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON LIMERICK GENERATING STATION UNITS 1 AND 2

Dear Dr. Seaborg:

At its 136th meeting, August 5-7, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application from the Philadelphia Electric Company for a permit to construct the two-unit Limerick Generating Station. The project was considered at Subcommittee meetings on November 10, 1970 at the plant site, and on March 31 and July 29, 1971, in Washington, D. C. During its review the Committee had the benefit of discussions with representatives and consultants of the applicant, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Limerick Station will be located in Pennsylvania on a 587-acre site on the Schuylkill River about midway between Philadelphia and Reading. The nearest population center is Pottstown (1960 population - 26,000; year 2000 predicted population - 55,000) with its nearest boundary 1.7 miles to the northwest. The low population zone radius is 1.3 miles. The estimated population in 1968 was 500 persons within one mile and 5,200 persons within two miles. The minimum exclusion distance is about 2,500 feet, which extends to the west bank of the Schuylkill River and includes a small uninhabited island owned by the State of Pennsylvania. The City of Philadelphia is 20.7 miles to the southeast with a 1970 census population of about 2,000,000.

Each unit of the Limerick Station includes identical boiling water reactors to be operated at a power level of 3293 MWt. The core designs, power densities, and other features of the nuclear steam supply systems are essentially identical to the Browns Ferry units of the Tennessee Valley Authority and Peach Bottom Units 2 and 3 currently under construction by the Philadelphia Electric Company. Waste heat is rejected to the atmosphere by two natural draft cooling towers. The normal cooling water requirement of 74 cfs, including 54 cfs for consumptive use, is supplied from the Schuylkill River. To provide another source during drought periods arrangements are being made to obtain water from the Delaware River.

The containment is of the over-under pressure suppression type similar to that of the Shoreham Nuclear Power Station. The drywell is a reinforced concrete, steel-lined truncated cone; the wetwell is a cylinder of similar construction. The drywell and wetwell are separated by a 3-1/2 foot thick reinforced concrete floor penetrated by 85 vent pipes. A low-leakage, Class I reactor building surrounds both units which share a single compartment above the level of the refueling floor and occupy separate compartments below this level. The building is designed to relieve through blow-out panels at an internal pressure of 7 inches of water, an arrangement which the applicant has stated serves to protect engineered safety equipment from excessive steam exposure while still maintaining offsite doses from postulated process steamline failures far below 10 CFR Part 100 guidelines.

The reactor building in-leakage at a differential pressure of 1/4 inch of water will be limited to 50% of the building volume per day. On isolation of the building a recirculation-filtration system starts automatically, continuously processing about 60,000 cfm through HEPA and charcoal filters. A small fraction of the discharge of this system is exhausted to the outside environment through the standby gas treatment system which includes deep-bed, charcoal filters.

The entire length of the main steam lines, up to and including the turbine stop valves, will be designed to Class I seismic standards. The main steam lines from the downstream isolation valve to the turbine stop valve will be designed and fabricated in substantial accordance with the requirements for AEC quality assurance Classification Group B. In addition, the Committee believes it appropriate to design and install all connected piping down to 2-1/2 inches in diameter to Class I seismic standards out to and including the first valve. The applicant has stated that he will install a third steam line isolation valve downstream of the two fast-acting valves or develop an equivalent water-seal system acceptable to the Regulatory Staff.

The biological shield is to be constructed of magnetite concrete placed between steel plates. The shield will be reinforced near openings to insure integrity for postulated ruptures in the vicinity of nozzles. The Committee believes that the entire biological shield should be designed to have reasonable ability to withstand internal pressure and jet forces.

The emergency core cooling system (ECCS) has been changed in several ways. The high pressure coolant injection (HPCI) system has been modified to inject water directly into the core through the spray sparger rather than into the downcomer region by the feedwater sparger. In addition, the applicant has stated that the turbine driven HPCI pump will also be modified to the extent feasible to increase the volume of water delivered to the core. The low pressure coolant injection (LPCI) system has been changed to inject water inside the core shroud through four vessel penetrations. Each of two pairs of LPCI pumps feed a header serving two nozzles. The applicant has stated that these changes provide increased reliability of these systems and reductions in the peak clad temperatures attained in the unlikely event of a loss-of-coolant accident.

The radioactive waste disposal systems include several features beyond those normally provided in boiling water reactor plants. The liquid waste system permits the recycling of equipment and floor drain wastes and the evaporation of chemical and laundry wastes before discharge to the environment. The gaseous waste system provides for the recombining of hydrogen and oxygen, condensing the vapor, hold-up for decay of short-lived isotopes, and cryogenic separation of the noble gases. Krypton and xenon may be stored for periods sufficiently long that krypton-85 becomes the only significant remaining radioisotope. Provisions will be made to utilize non-radioactive steam in the turbine gland seals and to process containment purge gases when deinerting. The Committee believes that these waste management systems are capable of limiting releases of radioactivity to the environment to levels that are as low as practicable.

The applicant has studied design features to make tolerable the consequences of failure to scram during anticipated transients, and has concluded that automatic tripping of the recirculation pumps and injection of boron could provide a suitable backup to the control rod system for this type of event. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for the Limerick reactors. However, further evaluation of the sufficiency of this approach and the specific means of implementing the proposed pump trip should be made. This matter should be resolved in a manner satisfactory to the Regulatory Staff and the ACRS during construction of the reactor.

The applicant has stated that a system will be provided to control the concentration of hydrogen in the primary containment that might follow in the unlikely event of a loss-of-coolant accident. The proposed system is not capable of coping with hydrogen generation rates in accordance with current AEC criteria unless the primary containment is inerted. Therefore, the Committee believes that the containment should be inerted and that the hydrogen control system should be designed to maintain the hydrogen concentration within acceptable limits using the assumptions listed in AEC Safety Guide 7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident."

The applicant has selected a value of 0.12 g for the acceleration representing the maximum ground motion at the site and on which Class I seismic design is to be based. The Committee recommends a minimum acceleration of 0.15 g be used for the design basis earthquake for this site.

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to the Limerick Station.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the Limerick Generating Station Units 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H Bush

Spencer H. Bush
Chairman

References

1. Philadelphia Electric Company Preliminary Safety Analysis Report (Volumes 1 through 5), for Limerick Generating Station Units 1 and 2

2. Amendments 1, 2, 3, 4, 6, 7, 9 & 10 to the License Application of Philadelphia Electric Company for the Limerick Generating Station Units 1 and 2

On October 3, 1973, Dr. Ralph E. Lapp, appearing before the ASLB on behalf of the Commonwealth of Pennsylvania, testified against issuance of a construction permit. His testimony is interesting in several ways. He stated his belief that the ACRS request for extra-conservatism in plant design, and its requirement for additional safeguards, is a reflection of the need for insuring a higher degreee of public protection because of the high population at risk at the site. He stated that, in his opinion, the ACRS letter of July 17, 1973 is a conditional approval of the station and that the Atomic Energy Commission should resolve this condition now before the contruction permit is approved. "When three highly qualified scientists on the ACRS express reservations about a reactor installation, I believe that their recommendation should be given the most serious consideration." Then he went on to note that the ACRS recognized an unusual populatin risk. He called attention to the fact that the Committee believed that plans for appropriate measures to extend several miles beyond the proposed LPZ radius of one mile were necessary. Dr. Lapp states "To the best of my knowledge this is the first time that ACRS has expressed a specific requirement for emergency plans to treat a Class 9 accident situation." Dr. Lapp concludes "I believe that the Board is faced with a problem that is greater than that of a single nuclear power site. If Newbold Island is approved, then the way is open for utilities to site future plants in locations of equal or higher population-at-risk situations."

Excerpts from Dr. Lapp's testimony follow.

Testimony of Dr. Ralph E. Lapp
In the Matter of the Newbold Island Nuclear Generating Station, Units 1 & 2
Docket Nos. 50-354, 50-355. Atomic Safety and Licensing Board, UASEC.
Trenton, New Jersey, October 3, 1973.

I appear here on behalf of the Commonwealth of Pennsylvania to testify as to my concern about the siting of twin nuclear power reactors at the proposed Newbold Island site.

The siting of high power nuclear reactors involves the most careful judgment as to the adequacy and reliability of engineered safeguards which are designed to prevent the release of dangerous quantities of radioactive material to the environment where it may pose a threat to the public health and safety. When the population at risk to such a potential radioactive release is as great as in the Newbold Island situation. I believe that the engineered safeguards should be subject to the most detailed public examination.

I am aware that the Applicant has submitted a detailed Preliminary Safety Analysis Report together with additional technical supplements in the public docket and that the Regulatory Staff has carried out a safety evaluation (Dec. 17, 1971) together with Supplement No. 1 (May 17, 1973) and Supplement No. 2 (August 30, 1973).

Both Applicant and the AEC Regulatory Staff have carried out analyses of various design basis accidents up through Class 8 on the AEC's accident scale which ranges from Class 1 for trivial events and up to Class 9 for accidents involving core meltdown.

The Advisory Committee on Reactor Safeguards (ACRS) has conducted its own safety evaluations of the proposed Newbold Island nuclear station and in its Sept. 10, 1969 letter to AEC Chairman Glenn T. Seaborg it defined the population in the Newbold Island area and stated:

> 'The Committee believes that, for this site, additional study of the problems related to possible degradation of reactor vessel integrity, such as leaks and vessel wall ruptures, is needed. Measures that will ameliorate these problems should be implemented to the extent that they are practical and significant to public safety. The features provided should be of such design as to prevent their interference with other engineered safety features.
>
> Other matters noted in previous ACRS letters pertaining to large water reactors should receive appropriately greater attention in the design of the plant. The Committee believes that a more conservative approach in the design of the plant at this site, with regard to the margins in the engineered safety systems, protection against possible internally-generated missiles, and the number of items to be resolved after the construction permit review.'

I believe that the extra conservatism in plant design and in a requirement for additional safeguards is a reflection of the need for insuring a higher degree of public protection because of the high population at risk at this site. Some of these safeguards were detailed in the ACRS letter to Chairman Seaborg dated Aug. 10, 1971.

These special features included:

1. A low-leakage secondary containment (reactor building) capable of withstanding 2 psig and equipped with a filtration, recirculation and ventilation system (FRVS).
2. Additional main steam line protection.
3. Reactor vessel improvements to insure its integrity.
4. Modification of the emergency core cooling system (ECCS).

The Committee expressed the belief that these design changes and others not specified above were suitably responsive to the concerns expressed in its Sept. 10, 1969 letter. However, Dr. H. O. Monson, Dr. D. Okrent and Dean N. J. Palladino added comments to the letter; they recommended that the 2 psig secondary containment capability be increased to 10 psig (similar to that for the Verplanck reactors) and stated:

> 'We believe that this improvement in safety capability is warranted for a more densely populated site like Newbold Island, and recommend that the issuance of a construction permit be contingent on the use of a high-pressure, low-leakage secondary containment.'

The three ACRS members also recommended HPCI (high pressure coolant injection system) modifications to provide redundant protection in the ECCS. In the July 17, 1973 ACRS letter on Newbold Island Station the penultimate paragraph reads:

> 'Dr. H. O. Monson, Dr. D. Okrent, and Dean N. J. Palladino, whose additional comments were appended to the Committee's letter of August 10, 1971, believe that those additional comments are still applicable to the Newbold Island Station.'

I submit that the ACRS letter of July 17, 1973 is a conditional approval of the station and that the Atomic Energy Commission should resolve this conditionality before a construction permit is approved. When three highly qualified scientists on ACRS express reservations about a reactor installation I believe that their recommendations should be given the most serious consideration.

The ACRS letter of July 17, 1973 recognizes the unusual population risk involved in the Newbold Island siting and stipulates:

> 'The applicant has prepared a preliminary emergency plan which considers, among other things, the feasibility of evacuating the population within the Low Population Zone (LPZ) in the unlikely event of a

major accidental release of radioactivity from the plant. The applicant has also described feasibility of evacuating an area extending as much as three miles from the plant, assuming the projected population that would result from the full development envisioned by the WRIC, and has concluded that such evacuation is feasible. Detailed emergency plans, to be developed by the State of New Jersey and the Commonwealth of Pennsylvania, have not yet been completed.

The Committee concludes that a suitable emergency plan can be developed for the Newbold Island site. The Committee believes also that plans for appropriate protective measures should extend several miles beyond the proposed LPZ radius of one mile. It is essential also that plant personnel be provided with those instruments, indicators, and measurements that will define clearly the nature and course of an accident so that off-site emergency plans can be initiated at a level and on a time scale consistent with the severity or potential severity of the accident.'

Insight may be gained into the thinking of the ACRS by referring to their meeting in Washington, D.C. of June 6, 1973. There was lengthy discussion of the population density and distribution near Newbold Island indicating the evident concern of ACRS members over the population problem. The following excerpts are taken from the transcript beginning on page 39:

Dr. Monson. 'I would like to ask the applicant whether he is giving any consideration to formulating plans or seeing to it that plans are formulated by the appropriate bodies in respect to evacuating population or taking other protective measures in the event of a serious accident or cases where the accident might be more than a Part 100 type accident, in other words, a Class 9?'

Mr. Krishna. 'As far as the feasibility of our plan is concerned, whether people can be evacuated, we have looked at transportation routes available within one mile and also up to 3 miles and we see no problem in evacuating the people, even if the hypothetical accident were to take place throughout the entire area.'

Dr. Monson. 'I just want to be sure that I am right in thinking that the applicant believes that even though the probability of a Class 9 accident is considered to be extremely low, that he does not limit his interest in evacuation or the taking of other effective protective measures in the event of accident is limited to Class 8 or lower accidents and does not include in any way plans for such protection in the event of a Class 9 accident.'

The distinction between a Class 8 and 9 accident was raised and Dr. Siess stated:

'There is one way in which you could distinguish, and that is if the evacuation plans that are ultimately developed by the States consider evacuation only of the low population zone, then you would have to think that they were considering only Class 8 accidents.

But if those plans are also looked beyond and said we have plans for 2 or 3 miles, or whatever, then I think that you are looking beyond the Class 8 accident.' .

To the best of my knowledge this is the first time that ACRS has expressed a specific requirement for emergency plans to treat a Class 9 accident situation. This is clear from the wording of its July 17th, 1973 letter, already quoted, put in the frame of reference of the June 6, 1973 ACRS meeting. I submit that this introduces quite new factors into reckoning the acceptability of a reactor site, especially when the development of an adequate emergency-evacuation plan is made mandatory.

The Applicant may contend that an adequate emergency plan beyond the LPZ area is feasible, but I believe that this should be a State responsibility. The Applicant has not presented in its Preliminary Safety Analysis or in its Environmental Impact Statement or in any document placed in the Public Docket 50-354/355 any technical estimates of a Class 9 accident consequence. Since ACRS has brought the Class 9 accident out into the open, presumably because of the high population at risk in the instance of Newbold Island, I believe that the Atomic Safety Licensing Board should require that the Applicant or the Atomic Energy Commission provide data on the character of the radiation hazards anticipated beyond the Low Population Zone so that emergency plans may be formulated for the protection of the public.

My understanding of the risk situation is that States, which are the risk-takers, have not received guidelines as to the character of this class of radioactive release from nuclear power plants. There is a need to know the probable lead-radionuclide endangering people beyond the LPZ. Is this radioiodine? Or the noble gases? Or ruthenium? Or tellurium?

Dr. F. R. Farmer, the noted reactor safety expert of England, stated in his paper "Reactor Safety and Siting: A Proposed Risk Criterion" (Nuclear Safety, 8, 539, 1967):

'It is generally agreed that the isotopes of iodine, and particularly ^{131}I, carry a greater threat to health than any of the other fission products that might be released in a reactor accident, and the International Commission on Radiological Protection (ICRP) has published findings of the risk of developing thyroid cancer through the inhalation of radioactive iodine.'

No doubt the British view of reactor effluent risks is influenced by the 20,000 curie accidental release of radioiodine that occurred in 1975. However, the radiation threat potential of iodine-131 is explicit in AEC publications on reactor safety, particularly in TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites." (March 23, 1962) and in 10 CFR 100.

After discussing possible thyroid doses as a function of distance, magnitude of release, and meterology, Dr. Lapp continued as follows:

Obviously the problem of planning for an upper limit Class 9 accident assumes dimensions beyond the capability of a metropolitan area to cope with the radiological consequences. Nonetheless, the significance of the ACRS comments, already specified as to Class 9 accidents in the instance of Newbold Island, is that communities must prepare emergency plans for what may be called a lower limit Class 9 situation.

If one assumes that the significance of the ACRS recommendation for emergency preparedness several miles beyond the LPZ is that one may have to plan for 300 rem doses to the adult thyroid in this zone, then this is tantamount to projecting infant thyroid doses in the range of 1,000 rem 4 or 5 miles from the reactor site, depending on how the word "several" is interpreted.

If a 1,000 rem dose to the infant thyroid is assumed for a 4 mile downwind distance, then Fig. 1 would predict that a

30 rem dose would be sustained out to 42 miles from the accident site. I believe that this is a straightforward extrapolation of the ACRS commentary. I believe that the problem of planning for such a large area evacuation or even implementing other protective measures within the area poses an immense challenge to the affected communities, the more so because of a lack of adequate planning data. Even with the best possible predictive data, it will be difficult in the extreme to selectively evacuate communities when one considers that in Pennsylvania they are contiguous with zones of high population density. It should be emphasized that the Philadelphia metropolitan area has a population approaching 5 million. The distribution of this population is quite different from that radiating out from the Indian Point and Zion sites that have been approved by the Atomic Energy Commission for reactor siting and operation.

In presenting this line of argument to the Board, I wish to point out the importance of providing States with authoritative information on which to base plans for emergency procedures to be implemented in the event of a serious reactor accident. For my part, I believe that it is not a prudent policy to site two high power nuclear reactors at Newbold Island and I would argue that the construction of these plants be disapproved.

If it is the decision of the Board to recommend construction of the plants, I would then urge that special precautions be taken to insure against the accidental release of any large quantity of radioactivity to the encironment. As noted, three members of the Advisory Committee on Reactor Safeguards have recommended additional safeguards and I agree with their recommendation.

Assuming that the Newbold Island reactors are authorized for construction I recommend that the Applicant be required to instrument the reactor complex with accident diagnostic equipment so as to permit prompt acquisition of information about the magnitude and nature of any accidental accumulation of radioactivity within containment and its rate of release from containment, together with accurate data on the prevailing meteorology. The Applicant should be required to develop models for prediction of the patterns of radioactive fallout and envelopment of downwind sectors.

I call attention of the Board to AEC document ORNL-NSIC-27 "Review of Methods of Mitigating Spread of radioactivity From a Failed Containment System" (Sept. 1968). I would recommend that the Applicant be required to review the 15 itemized modes of activity control and estimate their efficacy for the Newbold Island installation. In particular, I would stress the importance of considering sodium hydroxide sprays as a means of trapping elemental iodine within containment.

Rather than allow ground or low level release of radioactivity from a failed containment structure, I would urge that the Applicant be required to analyze the efficacy of a trap-stack for engineered release of radioactivity. This is a tall stack equipped with devices for trapping radioactive effluents or maximizing plate-out of radioiodine. Such a trap-stack serves a double purpose. One, it reduces the release of such emitters as radioiodine to the atmosphere. Two, the stack serves to inject the effluents at a sufficiently high altitude so that the concentration of radioactivity in the immediate vicinity of the plant, i.e. the LPZ and the region contiguous to it, is reduced by perhaps two orders of magnitude or more. Admittedly, certain atmospheric conditions can cause stack-released effluents to come to earth beyond this LPZ and contiguous zone, but such radioactivity will be markedly reduced in concentration due to dilution effects.

Prompt acquisition of reliable information about the time-rate of activity release is critically important to countermeasures within the LPZ and outlying zones. It will be hazardous in the extreme to undertake evacuation procedures that might catch people out in the open fully exposed to radioactivity; it would be preferable to keep people in a sheltered, buttoned-up, condition or to provide them with masks.

In conclusion, I believe that the Board is faced with a problem that is greater than that of a singular nuclear power site. If Newbold Island is approved, then the way is open for utilities to site future plants in locations of equal or higher population-at-risk situations. The logic of my argument is substantiated by reference to Figure 2 (taken from AEC Regulatory Supplement No. 1 to the Safety Evaluation, Docket No.s 50-354,355) in which the population at risk in the annular zones, 5-10, 10-15 and 15-20 miles, clearly exceeds any AEC approved reactor site. The May 17, 1973 Supplement No. 1, Regulatory Safety Evaluation, states on page 3: "Comparison with the population distribution characteristics of previously approved sites is a part of the review of population characteristics of the site environs in determining the acceptability of a site. The same page of this Supplement, referring to Figure 2 attached, states: From these figures, it can be seen that the Newbold Island curve lies within or very close to the population distribution curves for reactor plant sites that have been judged to be accepable for other facilities, departing from those curves at distances between 5 and 30 miles." It is not clear from this statement as to what criterion the AEC Regulatory Staff uses to judge site acceptability, but the inference is that since Newbold Island is close to Zion and Indian Point risk curves.

I contend that the ad hoc case by case reactor siting policy of the Atomic Energy Commission is a stairway to close-in metropolitan siting. Newbold Island is another step up on this escalatory policy and, if approved, it will open the way to Burlington-type sites once disfavored by the AEC.

In an unusual step, the Regulatory Staff on October 5, 1973, issued an opinion that, while the Newbold Island site was acceptable, they had applied considerations of the National Environmental Protection Act (NEPA) and concluded that the Applicant had other significantly less populated sites available. They, therefore, had concluded that the reactors should not be built at Newbold Island, but elsewhere. Shortly thereafter, Public Service announced it had decided to move the Newbold Island station to the relatively rural Salem site.

The Regulatory Staff did not take a similar position on the Limerick Station, however, and these reactors received a construction permit and were constructed. The Staff had previously argued to the ACRS that Limerick was "equivalent or worse" than Newbold Island when they combined population and meteorological considerations. Yet, the Staff did not apply NEPA; nor did they supply any rationale for the difference in actions taken on Newbold Island and Limerick.

The Regulatory Staff letter to the Applicant concerning Newbold Island is presented on the following pages, together with the response of Public Service.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

October 5, 1973

Mr. Robert L. Smith
President
Public Service Electric and Gas of New Jersey
80 Park Place
Newark, New Jersey 07101

Dear Mr. Smith:

The Regulatory staff is now in the process of completing a Final Environmental Statement for the Newbold Island Nuclear Power plants.

An important requirement in the preparation of an environmental impact statement for a nuclear power plant is, of course, a consideration of alternative sites. On the basis of balancing all the various factors which must be considered at this location, including, particularly, population distribution, the staff concludes that the alternative location of these facilities at Artificial Island, adjacent to Salem Units 1 and 2, which are presently under construction, is a more desirable alternative from an environmental standpoint. This conclusion will be incorporated in the Final Environmental Statement for the Newbold Island nuclear power plants.

The principal factor leading to this conclusion is the fact that the population density at the Newbold site is significantly larger than at the Salem location. For instance, our projections for 1980 show that within five miles' distance, the Salem location will have a population of about 4,700 persons, and the Newbold Island site will have approximately 125,000 persons. Within a 30-mile radius in 1980, Salem will have about 1,000,000 persons whereas Newbold Island will have over 4,500,000.

We are informing you of this conclusion prior to the issuance of a Final Environmental Statement so that if you should decide to accept the staff's position, an amendment to the application to change the plant location can be prepared as soon as possible. In the event you should decide to amend the application to use the Salem site, the staff would be prepared to be ready for a hearing within four months after receiving the amendment. This is possible because of the Final Environmental Statement issued for Salem 1 and 2 on April 4, 1973, as well as the fact that the Newbold plant has been subjected to a safety review. A change in the design of the Newbold plant, however, would require additional time for our review. If you elect to continue to pursue the Newbold location, please let us know promptly so that we can then complete the Final Environmental Statement and proceed to a hearing on it.

Very truly yours,

L. Manning Muntzing

L. Manning Muntzing
Director of Regulation

RECEIVED

L. Smith

Public Service Electric and Gas Company 80 Park Place Newark, N.J. 07101 201/622-

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U.S. ATOMIC ENERGY COMMISSION

October 19, 1973

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Mr. L. Manning Muntzing
Director of Regulation
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Muntzing:

Your letter of October 5, 1973 stated that the Regulatory staff has concluded that Artificial Island, adjacent to Salem Units 1 and 2, is a more desirable location for the two nuclear units presently planned for Newbold Island. Further, you explained that the staff had reached this conclusion during the consideration of alternative sites, which is an important requirement in the preparation of the final Environmental Impact Statement.

Discussions with you and your staff subsequent to the receipt of your October 5th letter have brought out the fact that Newbold Island is not an unacceptable site for a nuclear power plant but merely that Artificial Island is a more acceptable site. As you are well aware, the Advisory Committee on Reactor Safeguards has issued three letters during the past four years, all of which have reported favbrably on the Newbold Island site. The first letter, dated May 9, 1969, stated that "the Newbold Island site is not unacceptable with respect to the health and safety of the public." The last letter, dated July 17, 1973, stated that the Committee believes that "the Newbold Island Nuclear Generating Station Units 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public." It was on the basis of these letters issued by the Advisory Committee on Reactor Safeguards, together with a lack of any opposing views by the members of your staff, that we proceeded with plans for construction of two nuclear units on Newbold Island. You can appreciate that with this background of approval your suggestion that we relocate to Artificial Island came as something of a shock.

During the past two weeks, we have been evaluating the pros and cons of relocating the Newbold Island units and have come to the reluctant conclusion that we should amend our Application to locate these units adjacent to Salem Units 1 and 2. This relocation will impose an economic penalty because of the higher plant construction and transmission costs at Salem but we recognize the importance of unqualified AEC approval of the nuclear plant installation. Hopefully, the economic penalty will be mitigated by virtue of a more expeditious licensing procedure.

A few days ago, we received a letter from Richard L. Sullivan, Commissioner of the Department of Environmental Protection of the State of New Jersey, who is also Chairman of the New Jersey Nuclear Energy Council. In his letter, Commissioner Sullivan stated that if the Newbold Island units are relocated to the Salem site, the State of New Jersey will move to join with the Regulatory staff in support of the Application.

We appreciate your interest in expediting the licensing procedure for the relocated units and expect to submit an amended Application by November 1, 1973, so that a hearing can be held by March 1, 1974.

Sincerely,

R J Smith

2.18 SOME OTHER REACTOR SITE CONSIDERATIONS: 1973-PRESENT

Newbold Island was the site having the highest surrounding population density to be approved by the Regulatory Staff and the ACRS. The decision by the Staff to apply NEPA considerations to Newbold Island came as a surprise. A year earlier, the Regulatory Staff had submitted to the ACRS for discussion a reactor site index, RSI, that the Staff had developed. This draft Staff approach was discussed with the ACRS Subcommittee on Siting Evaluation at a meeting on January, 1973.

The reactor site index was the product of a site population factor (SPF) and a plant design factor (PDF). The site population factor was proportional to population, weighted by a factor—$(\text{distance})^{1.5}$—to allow for meteorological dispersion, using a sector approach in a fairly conventional way. The plant design factor took into account the differences in plant releases by comparing doses at the low population zone boundary, using the traditional approach of Part 100, allowing for engineered safeguards.

What is somewhat surprising is that, as late as October, 1972, the Regulatory Staff was still completely ignoring considerations of Class 9 accidents in the development of a reactor site index. In fact, the proposed method gave a high (bad) plant design factor to some plants having a large exclusion area and low population zone distance. The curious result was that on a scale where Indian Point 2 had a rating of unity, Newbold Island had a lower (substantially better) RSI, while relatively remote sites like Peach Bottom and McGuire were much worse than Indian Point 2.

The Subcommittee seriously questioned the proposed Staff RSI. Some excerpts from the meeting minutes and the Staff proposal are on the following pages.

SITING EVALUATION SUBCOMMITTEE MEETING
DES PLAINES, ILLINOIS
JANUARY 5, 1973

A meeting of the Siting Evaluation Subcommittee was held in Des Plaines, Illinois, on January 5, 1973, to discuss the Regulatory Staff's proposed reactor site index, RSI, transmitted to the Committee by letter dated October 4, 1972 (see Attachment 1). Present at this meeting were the following:

ACRS	Regulatory Staff
H. O. Monson	H. Denton, DL
M. Bender	B. Grimes, DL
H. M. Hill	A. Kenneke, DL
H. B. Piper, Consultant	J. Kohler, DL
M. C. Gaske, Staff	

Executive Session

Dr. Monson pointed out that the Regulatory Staff has proposed a site index (Attachment 1) and has requested comments regarding this index.

Dr. Monson reviewed the three specific requirements regarding Part 100 doses--the requirements regarding the exclusion area, the low population zone, and the population center distance. The low population zone distance can be increased through use of a better evacuation plan. In a sense, the Part 100 guideline values would permit any number of people to receive a whole body dose of up to 25R but not even one individual to receive as much as 26R whole body dose; in other words, all emphasis is on dose to the individual as compared to total dose to the population. Dr. Monson thought that the three Part 100 guidelines are such that a reactor could be made acceptable for almost any site if no other means of assessment were used. He stated that it is clear to reactor vendors and utilities that the regulatory bodies in the U.S. have refused to accept construction of reactors at some sites even though they met the three specific criteria in Part 100. It was reported that the British maintain control over all construction out to some distance from the site, but that they would approve the use of a site generally equivalent to Burlington on the basis of population.

Dr. Monson explained the general features and the history of the development of the SPI computational method. The Regulatory Staff was provided with a written description of the method of calculating SPI values in December, 1970. The SPI calculations do not include consideration of the effect of wind direction. Dr. Monson believed that, if meteorology is to be considered, not only wind direction, but also wind speed and stability conditions should be included on a probabilistic basis. Sufficient information regarding all of these parameters is ordinarily not available.

In calculating the Regulatory Staff's Site Population Factor (SPF), a man-Rem type value is calculated out to various distances from each site and compared with that for a reference site (see Attachment 1 for description of the method). Whatever distance results in the worst SPF is the one used in determining the maximum Factor. Dr. Monson thought that this tended to make the degree of conservation of the SPF different for each site. He pointed out that the Site Population Factor is multiplied by the Plant Design Factor (PDF) to obtain the Reactor Site Index (RIS), as indicated in Attachment 1. If the efficiency of the containment gas filter system is assumed to be .995 vs .990, this might tend to allow twice as many persons in the area surrounding a site to be present and the Reactor Site Index still be the same. Dr. Monson thought that there should not be such reciprocity allowed between calculated dose and population density. Mr. Bender indicated concern that, with this approach, there is a tendency to give credit for engineered safeguards that may not function.

Regulatory Staff

A series of viewgraphs was presented by the Regulatory Staff, Attachment 2. The Regulatory Staff indicated that their proposed Reactor Site Index represents a departure from previous indices in allowing engineered safety feature effectiveness to be quantitatively substituted for distance. The third figure, "Cumulative Population as a Function of Distance" contains a population vs distance curve for the assumed "standard site" having a population density of 1000 persons per square mile.

Dr. Monson pointed out that use of the maximum Site Population Factor would result in there not being the same degree of conservatism involved in the calculation of the Reactor Site Index for each site. Mr. Bender stated that the Reactor Site Index calculated by the Regulatory Staff indicates that the Diablo Canyon Facility is overdesigned relative to safety. Mr. Grimes stated that Part 100 requirements would still have to be met, and indicated that one advantage of the proposed method of rating sites is that the Regulatory Staff might be able to tell applicants not to bother to propose facilities where the Reactor Site Index would be greater than 1.0.

The last page of Attachment 2 lists the Regulatory Staff's conclusions. Mr. Denton said that there is an inclination in the Staff's present thinking to rule out the use of any sites whose population is greater than that of the standard site at any distance. Mr. Denton thought that reactor sites should be as remote as practical. He said that there is no evident trend toward utilities wanting to use high density sites. Dr. Monson pointed out that Consolidated Edison and Southern California Edison both have indicated reasons for strongly wanting close-in sites.

Dr. Denton thought that it was difficult to justify and countenance the differences in the RSI for Oconee 2 and 3 (1.4) vs Diablo Canyon (0.1). Dr. Monson said that, if adequate safety has been provided for both facilities, then there should not be a problem. Mr. Bender was concerned that the use of the RSI might cause applicants to lose incentive to use remote sites. Mr. Bender inquired whether the Regulatory Staff prefers distance in preference to safeguards, and Mr. Denton indicated that they do. Mr. Denton stated that the general mood is not to allow anything less conservative relative to reactor safety.

Dr. Monson said he believes the Commission needs more complete siting policy and that Part 100 is not sufficient. In particular, the Commission needs a policy that more specifically takes into account the remote possibility of accidents greater than Part 100. Mr. Denton reported that the "Rasmussen Group" has been asked to concentrate on probabilities and not consequences of accidents. He said that, if Class 9 accidents are considered "credible", this may preclude the construction of reactors in the Northeast United States. Dr. Monson indicated he believed that the number of man-Rem that might be sustained from a very wide range of magnitudes of accidents is an important consideration and is not adequately taken into account in the Regulatory Staff's proposed method of calculation.

Dr. Monson pointed out that the Committee had at one time considered a limit line which corresponds to an acceptable population as being one that, out of all distances (to a specified maximum distance), was less than 4000 R^2, where R is the distance from the reactor in miles. That is very close to the Regulatory Staff's proposal mentioned of a limit of about 1000 persons per square mile.

The Commission has stated in a number of places that Class 9 accidents need not be considered in environmental analyses because their probability of occurrence is so low (and, therefore, the risks). Dr. Monson stated that he questions whether that view can be substantiated.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

OCT 4 1972

Joseph M. Hendrie, Deputy Director for Technical Review, L

PROPOSED REACTOR SITE INDEX

Enclosed is a report which describes a proposed method of evaluating site acceptability under Part 100 considering both demography and plant design characteristics. The method permits a quantitative determination of whether the assumed effectiveness of the accident mitigation features of a plant at a particular site is sufficient to limit total population impact within an acceptable envelope.

We have weighted population by d^{-n} where d is distance and n is 1.5 (Site Population Factor, SPF). The relative effectiveness of plant characteristics in reducing accidental releases of radioactive materials was determined by comparing calculated LOCA doses at a uniform distance (Plant Design Factor, PDF). The product (Reactor Site Index, RSI) of these two indices is therefore a relative measure of potential population impact as limited by plant design. Some typical results for a range of representative sites are summarized below.

Facility	Site Population Factor, SPF	Plant Design Factor, PDF	Reactor Site Index, RSI
Ravenswood	58	---	---
Burlington	3.5	---	---
Indian Point 2	1.0	1.0	1.0
McGuire	0.27	8	2.2
Peach Bottom	0.3	6.5	1.9
Oconee 2 & 3	0.09	15	1.4
Zion	1.03	1.3	1.3
Newbold Island	1.6	0.5	0.78
Diablo	0.04	3.0	0.1

All the values above have been normalized to Indian Point 2, a PWR plant with chemical sprays, internal charcoal filters, and a low containment leak rate.

Those plants with a PDF calculated to be significantly greater than one but a relatively low SPF are those which could have added additional dose-mitigating features. For future cases, we propose that plants with a low SPF be required to maintain a PDF such that the RSI is about unity. This would be done by requiring more dose-limiting features for the LOCA than the Part 100 dose criteria alone might require. For plants with an SPF above unity, improvements to plants would be required for the entire spectrum of possible causes of accidental releases from the plant, such that the overall PDF would offset the higher SPF and likewise limit the RSI to about unity.

Harold R. Denton

Harold R. Denton, Assistant Director
for Site Safety
Directorate of Licensing

Enclosure:
Proposed Reactor Siting Index

cc: J. F. O'Leary
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REPORT ON A PROPOSED REACTOR SITING INDEX

October 5, 1972

By A. P. Kenneke
and J. E. Kohler

I. Introduction

The paper discusses an approach to reactor siting which utilizes a relative index for taking into account population distribution as well as plant and site characteristics. The index could be used to supplement existing site evaluation methods in the case of relatively more populous sites.

II. Need for Siting Index

The basic regulation governing AEC siting policy is 10 CFR Part 100, "Reactor Site Criteria". Part 100 governs siting through the use of the concepts of exclusion area (EA), low population zone (LPZ), and population center distance (PCD). With respect to individuals and populations in the immediate vicinity, Part 100 provides that calculated doses at the EA and LPZ boundaries due to a design basis accident (DBA) be less than specified values. However, Part 100 expresses limitations with respect to total population dose only in qualitative terms.

Cases where the need for quantitative population criteria have been more acute include Ravenswood (1964), Burlington (1966), and, more recently, Limerick and Newbold Island. As a result of concern about population, the first two sites were withdrawn from consideration. Although the staff has proposed granting construction permits in the latter two, it has nevertheless placed more stringent design requirements on the applicant. Without a generally accepted siting index, judgments about the adequacy of the added requirements have been largely qualitative.

III. The Proposed Reactor Site Index

A useful siting index has been developed which includes these characteristics:

1. It does not supplant the present individual dose approach, but complements it;
2. It accounts for: a) population distribution; b) plant characteristics, including engineered safety features (dose-mitigating); and c) site characteristics, especially meteorology.

The proposed Reactor Site Index (RSI) is the normalized product of a site population factor (SPF) and a plant design factor (PDF). The factors are normalized to a given site. As defined, lower RSI values correspond to more favorable plant-site combinations.

The site population factor takes into account the fact that the risk to the population a) decreases with distance from the source, and b) is not the same in all directions. The first can be accounted for by means of a simple (distance)$^{-1.5}$ factor in statistically weighting the population at various distances, and the second by use of calculated annual average dilution factors by wind direction or, more approximately, by wind direction frequency in statistically weighting the population in each direction.

The plant design factor takes into account the difference in plant releases by comparing the calculated DBA doses at the LPZ, and adjusting for differences in the LPZ distances. The doses, both whole body and thyroid, are summed as fractions of their Part 100 dose criteria. The doses are adjusted for the difference in LPZ distances by means of $d^{-1.5}$ relation.

The overall reactor site index is a relative measure of total population impact as limited by plant design. A mathematical expression of the index is defined as follows.

$$RSI = SPF \cdot PDF$$

where:
RSI = Reactor Site Index
SPF = Site Population Factor
PDF = Plant Design Factor

$$SPF = \frac{\sum_I \sum_J w(I) \cdot w(J) \cdot P(I,J)}{\sum_I \sum_J w_o(I) \cdot w_o(J) \cdot P_o(I,J)}$$

where:
$w(I)$ = weighting factor for sector I
$w(J)$ = weighting factor for distance J
$P(I,J)$ = population contained in sector I between the (J-1)st and the Jth distance
(The subscript, o, denoted the reference site values.)

$$PDF = Q/Q_o \approx (d/d_o)^{1.5} \quad (F/F_o)$$

where:
Q = activity released
d = distance to outer boundary of the LPZ
F = W/25 + T/300

where:
W = Whole body dose (rems) for the LOCA at the LPZ distance
T = Thyroid dose (rems) for the LOCA at the LPZ distance

IV. Discussion of Results

A. Site Population Factor

Distance Weighting. Site population factors were calculated using a number of distance-weighting curves, and as shown in Table I, the differences between factors calculated by different curves for a given site are about ±25%. The Indian Point site has been used as the reference site.

TABLE I

SPF FOR DIFFERENT DISTANCE WEIGHTING CURVES

Plant	A	B	C	D
Ravenswood	27.9	10.4	-	-
Burlington	2.34	1.30	-	-
Indian Point	1.0	1.0	1.0	1.0
Limerick	.75	.56	.55	.55
Zion	.63	.49	.49	.48
Newbold Island	1.19	.80	.76	.93
Midland	.35	.16	.26	.25
Surry	.11	.11	.11	.11
Diablo	.019	.017	.018	.017
Prairie Island	.069	.10	-	-

where:
Column A: Weighting curve suggested by ACRS
Column B: $d^{-1.5}$ curve
Column C: "Realistic" accident curve
Column D: Rogers and Gamertsfelder curve for annual average

B. Plant Design Factor

Up to this point the calculated site population factors are independent of the plant design, and are comparable as a reactor site index only on the assumption that the source term is the same from plant to plant, whereas, in point of fact, the source term will vary depending on plant design. Plant factors considered in calculating the Part 100 doses include reactor power level, primary containment leak rate, and other dose-mitigating engineered safety features such as chemical sprays, charcoal filters, recirculation systems, and ice condenser systems. Consequently, population impact should be measured by correcting the SPF by a Plant Design Factor (PDF). Because of differences in their PDF's, different plants on the same site would have different RSI's.

For present purposes the calculated DBA Loss-of-Coolant Accident (LOCA) doses at the LPZ distance were chosen as a measure of plant release. Since dose depends on source and distance, the relative release between plants can be measured by adjusting the doses to a common distance and accounting for initial dose differences.

The dose at the EA boundary was not used because it is limited to 2 hours; if it were more limiting than the LPZ dose, this would be reflected in the LPZ dose.

Since the Part 100 dose criteria imply comparability in risk between 25 rems whole body and 300 rems thyroid, the dose adjustment can be based on the relative importance of each dose expressed as a fraction of its corresponding dose criterion.

The calculated PDF values are listed in descending order in Table IV. Indian Point 2 was used as the reference plant.

TABLE IV

PLANT DESIGN FACTORS

Plant	PDF
Oconee 2 & 3	15.6
Hutchinson Island	8.60
McGuire	7.99
Peach Bottom 2 & 3	6.52
Pilgrim	5.74
Prairie Island	5.50
Diablo Canyon	2.94
Fermi 2	2.84
Three Mile Island	2.35
Millstone 2	1.99
Limerick	1.36
Zion	1.31
Indian Point 2	1.00
Surry	.93
Midland	.63
Newbold Island	.50

Note that the PDF is given for a specific plant, and that two plants on a given site may differ in design and have different values of PDF. All plants are normalized in Indian Point 2, which has chemical sprays and containment charcoal filters. Oconee, for example, has borated sprays which are not credited in reducing LOCA doses and its design leak rate is 2.5 times that for Indian Point 2. To Part 100 dose criteria, some plants, because of site size or other site parameters, have needed little in the way of additional safeguards.

For the plants with a low PDF, the whole body dose is typically more significant than is the thyroid with respect to Part 100 values (25 and 300 rems, respectively). Therefore, achievement of PDF's below about 0.5 will be dependent upon design that provide further reduction of noble gas releases.

C. Reactor Site Index

The values of SPF and PDF given in Tables II and IV, respectively, are multiplied to give an RSI. Values for 16 plants are arranged in descending order in Table V.

TABLE V

MAXIMUM RSI VALUES FOR EACH SITE

Plant	Max. SPF	PDF	RSI
McGuire	.27	7.99	2.16
Pilgrim	.36	5.74	2.07
Peach Bottom 2 & 3	.30	6.52	1.96
Oconee 2 & 3	.092	15.56	1.43
Zion	1.03	1.31	1.35
Fermi 2	.41	2.84	1.16
Three Mile Island	.48	2.35	1.13
Limerick	.81	1.36	1.10
Indian Point 2	1.00	1.00	1.00
Hutchinson Island	.11	8.60	.95
Millstone 2	.47	1.99	.94
Newbold Island	1.57	.50	.78
Prairie Island	.11	5.50	.61
Midland	.60	.63	.38
Surry	.16	.93	.15
Diablo Canyon	.035	2.94	.10

In calculating these values directional weighting was not used; correcting might increase or decrease these values by about 50%.

As indicated the PDF is a major factor in determining the RSI. The top four on the RSI list are among the top five on the PDF list. For these plants the use of additional credited safeguards would lower their values of RSI.

The RSI's include the maximum calculated SPF wherever it occurred. In the case of Pilgrim, this was at one mile, whereas the Pilgrim RSI was significantly smaller at all other distances. On the other hand, the Zion SPF also peaked at one mile but is within a factor of two of the one mile index at all other distances.

An objection to using an SPF which is maximized at close-in distance (say, less than the LPZ) is that it does not include a sufficiently large population to be representative of the site. On the other hand, choosing the maximum SPF assures a conservative approach in considering the need for additional safeguards, as the RSI would also be considered the maximum.

V. Conclusions

The paper has described an approach to developing a rational and easily applied index, which could provide a conservative rating of reactor sites which takes into account both population distribution and plant design. Work is continuing to apply the approach described to additional sites.

> The results thus far obtained indicate that the index may provide an objective means for ascertaining plant acceptability with respect to population considerations under the requirements of 10 CFR 100.

The Regulatory Staff "Reactor Site Index" was dropped. In a press release dated April 9, 1974 the AEC made public an internal working paper on siting in response to several requests for release of the document. This document dealt with a site population factor (without any plant design factor) and proposed the possible issuance of population guidelines which would limit or make difficult the use of sites whose projected cumulative population exceeded about 1000 R2 (R in miles) and whose low population zone distance was less than two miles.

The intent of the draft guide was to encourage siting in areas of low population density. Under the proposed guide

> "the applicant would need to show that the high population density site offers significant advantages from the standpoint of environmental, economic or other factors. In addition, engineered safety features would be provided, above those required to meet the guideline doses of Part 100, so that there would be additional assurance that the risk had been minimized. Such engineered safety features would probably include the use of containment sprays with chemical additives and a secondary confinement system (sometimes referred to as a fuel containment) to hold up and filter fission products released in a postulated dersign basis accident."

The proposed guide was not adopted in 1973. In October, 1974, the Regulatory Staff issued WASH-1235, "A Technique for Consideration of Population in Site Comparison". This merely formally recorded a possible method of comparing sites and gave results using it.

However, in 1975 in Section 2.1.3 of the USNRC Standard Review Plan (NUREG-75/087) the Staff defined a quantitative approach to acceptability of population distribution, as follows:

> If, at the CP stage, the population density, including weighted transient population, proejcted at the time of initial plant

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December 1, 1977)

NRC STAFF RECOMMENDS EARLY SITE REVIEW OF PERRYMAN, MARYLAND, PROPERTY BE DISCONTINUED

The Nuclear Regulatory Commission staff has notified Baltimore Gas & Electric Company that, based on an initial review, it recommends that the Early Site Review of the company's Perryman, Maryland, property should be discontinued.

Baltimore Gas & Electric applied to the NRC for an Early Site Review of the Perryman property seeking a determination as to its suitability as a potential nuclear power plant site. The 708-acre site is in Harford County near the town of Perryman on the east bank of the Bush River, about 16 miles northeast of Baltimore. The site lies between the Consolidated Railroad Corporation rail line and the Aberdeen Proving Ground military reservation.

Due to the relatively high population density in the vicinity of the proposed site, the staff's initial review effort focused on a comparison of the significant safety-related and environmental characteristics of alternative locations. The review consisted of (1) an evaluation of the company's site selection methodology and (2) a comparative evaluation of the merits of the identified candidate sites to determine if there were obviously superior alternative sites. The staff of the Maryland Power Plant Siting Program assisted in the review. The results of the review have been published in a staff report entitled "Evaluation of Alternative Sites - Perryman Early Site Review."

Based on its review, the staff has concluded that there are sites which are obviously superior to the Perryman site from the standpoint of population density, risks posed by the proximity of potentially hazardous activities and the overall project costs and which are approximately equivalent to the Perryman site from the standpoint of other environmental considerations. Therefore, the staff recommends that the balance of the Early Site Review not be carried out and that the application be denied. This would be the staff position on the alternative site question should the application proceed to a hearing.

Some of the considerations leading to the adverse conclusion on the Perryman site include the availability of sites with less population density; the presence of the Aberdeen Proving Ground complex with its activities of transportation, testing and use of military explosives and toxic materials; and the presence nearby of the Penn Central mainline railroad track on which hazardous materials are regularly transported. The staff also took into account the fact that military missions at Aberdeen could change substantially in the future.

The review of Perryman and other sites considered by the applicant failed to identify any environmental considerations to suggest that Perryman offers significant advantages over the alternative sites. In addition, the staff raised questions as to whether the submitted candidate sites represented the realistic siting resources available to the applicant.

2.19 SOME LEGALISTIC ASPECTS OF SITING

The draft version of the Reactor Safety Study, WASH-1400, was released in the summer of 1974; the final report was published in late 1975. The authors of WASH-1400 conclude that such risk as may ensue from accidents in light water reactors clearly arise from "worse than Part 100" accidents in which core melt and a loss of containment integrity are involved. Review of WASH-1400 results by Cave (1977) had led him to conclude that the containment as currently designed reduces the overall risk by about a factor of ten from what it would be for the uncontained reactor, this on the assumption that the accident probabilities and consequences in WASH-1400 are correct. The actual factor may, of course, be different. Nevertheless there is reason to doubt that the reduced containment leak rates and safety features added to make reactors fit Part 100 at sites having a small exclusion areas and/or low population zone distance have that much effect on the actual risk.

The continued existence of Part 100 and the legal interpretations of its words by ASLB hearing boards and other judicial bodies has resulted in situations which are seemingly anomalous from the public safety point of view. The matter is well illustrated by reviewing the decision of April 7, 1977 of the Atomic Safety and Licensing Appeal Board in the matter of New England Power Company, et al. (NEP Units 1 and 2) and Public Service Company of New Hampshire, et al. (Seabrook Station, Units 1 and 2). Large sections of the decision are excerpted below.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

Alan S. Rosenthal, Chairman
Dr. John H. Buck

Michael C. Farrar
Richard S. Salzman
Dr. W. Reed Johnson

In the Matter of

NEW ENGLAND POWER COMPANY, ET AL.
(NEP Units 1 and 2)

PUBLIC SERVICE COMPANY OF
NEW HAMPSHIRE, ET AL.
(Seabrook Station, Units 1 and 2)

Opinion of the Board by Mr. Rosenthal, Dr. Buck and Dr. Johnson:

These two construction permit proceedings involving proposed nuclear power facilities in New England have brought before us a common question: whether, under existing Commission regulations, consideration is to be given in a licensing proceeding to the feasibility of devising an emergency plan for the protection (in the event of an accident) of persons located outside of the low population zone for the particular facility (hereinafter "emergency plan issue"). As will be seen, this is not a question of first impression. To the contrary, it has been squarely presented, and answered in the negative by us, in several earlier cases. For the reasons set forth below, we adhere to our prior rulings on the point.

The low population zone concept is firmly rooted in the provisions of 10 CFR Part 100, the portion of the Commission's radiological health and safety regulations which is concerned with reactor site criteria.* Three sections of Part 100 are of particular relevance to our inquiry here. Section 100.3b) defines the low population zone in terms of that

> area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as

*Part 100 was adopted by the Commission in April 1962. 27 Fed. Reg. 3509. In no respect material to the issue before us have the terms of the Part been altered since their original promulgation.

location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

[Emphasis supplied.] Section 100.11(a)(2) requires that the low population zone be

'of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.'

Finally, Section 100.11(a)(3) stipulates that "the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents"* be 'at least one and one-third times the distance from the reactor to the outer boundary of the low population zone'.

A. In Southern California Edison Co. (San Onofre Nuclear Generating Station, Units 2 and 3), ALAB-268, 1 NRC 383, 404-05 (1975), we explained how, in light of those provisions, 'the low population zone concept operates to enhance safety':

'A low population zone must be of such size that a person located at its outer boundary who is exposed to the post-accident radioactive cloud during the whole period of its passage overhead would not receive a radiation dosage in excess of certain 'reference' levels. Additionally, the situation in the interior of the zone must be such that persons located therein (a relatively low number) can be protected, by evacuation or otherwise, from receiving a larger radiation dosage in the event of an accident. Finally, of course, there must be no "population center" reaching to within one-third again the distance from the reactor to the zone's outer boundary.'

'In other words, the maximum possible size of the LPZ for any particular reactor is inflexible being set by the proximity of that reactor to the nearest population center. It may not be permissible to utilize an LPZ of that size, however, for it may include more people than can be protected by evacuation or other measures following an accident. An LPZ of smaller radius may thus have to be selected.

*I.e., "population center distance" as defined in 10 CFR 100.3(c).

> 'In that connection, the maximum permissible size of an LPZ depends on the nature of the engineered safeguards designed into the particular facility to limit radioactive emissions. That minimum size is, therefore, flexible and can be reduced as the extent of the engineered safety features is increased.

And much the same analysis had been set forth in our opinion rendered several months earlier in the same case (ALAB-248, 8 AEC 957, 961 (1974)):

> 'the design of any facility must be such as to avoid (in the event of an accident) excessive radiation dosages to persons beyond the low population zone boundary, even if those persons take no steps to protect themselves. Inside the low population zone, however, protective measures might be necessary. For this reason, the suitability of the low population zone depends upon the feasibility of protecting persons located there. Specifically, the Commission requires that the total number and density of residents within the low population zone be such that "there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident.' 10 CFR 100.3(b). The regulation makes it plain that "many factors", which "vary from case to case", must be considered in ascertaining whether, for example, "a specific number of people can ... be evacuated from a specific area, or instructed to take shelter, on a timely basis ....'

[Emphasis supplied.]

These were not just passing observations, unnecessarily to the disposition of the matter there at hand. To the contrary, our analysis had clear operative significance. One of the principal questions litigated before the Licensing Board in San Onofre related to the feasibility of evacuating persons from within the low population zone which had been proposed by the applicants--the outer boundary of which was to be three miles from the reactor. Because of the requirements of 10 CFR 100.11(a)(3), we directed, however, that the low population zone be substantially reduced in size, with the result that its outer boundary would be only 1.95 miles from the reactor. ALAB-248, supra, 8 AEC at 959-61. This action we found to provide a sufficient response to the concerns expressed with respect to evacuation: 'the reduction in size of the low population zone which must take place will make it necessary to evacuate persons in areas formerly within, but now outside that zone. As a consequence, evacuation of persons who find themselves within the reduced low population zone would be more readily achievable.' Id. at 962-63; emphasis supplied.

Midland, ALAB-123, supra, involved a proposed reactor to be located adjacent to the corporate limits of the City of Midland, (a municipality of some 35,000 persons), in an area of highly developed commercial and industrial activity.* On their appeal from the initial decision authorizing the issuance of construction permits, the intervenors complained, inter alia, of the Licensing Board's acceptance of the applicant's proposed emergency evacuation plan. As summarized by us, one of their claims in this regard was that the Board had "erroneously disregarded testimony***that it was impossible to evacuate either the low population zone or the City of Midland in the time required by the regulations". Our rejection of this assertion was short and direct:

> The regulations require a showing of the possibility of evacuation only from the low population zone, and only one outlying area of the City of Midland (primarily occupied by Dow) falls within that zone. There is ample evidence of record which indicates that evacuation of the low population zone within the requisite time period is feasible. Accordingly, there was warrant for the Board's findings on this subject.

6 AEC at 343 (emphasis supplied; footnotes omitted).

Thus, San Onofre was no more than a reiteration of prior holdings. Beyond that, it does not stand as our final word on the subject. Within the last year, in response to an assertion by an intervenor that the St. Lucie 2 proposed site did not confirm to the population standards established by 10 CFR Part 100, we had this to say:

> We have discussed the nature and application of those standards in considerable detail in other opinions. It suffices to note here that those standards call for the creation of an "exclusion area" and a "low population zone" around a reactor. The applicant must control and territory within the exclusion area. It need not have such control over the low population zone, but there must be a sufficiently small number of people in that zone to assure that steps for their protection (such as evacuation) can readily be taken in the event of an emergency. Equally important, the plant must be designed so that in the event of an

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*See Final Environmental Statement in Dickets No. 50-329 and 50-330, at p. II-2. It there also appears that the resident population within five miles of the site was approximately 41,000; in addition 34,000 persons were employed or transacted business within the same area.

accident, radiation dosages at the respective zone perimeters will not exceed certain levels. What this means (all other things being equal) is that the smaller these two areas are drawn, the greater the efficacy of the safety devices built into the plant must be in order to retain post-accident radiation dosages below the guideline levels.

The population standards contain the additional requirement that no "population center" larger than 25,000 persons may be closer to the reactor than one and one-third times the distance from the reactor to the outer boundary of the low population zone. If that requirement is not met, however, a proposed reactor does not necessarily have to be relocated nor an existing one abandoned. Instead, a smaller low population zone may be selected so long as the plant has the capability, or can be redesigned, to limit further the potential radiation dosages that could be encountered at the boundary of that zone.

The intervenor's arguments, as well as the evidence adduced below, had as their starting point the applicant's proposal that the low population zone would have a five-mile radius. As it had the authority to do, however, the Licensing Board imposed a condition upon the applicant which had the effect of requiring it to utilize only a one-mile low population zone. This had a most significant effect, for the controversy below was concerned almost exclusively with population growth at greater distances. In contrast to the land area lying between one and five miles from the plant, virtually all the land within a one-mile radius of the reactor is owned by the applicant. Accordingly, there is no longer any room for an argument that the population within the [St. Lucie] low population zone may become too large to permit protective steps, such as evacuation, to be taken in the event of an accident. Nor is there any evidence that projected nearby "population centers" will come too close, i.e., to within one and a third ones of the reactor.

Florida Power & Light Co. (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-335, NRCI -76/6 830, 833-34 (June 29, 1976), modified on other grounds, sub nom. Hodder v. NRC (D.C. Cir. No. 76-1709, October 21, 1976) (emphasis supplied; footnotes omitted).

III.

It is clear from the foregoing that the staff and the intervenors are here asking us to overturn a line of appeal board

authority which, at the very minimum, is well-entrenched.*
Although we do not suggest that the doctrine of state decision admits no exception, in the present circumstances there are compelling reasons why we should be slow to accept that invitation.

There are still other considerations which militate in favor of leaving the staff and the intervenors to their rule-making remedy. From the tenor of much of the argument presented to us, it appears that the attempt to have us overturn our prior holdings on the issue here-involved has been prompted in large measure by a current belief that, in some situations at least, there may be good reason to include persons outside the low population zone within the scope of the emergency planning requirement, notwithstanding the provisions of 10 CFR 100.11(a)(2) establishing radiation dose limits for the low population zone boundary. But whether or not this belief is meritorious is a question more appropriately explored in rule-making where (1) all information bearing upon the matter can be received and evaluated (as it should be) on a generic basis; and (2) be it then concluded that emergency plans should not always be restricted in ambit to the exclusion area and low population zone, specific standards can be prescribed for determining, with respect to each proposed reactor site, whether and to what extent the applicant must concern himself with devising protective measures for persons outside the low population zone.**

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*Moreover, as has been seen in the case of the staff we are being called upon now to reject an interpretation of unaltered Commission regulations which the staff itself pressed upon the adjudicatory boards many years ago and in the adoption of which the staff apparently acquiesced for a considerable period of time.

**Although the staff insists that in some (albeit not all) instances such measures must be devised, it points to nothing in Part 100 or elsewhere in the regulations which might be taken to indicate, even in broad outline, what those instances might be. The absence of standards is a still more serious matter when viewed in the context of the Coalition's argument that population density or distribution outside of the low population zone might be reason enough for a licensing board to find the site unacceptable. Surely, in assessing the acceptability of a site which it has under consideration, a utility should have some basis--derived from the content of Commission regulations--for forecasting whether the situation obtaining in the area beyond the low population zone (an area which, unlike the LPZ, has no fixed boundaries) might occasion the outright rejection of that site.

Insofar as the area within the low population zone is concerned, Section 100.10(b) clearly does carry the message--at least when read in conjunction with Section 100.3(b)--that the site's acceptability will hinge upon the population density and use characteristics not being such as to preclude the taking of appropriate measures to protect persons within the area in the event of an accident.* But with respect to the area outside of the low population zone, no similar message is conveyed. Indeed, the quite different import of the reference in Section 100.10(b) to the population center distance becomes readily understandable when one focuses upon how Part 100 defines that term and then utilizes it in the ascertainment of site suitability. As previously noted, (1) the population center distance is "the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents" (Section 100.3(c); and (2) that distance must be "at least one and one-third times the distance from the reactor to the outer boundary of the low population zone" (Section 100.11(a)(3)). The population center boundary being ascertained "upon consideration of population distribution" (ibid.), the significance of the population density of the "site environs" in terms of the population center distance is thus manifest. The closer the population center, the smaller the permissible low population zone. The smaller the low population zone, the larger the possibility that the site will not be acceptable for the reason that the applicant will not be able to provide sufficient engineered safeguards to insure observance, in the event of an accident, of the dose limitations applicable to the low population zone boundary (Section 100.11(a)(2)). See San Onofre, ALAB-249, supra, 8 AEC at 959-61; ALAB-268, supra, 1 NRC at 404-06.**

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*In this connection, as early seen Section 100.3(b) provides that no "permissible population density or tital population within [the low population] zone" is being specified "because the situation may vary from case to case". It adds that: Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

**It is argued to us that the use of the word "including" in Section 100.10(b) implies that the Commission there had in mind more than just the exclusion area, low population zone and population center distance. Perhaps so; perhaps not. In any event, there is no room for the still further inference that the Commission was making site acceptability hinge upon the feasibility of protecting persons in some unspecified area.

Consideration of Accidents in Implementation of the National Environmental Policy Act of 1969

The Atomic Energy Commission has under consideration amendments to Appendix D of its regulation 10 CFR Part 50, Licensing and Production and Utilization Facilities, an "Interim Statement of General Policy and Procedure: Implementation of the National Environmental Policy Act of 1969 (Public Law 91-190)." The proposed amendments would, by the addition of an annex to Appendix D, specify certain standardized accident assumptions to be used in Environmental Reports submitted by applicants for construction permits or operating licenses for nuclear power reactors pursuant to Appendix D.* The accident assumptions and other provisions of the proposed amendments would also be applicable to AEC draft and final Detailed Statements.

The Commission invites written comments or suggestions from all interested persons on the proposed amendments set forth below as well as on the treatment of the probabilities of the accidents.

Further insight into the Regulatory Staff approach to the role of Class 9 events in reactor licensing comes out of the record of the ASLB hearing in 1977 on the Black Fox reactors. During a pre-hearing conference, a member of the ASLB raised a question of the extent to which Class 9 accidents can be dealt with in a hearing.

The Counsel for the NRC Staff filed a memorandum (as did the applicant), following which the ASLB reached a decision. The NRC memorandum and the ASLB memorandum and order are reproduced below.

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*In conjunction with the revision of Appendix D on September 9, 1971 (36 F.R. 18071), there was transmitted to applicants for licenses to construct or operate nuclear power plants, and made available to the public, a document dated September 1, 1971, entitled "Scope of Applicant's Environmental Reports with Respect to Transportation, Transmissions Lines and Accidents." This document was a supplement to the guidance provided to license applicants in the "Draft AEC Guide to the Preparation of Environmental Reports for Nuclear Power Plants," dated February 19, 1971, also made available to the public.

3/24/77

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of PUBLIC SERVICE COMPANY OF OKLAHOMA, ASSOCIATED ELECTRIC COOPERATIVE, INC. AND WESTERN FARMERS ELECTRIC COOPERATIVE, INC. (Black Fox, Units 1 and 2)	Docket Nos. STN 50-556 STN 50-557

STAFF MEMORANDUM REGARDING TREATMENT OF EXTREMELY LARGE ACCIDENTS IN THE HEALTH AND SAFETY HEARING

I. INTRODUCTION

A third prehearing conference in the above captioned proceeding was held on February 15, 1977, in Tulsa, Oklahoma. During this conference, Mr. Shon, a member of the Atomic Safety and Licensing Board (Board) raised a question "of the extent to which they (Class 9 accidents) can be dealt with in the safety phase of the hearing." This question was discussed during the prehearing conference.(Tr. 215-220).

In the Order dated March 9, 1977, following the third prehearing conference, the Board indicated that it has taken the matter of treatment of extremely large accidents under advisement since such matters would be addressed "in the health and safety hearing, if at all." However, the Board stated that it "is not convinced that treating such accidents is generically excluded irrespective of the hypothesized mechanism [e.g. pressure vessel failure or ECCS failure]."

The Board gave the Staff and Applicants an opportunity to introduce "authority showing a generic exclusion" of the treatment of such accidents in the health and safety phase of the hearing. This memorandum is submitted in response to that opportunity.

II. DISCUSSION

The question presented here is whether accidents with consequences greater than the design basis accidents addressed in an Applicant's Preliminary Safety Analysis Report (PSAR) and Staff's Safety Evaluation Report (SER) may be generically excluded from consideration during the health and safety hearing irrespective of the hypothesized mechanism for such accidents. For the reasons indicated below, it is the Staff's position that such accidents may not be generically excluded from consideration in Commission proceedings.

This question arose during a discussion of Class 9 accidents. The Class 9 accident, a term developed to facilitate analysis of the environmental impacts of nuclear power plants, is an accident which

> "involve[s] sequences of postulated successive failures more severe than those postulated for the design basis for protective systems and engineered safety features." Proposed Annex to Appendix D to 10 CFR Part 50, 36 Fed Reg 22851 (Dec. 1, 1971).

The Commission characterized a Class 9 accident as an accident which is "sufficiently remote in probability that the environmental risk is extremely low," and indicated that such an accident need not be addressed further in environmental reports and statements.

As a result of these two aspects of the original definition, the term Class 9 has been used to mean either: an accident with massive offsite consequences regardless of how improbable it may be; or, events leading to an accident with consequences greater than those of the largest design basis accident where such events are asserted to have a sufficiently high probability that it should properly be considered in evaluating a facility.

It is this latter aspect of the term Class 9 which is proper for consideration in the safety hearing as the Board may need to consider whether the design basis accident has been appropriately determined.

The focus of the safety phase of the hearing is on whether the facility conforms to the Commission's safety regulations (principally 10 CFR Parts 20, 50, and 100). In order to satisfy the Commission regulations, a facility must be designed to protect against a range of transient and accident events described in 10 CFR Part 50, including Appendix A to Part 50. The facility must have an ECCS system capable of providing adequate cooling in the event of a loss of coolant accident, as defined in 10 CFR §50.46 and Appendix A. The facility must also be designed to withstand natural phenomena in accordance with General Design Criteria (GDC) 2 of Appendix A to Part 50 and Appendix A to Part 100. In addition, the facility must be designed to withstand anticipated occurrences and accidents, even though offsite or onsite power is lost. (GDC 17) Onsite power must withstand a single failure. (GDC 35) The primary coolant pressure boundary components must conform to the Code requirements which cover various upset, emergency and faulted condition loadings. 10 CFR §50.55a.

In addition to the Part 50 analyses, the adequacy of facility design must also be tested by application of the requirements of Part 100. Part 100 requires the identification of a

> major accident, hypothesized for purposes of site analysis or poltulated for consideration of possible accidental events, that will result in potential hazards not exceeded by those from any accident considered credible." 10 CFR §100.11 (a). footnote 1.

The consequences of this accident are analyzed to determine the radiation doses at the boundary of the exclusion area and low population zone. A reactor must be designed and located such that the doses at the boundary of the low population zone for the entire course of the accident do not exceed 25 rem wholebody and 300 rem thyroid for any credible accident. 1/

1/ Part 100 also sets forth requirements for the exclusion area boundary and for population center distance, etc.

If any mechanism or accident event results in consequences which exceed the specified boundary doses and is shown by the evidence to be credible, the facility cannot be licensed unless it is modified to conform to Part 100. Stated another way, if any credible mechanism or accident event results in consequences larger than those of the design basis accidents addressed in the PSAR and Staff SER, that event must become a design basis for the facility and the facility design must be modified to satisfy Part 100 before the facility may be licensed.

Consequently, with respect to Part 100 considerations, it is always relevant to determine whether a given postulated large accident event is credible. If it is not, it need not be considered in determining compliance with Part 100. If it is credible, it must be considered. Consideration of consequences of credible accidents may be limited to the analysis of the doses at the exclusion area and low population zone boundaries. There is no need to consider such consequences further because the facility design must be modified or the site changed if the Part 100 boundary doses are exceeded.

Notwithstanding the general right to question the credibility of accidents, some issues may not be raised. For example, challenges to the Commission's regulations are not permitted in absence of a waiver based upon a showing of special circumstances. 10 CFR §2.758. Similarly, evidence concerning the consequences of failure of the pressure vessel, absent circumstances of special safety significance is not admissible. Consolidated Edison Company of New York (Indian Point, Unit 2), CLI-72-29, 5 AEC 20, 21 (1972).

III. CONCLUSION

Consideration of accidents with consequences greater than those of the largest design basis accident are not per se excluded from consideration in a health and safety hearing. Such accidents may be considered to determine whether the appropriate maximum credible accident was analyzed in accordance with 10 CFR Part 100.

Respectfully submitted,

James Lieberman

James Lieberman
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 24th day of March, 1977.

2-497

For your information.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

DOCKETED USNRC MAY 27 1977

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of PUBLIC SERVICE COMPANY OF OKLAHOMA, ASSOCIATED ELECTRIC COOPERATIVE, INC. and WESTERN FARMERS ELECTRIC COOPERATIVE, INC. (Black Fox Station, Units 1 and 2)	)	Docket Nos. STN 50-556 STN 50-557

MEMORANDUM AND ORDER

The purpose of this Memorandum and Order is for the Atomic Safety and Licensing Board (the Board) to rule on certain pending matters. These matters include the contentions relating to Class Nine accidents (Third Prehearing Conference Order, pp. 7-8) and the contention relating to the handling, disposal and environmental effects of radioactive wastes (Third Prehearing Conference Order, p. 14). In addition, the Board has under advisement a motion by the Intervenors CASE and Ilene Younghein to compel discovery with regard to Interrogatory 105 submitted to the Applicants.

I. Class Nine Accidents

In its Third Prehearing Conference Order the Board elected to take under advisement the matter of admissibility of contentions concerning very large (loosely, "Class Nine") accidents. The Board invited submittals by the parties on this subject. The NRC Staff (the Staff) and Applicants were to address whether such contentions were generically excluded by regulation or case law, regardless of mechanism. Intervenors were to address postulated mechanisms for such accidents.

Staff submitted its Staff Memorandum Regarding Treatment of Extremely Large Accidents in the Health and Safety Hearing (Staff Memorandum). Applicants submitted Applicants' Memorandum With Respect to Class Nine Accidents (Applicants' Memorandum). Intervenors submitted two documents, Intervenors CASE, Ilene Younghein, Lawrence Burrell, and Clark Glymore Response to Board's Request For Credible Mechanisms (Intervenors' Response), and Intervenors CASE, Ilene Younghein, Lawrence Burrell, and Clark Glymore Supplemental Submission Re: Class Nine Accidents (Supplemental Submission).

The Board has now evaluated the material set forth in those submittals and is prepared to rule on the matter. As to the threshold question, viz, whether such accidents are generically excluded, it appears to the Board that they are not. Both the Staff and the Applicants agree that no such blanket exclusion exists (Staff Memorandum, p. 2; Applicants' Memorandum, passim). Both the Staff and the Appliants, however, point out that such accidents can be considered only if their potential for occurrence passes some test of "credibility".[1] The Board agrees. Both the Staff and the Applicants assert that certain specific mechanisms (e.g., spontaneous pressure vessel failure and failure of a properly analyzed ECCS system) are excluded by regulation and case law. The Board agrees with this proposition.

It thus remains for the Board to determine which, if any, of the mechanisms proposed by the Intervenors might pass the test of credibility and might also yield accidents larger than those previously analyzed.

The mechanisms mentioned in Intervenors' Response seem to the Board to fall for the most part into four general categories:

[1] Spontaneous pressure vessel failure (7, 26, 27);

[2] Control rod drop or ejection concommitant with scram failure or other independent failure (1, 2, 3, 4, 6, 18, 19, 21, 22);

1/ The Staff states:

> ... if any credible mechanism or accident event results in consequences larger than those of the design basis accidents addressed in the PSAR and Staff SER, that event must become a design basis.... (Staff Memorandum, p. 5)

The Applicant states:

> ... the consequences of accidents larger than design basis accidents need only be considered if a particular type of such large accident is shown to be reasonably possible. (Applicants' Memorandum, pp. 1-2)

[3] Transients without scram (5, 8, 9, 11, 14);

[4] Simultaneous independent mechanical failures (10, 12, 16, 17, 20, 23).

One of the mechanisms is not defined at all (25); another is clearly inapplicable to the reactor at hand (24); one raises the issue of sabotage (20); one simply repeats others under varied operating conditions (13); and one assumes spontaneous fuel crumbling (15).

Intervenors' Supplemental Submission consists of three parts: Parts I and II attempt to establish the credibility of spontaneous pressure vessel failure; Part III suggests that all accidents treated in the Reactor Safety Study, WASH-1400, should be treated in this hearing, however remote their probability of occurrence.

In the Board's view, mechanisms based upon the failure of a properly designed pressure vessel must be excluded from consideration because of the Commission decision in Consolidated Edison Company of New York (Indian Point Unit No. 2), 5 AEC 20, 21, fn. 5 (Oct. 26, 1972), where, as in the case at bar, no special considerations involving the particular facility have been shown to exist. In effect, such a mechanism has been defined by the Commission as not being "credible" or "reasonably possible" in the sense of the Staff Memorandum and the Applicants' Memorandum. Further, the Board will take this specific failure as representing a paradigm for a test of credibility. Using such a criterion of credibility, the Board is led to the conclusion that mechanisms which depend upon independent simultaneous failure of more than one piece of safety grade equipment also fail the test. Indeed, the single failure criterion embodied in 10 CFR Part 50, Appendix A, supports the view that the Commission does not intend such mechanisms to be examined. In this regard, the Board gives no weight whatever to the fact that a study such as that reported in WASH-1400 may have considered and evaluated such mechanisms; we do not view "credible" in the present sense as being synonymous with "worth looking at" for the purposes of the Reactor Safety Study. Indeed, the fact that WASH-1400 examined pressure vessel failure, a mechanism excluded from our proceedings by Commission decision, suggests that the Reactor Safety Study surely reviewed things we may regard as incredible for our purposes.

We are thus led to reject as failing the "credibility" criterion the general mechanisms mentioned in Categories [1], [2] and [4], supra. Category [3], ATWS, has already been assumed credible, and a contention has been admitted with it as a basis. A contention concerning sabotage has also been admitted.

As to spontaneous large scale fuel crumbling, the Board feels this matter is equivalent in credibility to multiple failures in safety grade equipment. We therefore exclude it.

We are thus led to the conclusion that the additional contentions already admitted in our Third Prehearing Conference Order at pp. 5-6 adequately permit exploration of extremely large accidents under the Commission's rules. We have further reviewed the contentions which Applicants asked us to reject as Class Nine accidents (Applicants' Answer to Revised Petitions to Intervene, dated Febuary 8, 1977, p. 23) and we conclude that these contentions either (1) involve mechanisms excludable by our credibility criterion, (2) do not suggest a specific mechanism, or (3) are effectively covered by the two contentions admitted in our Third Prehearing Conference Order.

II. Waste Disposal

This particular issue was raised by the Petitioner Roberta Ann Paris Funnell and it relates to the handling, disposal and environmental effects of radioactive wastes. In pertinent part, the contention reads as follows:

> 4. Inadequate consideration is being given to the high cost and uncertainilty[sic] of waste disposal and the moral and environmental damages are not being

> adequately considered. Eight Billio[sic] alone (per ERDA estimate) for military waste is staggering and absurd. Funnell Amended Petition, p. 5.
>
> * * * *
>
> 2. Applicants failed to properly analyze and safely handle the storage and disposal of wastes. Funnell Petition, p. 1.
>
> * * * *
>
> 4. Applicants have underestimated the environmental effects of radioactive wastes from Black Fox. Id.

The Board deferred ruling on this contention since the Commission had before it consideration of an interim rule on waste disposal. That interim rule was subsequently on March 14, 1977 promulgated by the Commission in its "Uranium Fuel Cycle Impacts From Spent Fuel Reprocessing and Radioactive Waste Management" 42 FR 13803. Further, in a Memorandum and Order of April 1, 1977 covering various proceedings, the Commission indicated that the interim rule should be used in connection with the NEPA cost-benefit balancing in particular proceedings. The Commission noted that the interim rule should act to resolve proceedings in which the issue of environmental impacts of the uranium fuel cycle is raised.[2/] Vermont Yankee Power Corp. (Vermont Yankee Nuclear Station), et al., Commission Memorandum and Order dated April 1, 1977. The contention at issue, however, does not

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2/ Although this Commission Memorandum and Order was directed to the Appeal Board, it is equally applicable to Licensing Board cases in which the environmental impact of waste disposal is sought to be raised.

relate to the interim rule. The contention, therefore, must be rejected as not being sufficiently particular, as required by 10 CFR 2.714(a). Also, the Amended Petition attempts to raise the issue of the handling, disposal and environmental effects of radioactive wastes on a generic basis. As framed in the Amended Petition, the contention is more properly a subject for any proceedings relating to the interim rule itself rather than to this particular proceeding. The Board, therefore, excludes this contention relating to radioactive waste disposal.

III. <u>Intervenors' Motion to Compel Discovery</u>

Intervenors CASE and Ilene Younghein (Intervenors) moved to compel discovery from the Applicants with regard to Interrogatory No. 105 which states:

> Describe each instance in which Applicant disagrees with the evaluation performed by the U.S. Environmental Protection Agency on the Rasmussen Report. Reference is made to the U.S. EPA Report dated August 1975 and June 1976 (EPA 520/3-75-012 and EPA 520/3-76-009).

Applicants object to the interrogatory on the basis that it does not relate to any of the issues in controversy. Applicants acknowledge that it could relate to the Class Nine accident contention which the Board had under advisement at the time the Applicants filed their response to the motion to compel. Applicants also suggest that the responses to the interrogatory would represent an undue burden in that it requests a formulation of an analysis rather than a disclosure of facts in the Applicants' possession.

In the Board's view, this interrogatory does not have a proper foundation. It has not been established that the Applicants performed any analysis of the EPA evaluation of the Rasmussen Report and Applicants' response indicates that they have not done so. Since the Applicants have apparently not performed any such analysis, it is not the proper function of discovery to require the Applicants to take such action in response to an interrogatory. Therefore, the motion to compel is denied.

BY ORDER OF THE ATOMIC SAFETY
AND LICENSING BOARD

Daniel M. Head
Chairman

Issued at Bethesda, Maryland,
this 25th day of May, 1977.

Thus, the opinion is that Class 9 accidents are not generally excluded. However, Class 9 accidents need not be considered if they are not "credible". How the finding of "credibility" is to be made, however, is ill-defined, if at all, in the NRC memorandum. The reasoning used by the ASLB was somewhat more specific, but highly qualitative.

The Commissioners, themselves, appeared to pursue a qualitative approach similar to that of the Regulatory Staff in a ruling on Indian Point Unit No. 2 in 1972, which is reproduced below.

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

CLI-72-29

COMMISSIONERS

James R. Schlesinger, Chairman
James T. Ramey
Clarence E. Larson
William O. Doub
Dixy Lee Ray

In the Matter of

CONSOLIDATED EDISON COMPANY OF NEW YORK

(Indian Point Unit No. 2)

Docket No. 50-247

MEMORANDUM AND ORDER

On September 27, 1972, the Atomic Safety and Licensing Appeal Board (Appeal Board) issued a memorandum (ALAB-71, WASH-1218(Suppl 1) at p. 488) dealing with a question certified by the presiding Atomic Safety and Licensing Board. On October 12, 1972, the Commission extended the period for review of the decision under 10 CFR 2.786 to November 1, 1972. By letter dated October 16, 1972, the Appeal Board made a correction to its decision.[1] Pursuant to 10 CFR 2.786, we now review the decision as corrected.

The certified question is as follows:

> Is it the position of the Commission that the measures taken to assure the integrity of the pressure vessels for light water reactors have been demonstrated and documented sufficiently that protection against the consequences of failure of the reactor vessel need not be included in the design of the plant and evidence concerning the integrity of the pressure vessel should not be adduced in the licensing proceedings?

The Appeal Board concluded that such protection need not be included in plant design, and that evidence concerning vessel integrity was admissible only to the extent necessary to demonstrate compliance with applicable regulations.

The pressure vessel is a cylindrical, strong-walled container housing the reactor core, which is composed of fuel elements and control rods. The reactor vessel has inlet and outlet pipe connections which convey the reactor coolant into and out of the reactor vessel. Vessels are designed in accordance with applicable ASME codes. In the pressurized water reactor at bar, the steel vessel, over 40 feet high, is constituted of walls over 8" thick, and is designed to contain water at an operating temperature of 2485/2235 psi, with inlet and outlet temperatures of approximately 555°F and 613°F, respectively.

Pursuant to its research and development responsibilities, the Commission has examined the subject of vessel integrity and continues to do so in an effort to assure utmost plant safety.[2] AEC regulations lay down strict standards to assure integrity of the vessel. The regulatory staff asserts that in all cases evaluated by it the probability of vessel failure has been found to be "so low" as not to require "consideration of the consequences of such failure in the assumptions employed in determining site suitability" (Staff Brief, August 21, 1972, p. 7). In expressing confidence as to the low probability of vessel failure, the staff referred

[1] The Appeal Board's decision referred to pressure vessels for reactors licensed for operation after July 1, 1971. The letter changed the date to January 1, 1971 (WASH 1218(Suppl 1)491).

[2] See, e.g., "Fundamental Nuclear Energy Research," Supplemental Report to Annual Report, AEC (1967), p. 114 (involving tests on vessel used for almost three years by U. S. Army in Greenland). See also "Fundamental Nuclear Energy Research," Supplemental Report to Annual Report, AEC (1971), p. 19 (outlining AEC Heavy Section Steel Technology program, which involves testing of specimen vessels).

to the regulations mentioned by the Appeal Board, certain other regulations,[3] certain proposed regulations now used for interim guidance,[4] and empirical evidence gained through years of operating experience during which there has been no evidence of a problem.

On the other hand, the regulations need not be read as excluding pressure vessel integrity as a proper area of inquiry during a licensing proceeding. The Appeal Board relied upon 10 CFR 50.55 a(c) and (g), which provide, *inter alia*, that vessels must conform to certain requirements (Appeal Board decision, WASH-1218(Suppl. 1)491). But 10 CFR 50.55 a(a)(2), which precedes these provisions, states, *inter alia*:

> *As a minimum*, the systems and components *** specified in paragraphs (c), (d), (e), (f), and (g) of this section shall meet the requirements described in these paragraphs *** (emphasis added.)

The Statement of Considerations for this regulation also refers to the requirements as a "minimum" and further provides that:

> Compliance with the provisions of the amendments and the referenced codes is intended to insure a basic, sound quality level. It may be that the special safety significance of a particular system or component will call for supplementary measures. If analysis of the system shows that such is the case, appropriate supplementary measures are expected to be adopted by applicants and licensees, or will be required by the Commission. (36 *F.R.* 11423, June 12, 1971).

In these circumstances, we cannot agree with the conclusion of the Appeal Board that compliance with the standards is sufficient automatically to foreclose further inquiry in the course of a licensing proceeding.

The regulatory staff, despite its confidence in the low probability of vessel failure, recognized that the subject of pressure vessel integrity could, in special circumstances, be a proper area of inquiry during a licensing proceeding. The staff contended that protection against the consequences of vessel failure need not be required for a particular facility "unless it has been determined that for such facility there are special considerations that make it necessary that potential pressure vessel failure be considered." (Staff Brief, *supra*, at p. 10).

We adopt the view expressed by the staff as consistent with the language of the regulation and the underlying Statement of Considerations. Where there are matters raised in a case that are of "special safety significance", supplementary measures in respect to the facility under review are an appropriate subject of hearing exploration. The certified question, insofar as it deals with the admission of evidence pertaining to pressure vessel integrity in licensing proceedings, is therefore answered in the negative.[5]

The Appeal Board further held that Licensing Boards should receive evidence necessary to satisfy them that the vessel in issue does meet the requirements of applicable regulations. In this connection, we note the statement in the Appeal Board's decision that,

> . . . all of the pressure vessels for reactors licensed for operation after January 1, 1971 are required to be designed, constructed and inspected in accordance with Section III or Section III and Addenda, and Section XI of the ASME Code,*** (Appeal Board decision, WASH-1218(Suppl. 1)491).

This language may convey the impression that all vessels in reactors licensed for operation after January 1, 1971, must conform to current codes. However, this is not correct for pressure vessels for which construction permits were issued prior to that date. For such reactors, the regulation requires that the vessel "shall" conform to the Code, Code Cases and Addenda in effect on the date of order of the vessel, and "may" conform to subsequent Codes, Code Cases and Addenda. 10 CFR 50.55 a(c)(1).[6] In addition, the Appeal Board erroneously construed the regulations as requiring compliance with Section XI of the Code

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[3] 10 CFR Part 50, App. B, "Quality Assurance Criteria for Nuclear Power Plants".

[4] 10 CFR Part 50, App. G (proposed), "Fracture Toughness Requirements". 10 CFR Part 50, App. H (proposed), "Reactor Vessel Material Surveillance Program Requirements".

[5] To warrant inquiry, the evidence must be directed to the existence of special considerations involving a particular facility in issue. Licensing Boards, in their discretion, are empowered to exclude contentions or challenges which have no substantial or *prima facie* basis, or which merely amount to generalized attacks upon the standards presently required by the regulations.

[6] The reactor vessel for the facility at bar was apparently designed in accordance with the 1965 Edition of Section III of the Code, and the 1965 Summer Addenda and Code Cases (see Initial Decision, July 14, 1972, p. 14). We intimate no view on the merits of the case, which are not now before the Commission.

The Commission ruled that compliance with the standards is not automatically sufficient to foreclose further inquiry into the matter of vessel failure in the course of a licensing proceeding. However, they supported the Staff position that protection against the consequences of vessel failure need not be required for a particular facility "unless it has been determined that for such facility there are special considerations that make it necessary that potential vessel failure be considered. However, no basis for deciding that the matter need be considered for a particular facility was given in the decision. Nevertheless, the Statement of Consideration which accompanied publication of a proposed version of 10CFR 100 on February 11, 1961, indicated that one of the proposed Part's basic objective was to assure that "even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic". This stated objective was modified in the Statement of Considerat - ions which accompanied publication of the effective Part on April 12, 1962. It was there stated that an underlying objective was to assure that "the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low.. " Furthermore, the "population center distance" criterion was included in 10 CFR 100.11)2)(3) "to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible" in recognition of the fact that "accidents of greater potential hazard than those commonly postulated as representing an upper limit (i.e., the "maximum credible accident") are conceivable, although highly improbable"* In addition, the effective Part that (where very large cities are involved, a greater distance (than that required by the population center distance criterion) may be necessary because of total integrated population dose consideration." 10 CFR 100.11)a)(3). Related to this was the indication in the Statement of Considerations that the Part was "intended to reflect past practice and current policy of the Commission of keeping stationary power and test reactors away from densely populated centers."

Thus a possible legal view would be that where such "very large cities" are involved, consideration of the "total integrated population dose" would be appropriate, and that this dose should include "non-credible accident" doses in light of the underlying purpose of the population center distance criterion to protect people in large centers from non-credible accidents. The Staff does not routinely include such consideration of "non-credible"

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*It is not clear from the Part how it was expected that this population center distance criterion, which was dependent upon calculated doses from a limiting "credible" accident, should in practice afford protection to people in large centers against "non-credible" accidents. Indeed in practice the population center distance criterion can easily be met even for large nuclear power reactors near very large population centers by addition of engineered safety features. The net result is that the Commission's regulations provide no effective quantitative limits on siting of nuclear power reactors near very densely populated sites.

accidents where large cities are involved.* Thus, an accident seems to be regarded as "credible" for some purposes and "non-credible" for others. Part 100 contains no definition of the term "credible". However, Commission precedent indicates that the term is intended as a measure of the probability of the accident in question, e.g., Long Island Lighting Company (Shoreham Nuclear Power Station), ALAB-156, RAI-73-10 at 846 (October 26, 1973).** The Commission itself has never specified the probability of occurrence which separates "credible" accidents from "non-credible" accidents, but the Commission's staff has taken the position that the figure 10-7 per reactor per year should generally be used for this purpose. (See WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Power Reactors," September, 1973.) However, in actual implementation of Part 100 the Staff does not routinely evaluate the consequences of certain accidents where the occurrence of the accident would indicate defective design of a system or component notwithstanding compliance that system or component with a Commission Regulation embodying a design requirement. In such cases the accident is dismissed as a "non-credible" accident without consideration of the probability of its occurrence.

One of the conclusions of WASH-1400 was that the probability of certain reactor accidents was higher than was previously assumed to be the case. In the past, the staff had considered that the probability of some of these accidents including core melt followed by containment failure was considerably lower--in the range of 10-6 per reactor year or lower. For this reason, core melt accidents followed by containment failures have not been regarded in the past as "credible" accidents within the meaning of Part 100. Thus the report raises a question whether these accidents should continue to be regarded as outside the plant design basis or "non-credible".

*However, since the promulgation of Part 100 in 1962, supplemental methodologies involving implicit and general consideration of "non-credible" accident consequences have always been used by the Commission in site suitability evaluations from the radiological safety standpoint. The methodology was evolved from the Commission's so-called "policy against metropolitan siting" in the early to late 1960's, to the population envelope" of Indian Point and Zion that was used in the late 1960's and early 1970's. More recently, the issue has become integrated with NEPA review of alternative sites.

** In one case prior to promulgation of Part 100, the Commission held that the possible consequences of an "extremely remote" accident that was "beyond any known accident possibility" should still not be ignored. Power Reactor Development Company, 1 AEC 65, 74 (December 10, 1958). Under this holding, it was unclear whether "conceivability" as opposed to "probability" was to be the determining factor in accident analyses.

Thus it can be argued that present Staff implementation of Part 100 has certain deficiencies--deficiencies associated with failure to consider "non-credible" accident consequences where very large cities are involved, and the failure to evaluate routinely certain core melt accidents.

Second, because of the limited range of possible accidents evaluated by the Staff under Part 100, the trend toward reducing "conservatisms" in both the definition of accident source terms and the calculation of accident consequences under Part 100, and the increasing reliance on engineered safety features to accomplish reductions in calculated accident doses, Part 100 no longer serves as an effective criterion for reactor site suitability from the standpoint of radiological accident risk. Indeed, as presently implemented by the Staff, Part 100 would permit location of a large nuclear power reactor in Central Park, New York City. Since Part 100 is the only written Commission regulation which addresses the subject of reactor site suitability criteria, the net effect is that site suitability is addressed by the Staff on a case-by-case judgment basis, using standard review plans, regulatory guides, etc.

In 1977 and 1978, the question of a potentially explicit role of Class 9 accidents in reactor siting received increasing emphasis within various sectors of the NRC, including the licensing Staff, the ACRS and the Commissioners themselves. The outcome of this study and evaluation cannot be predicted. However, a potential trend toward some explicit use of Class 9 events is indicated.

3. SOME ASPECTS OF THE REGULATORY PROCESSS

3.1 General Design Criteria

The Regulatory Staff began developing general criteria for nuclear power plants in the spring of 1965 under pressure from the Commissioners and with the expectation or knowledge that the AEC Regulatory Review Panel planned to recommend considerable emphasis on the development of criteria.

The first (rough) draft sent to the ACRS for information was entitled "Design and Operating Criteria for Nuclear Power Plants." On May 26, 1965 Mr. Price, the Director of Regulation, fowarded Draft IV dated May 20, 1965, which was entitled "Criteria for Nuclear Power Plant Construction Permits." It was explained by the Staff that design criteria were needed more urgently than criteria for the operating stage; hence, priority was being given to criteria for construction.

At the first subcommittee meeting on the subject, held June 5, 1965, Mr. Price indicated that the reasons for development of guides at that time were as follows:

(a) Continued pressure from the nuclear industry for better definition of the information required at the various stages of review. The Jersey Central/Niagara Mohawk Hearing Board decisions had emphasized the need for clarification.

(b) The desire of the Commission to clarify its position by a redefinition of the rules rather than a policy statement.

(c) The increased workload anticipated in the future which would require a set of standards against which proposals can be judged. In addition, they would help to identify important areas so that submission by applicants could be more to the point with a reduced amount of extraneous material which must be reviewed.

Actually, design criteria had been evolving over the previous several years. Typically, with each new construction permit review (and with the review of some difficult sites, such as Corral Canyon for the Los Angeles Department of Water and Power), new design criteria were introduced as the result of something specific to a site or reactor, from questions concerning possible low-probability events not previously considered, and from unusual operating experience which had generic implications.

A simple example of the development of a site-related criterion relates to tornadoes. When the General Electric Company proposed to build the

Southwest Experimental Fast Oxide Reactor (SEFOR) in Arkansas, the need to provide suitable protection against tornadoes came into focus. It was necessary to establish design basis tornadic wind speeds and to consider other possible effects such as tornado-induced missiles. As a result of this review and additional reviews of the matter in the ensuing months, it became clear that there was a "tornado belt" in the U.S. which included Arkansas. But for almost all regions east of the Rockies tornadoes occurred at a smaller frequency, but not radically so. Very destructive tornadoes had occurred in Massachusetts and other places far from the "tornado belt." Thus, tornado requirements were applied to all future power reactors east of the Rockies.*

Floods were taken into account in the reactors constructed in the early 1960's; however, such criteria became much more stringent with the evolution of the Probable Maximum Flood approach in the late sixties and early 1970's.

Hurricanes became one focus of the site-related phenomena for the Turkey Point site in Florida during its construction permit review in 1966. Afterwards, hurricanes began to receive increased emphasis for all East Coast and Gulf Coast sites; they were also considered as a possibly serious cause of inland flooding at many sites.

Seismic criteria evolved in a more complex way. There was little if any seismic requirement beyond the uniform building code for reactors east of the rockies in the 1950's. The matter received considerable attention for sites in California, of course, beginning with the site reviews for the Los Angeles Dept. of Water and Power for Southern California Edison, and progressed rapidly with San Onofre, Bodega Bay and Corral Canyon (Malibu).

In the Eastern U.S., seismicity was not given much emphasis, and, in fact, the proposed seismic design basis for the Connecticut Yankee reactor of 0.17g peak horizontal acceleration was thought by some to be unnecessarily high. We shall treat seismic matters in Chapter 5 in some detail, and hence not dwell on it here.

Engineered safety features also evolved considerably during the period 1960-65. Dresden, for example, a 1950's vintage reactor, had as many as six off-site power lines feeding the plant, but no on-site emergency AC power source. (In the 1960's a swath of tornadoes knocked out all lines concurrently, but no accident ensued). With the passage of time, first one small on-site diesel, then a larger diesel to run containment cooling systems, then redundant diesels to drive containment-related safeguards became the standard. And in 1966, redundant on-site power had to be available to power the ECCS, requiring still larger diesels (or their equivalent).

In any event, design criteria had been evolving and no effort was under way to generate an AEC Regulation. The general statement preceding "Draft

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*and later west of the Rockies.

IV" of the general criteria began

> Attached hereto are a number of proposed criteria for the design of a nuclear power facility which, upon the supplying by an applicant of adequate information to permit a conclusion that the criteria are fulfilled, would by definition justify a finding of reasonable assurance that the facility can be built and operated without undue risk to the health and safety of the public.

The ACRS had a number of problems with this philosophy. They generally felt the criteria, after they had been worked over and found acceptable, might represent necessary but not sufficient design requirements, and that there would be a continuing need for engineering judgment.

Reactors had been receiving construction permits based on a commitment to meet rather general criteria, but the criteria had been rapidly increasing in number and in detail. And, as the passing of time would soon show, the regulatory process was on the verge of shifting from the requirement of general criteria and the plausibility of their being met, to a continually increasing interest in specific design information prior to issuance of the construction permit.

At the Special ACRS meeting, June 18, 1965, the Committee had divided opinion as to the need for such criteria. Some believed the criteria would be of use to the Hearing Boards but not particularly to applicants. Others believed that the criteria were worthwhile. ACRS member Thompson thought that the time was too early for issuance of such design criteria. And at the next Subcommittee meeting, July 15, 1965, ACRS member Rogers was concerned that the existing list may be insufficient, and that the criteria should not be completed until four or five reactors situated near cities had been reviewed and analyzed.

Nevertheless, the Subcommittee, which was chaired by Dr. Leslie Silverman, worked actively on detailed formulation of criteria with the Regulatory Staff; a two day Subcommittee meeting was held the same month, July 21-22, 1965, to continue the review, discussion and revision.

At the July 15 Subcommittee meeting, Mr. Price was asked by ACRS member Okrent how the criteria handled the problem of different requirements for sites in the country, near to cities, and in cities. Mr. Price stated that the only item related to this matter was the amount of credit to be allowed for engineered safety features in reducing the calculated off-site doses (in the arbitrarily postulated MCA which assumes an intact containment).

The minutes of the July 21-22, 1965 Subcommittee meeting record considerable discussion on the criterion dealing with containment design basis. Dr. Doan, Director of the Division of Reactor Licensing, wated to define the MCA as the accident involving complete loss of coolant, followed by melt-down of the enitre core and the occurrence of a metal-water reaction. It was noted that in the Staff Analysis for San Onofre 1, the Staff gave credit for the ECCS preventing more than 6% of the core from melting. Nevertheless, Dr. Doan believed that reactors should be designed to prevent accidents but that the containment should be designed to withstand the worst consequence of any credible accident, should the other engineered safeguards fail.

At the end of the discussion it was decided to reword the particular criterion approximately as follows:

> Provisions must be made for the removal of heat from within the containment structure to maintain the structure within design limitations. The containment must be designed on the basis of complete depressurization of the primary system, melt-down uninhibited by core protective systems, and the occurrence of a metal-water or other chemical reaction. If engineered safeguards are needed to prevent containment vessel rupture due to heat release by an accident including these assumptions or by decay heat, at least two independent, differently designed systems which can accomplish the necessary functions must be provided. Each of these systems must be redundant in vital components so that no credible failure of a single component could prevent either of the systems from functioning properly.

An example of the kind of metamorphosis which took place on some of the proposed criteria is illustrated by the following:

In the early draft forwarded to the ACRS on May 26, 1965, there was a criterion which stated:

> The design goal for instrumentation and control systems including all electronic and mechanical devices, should be that all safety systems are fail-safe, including consideration of effects of fire, steam and other possible environments.

At the July 21-22, 1965 meeting, ACRS member Hanauer suggested the following wording, which was accepted as a basis to be used in rewriting the criterion:

> The vital instrumentation systems must be designed so that no credible combination of circumstances can interfere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (heat from loss of instrument cooling, extreme cold, fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

The Staff's early draft, General Design Criteria, reflected what appeared to be the then accepted practice of the Regulatory Staff. As the criteria went through hard review and evaluation in successive drafts, they were made, more and more, to reflect what the Regulatory Staff and the ACRS thought should be employed as general criteria for upcoming plants. The process of trying to write the criteria led to rethinking of the adequacy of what had been accepted on past plants.

On August 4, 1965 the Regulatory Staff issued draft IX of the Principal Design Criteria, and these were reviewed at the 65th ACRS meeting, August 5-7, 1965. The minutes of the Executive Session of the Committee reflect some of the mixed opinions among ACRS members.

REACTOR DESIGN CRITERIA

Executive Session

Dr. Silverman reported on three subcommittee meetings at which the drafts of the "Compilation of Principal Design Criteria for Nuclear Power Plant Construction Permits" were reviewed. To Dr. Silverman the preamble, or philosophy of the criteria, appears to merit the most attention from the ACRS now; this preamble might be considered general overall criteria. Although several considered the criteria premature and not in sufficient detail for guidance to the industry, it was recognized that the newcomers to the reactor field might benefit from these guides. Application of the criteria to the Dresden II Reactor as a trial was suggested by Dr. Okrent.

The evolution of the reactor site criteria was recounted by Dr. Thompson, who believed the proposed design criteria so general as to be of little use; specific requirements, e.g., engineering numbers, are not yet available. To him, the design guidance is too specific and may not be in the proper direction, e.g., no credit was given for active engineering safeguards such as core sprays; this could discourage the development of engineering safeguards. He also pointed out the illusion that these criteria are sufficient to insure safety. The checklist arrangement tried by the RS sometime ago was reviewed; its limited application may have resulted from the lack of sophistication of the applicant.

A need for reasonable criteria was seen by Dr. Kouts, who doubted if sufficient attention had been given to the guidance implicit in past RS and ACRS reports and conclusions. Much has probably been learned in the last few years from the criteria attempts. Dr. Hanauer was inclined against too much delay in facing this criteria issue. Several believed that AEC issuance of the criteria, formally or informally, was certain; hence contributions from the ACRS appeared desirable. Col. Stratton was disturbed over the imperative nature of the criteria, which seems to make them into a restrictive code. Most of the effort towards these criteria have probably been at a relatively high level of the RS; soliciting comments nearer the working level was suggested. To some the criteria appeared formulated around water reactors; however, nothing was included on metal water reactions.

Some conjectured that the design criteria attempt stems directly from the Commission and the Regulatory Review Panel pressure on Mr. Price and the RS; perhaps, the existance of criteria rather than their content, is important to the RS.

Dr. Kouts recommended modifying the present criteria, checking this formulation with action on previously approved cases, and finally trying the criteria on a current case.

ACRS discussion of the proposed criteria continued at a Subcommittee meeting held September 2, 1965. Excerpts from the meeting minutes follow:

Executive Session

Dr. Silverman reported that he had received several suggestions regarding revision of the introduction but no suggested changes to specific criteria.

Mr. Etherington suggested several tests for the criteria to try and identify their real purpose.

Test	Conclusion
1. Will they really help knowledgeable applicants/vendors.	Not much
2. Will they help less sophisticated applicants/vendors.	Probably yes - depends on applicants competence in nuclear matters
3. Will they form a framework for review and the eventual development of more detailed criteria.	Yes

In connection with 3 above, Mr. Etherington suggested that the AEC and ACRS should try to decide what each criteria really means in terms of plant design before they are promulgated. For example, what are the codes that are really acceptable under Criterion 1.a.

Mr. Rogers noted that the criteria are in part to tell the public what factors are considered in an AEC review and also to satisfy Mr. Price's desire for a legalized regulatory process where licenses can be issued with little engineering judgment.

Dr. Thompson maintained, however, that the criteria are premature at this time and may force the nuclear industry into a mold that is not necessarily optimum. He also maintained that several important criteria are not included. For example limits related to small local critical masses in large cores are not included. Other Subcommittee members strongly encouraged Dr. Thompson to identify these areas.

It was agreed by a majority of the members that approach number 3 should be pursued. Dr. Silverman suggested that a document similar to TID-14844 be developed to support the criteria. Dr. Thompson agreed that this would provide for a more thoughtful development of the criteria and noted that it might well result in several changes.

At the 67th meeting, October 7-9, 1965, the ACRS reviewed a comparison of the design criteria to the design bases for the upcoming Dresden 2 BWR, and then gave a favorable Committee opinion concerning the publication of the latest draft of the criteria in the Federal Register for public comment.

(Parenthetically, the minutes of the 67th meeting note that Dr. Swartout, the AEC Assistant General Manager for reactors, had been appointed Chairman of the AEC Steering Committee for Reactor Research, and that, broadly, the purpose of the Steering Committee is to arrive at designs for making reactors suitable for metropolitan locations).

At the 68th meeting, November 10-12, 1965, the ACRS again reviewed a revised draft, and on November 22, 1965, the Atomic Energy Commission issued a press release announcing the proposed criteria and requesting public comment.

The complete criteria, as first published, are reproduced on the following pages.

The criteria were left flexible, quite deliberately, thereby allowing many individuals holding a wide range of opinion, to agree on them. The same flexibility was expected to be necessary in their application.

There was continuing discussion on a proposed "backup" document, which was to explain and elaborate on the published draft criteria. The act of trying to write a more specific explanation provided benefits, frequently in the form of new matters to be studied. An example is available in the minutes of the January 26, 1966 Subcommittee meeting. An ACRS member questioned the extent of the review which had been made in the past to consider the effects of a fire on the control system. The minutes go on, "The review has not been extensive; however, some consideration has been given to use of separate cable runs, redundant channels in separate cabinets, etc." Another member noted that vital buses often were not redundant, and that a requirement for duplication should be added.

AEC
UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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FOR IMMEDIATE RELEASE
(Monday, November 22, 1965)

AEC SEEKING PUBLIC COMMENT ON PROPOSED DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The Atomic Energy Commission is seeking comment from the nuclear industry and other interested persons on proposed general design criteria which have been developed to assist in the evaluation of applications for nuclear power plant construction permits.

The proposed criteria have been developed by the AEC regulatory staff and discussed with the Commission's Advisory Committee on Reactor Safeguards (ACRS). They represent an effort to set forth design and performance criteria for reactor systems, components and structures which have evolved over the years in licensing of nuclear power plants by the AEC. As such, they reflect the predominating experience to date with water reactors but most of them are generally applicable to other reactors as well.

It is recognized that further efforts by the AEC regulatory staff and the ACRS will be necessary to fully develop these criteria. However, the criteria as now proposed are sufficiently advanced to submit for public comment. Also, they are intended to give interim guidance to applicants and reactor equipment manufacturers.

The development and publication of criteria for nuclear power plants was one of the key recommendations of the special Regulatory Review Panel which studied ways of streamlining the Commission's reactor licensing procedures.

In the further development of these criteria, the AEC intends to hold discussions with organizations in the nuclear industry and to issue from time to time explanatory information on each criterion. Following such discussions with industry and receipt of other public comment, the AEC expects to develop and publish criteria that will serve as a basis for evaluation of applications for nuclear power plant construction permits.

(more)

It is recognized that additional criteria may also be needed, particularly for reactors other than water reactors, and that there may be instances where one or more of the presently proposed criteria may not be applicable. Application of the criteria to a specific design continues to involve a considerable amount of engineering judgment.

These proposed criteria are part of a longer-range Commission program to develop criteria, standards and codes for nuclear reactors, including identification of codes and standards that industry will be encouraged to undertake. The ultimate goal is the evolution of industry codes based on accumulated knowledge and experience, as has occurred in various fields of engineering and construction.

A copy of the proposed "General Design Criteria for Nuclear Power Plant Construction Permits" is attached. Comments should be sent to the Director of Regulation, U. S. Atomic Energy Commission, Washington, D. C. 20545, by February 15, 1966.

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11/22/65

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Attached hereto are general design criteria used by the AEC in judging whether a proposed nuclear power facility can be built and operated without undue risk to the health and safety of the public. They represent design and performance criteria for reactor systems, components and structures which have evolved over the years in licensing of nuclear power plants by the AEC. As such they reflect the predominating experience to date with water reactors but most of them are generally applicable to other reactors as well.

It should be recognized that additional criteria will be needed for evaluation of a detailed design, particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Moreover, there may be instances in which it can be demonstrated that one or more of the criteria need not be fulfilled. It should also be recognized that the application of these criteria to a specific design involves a considerable amount of engineering judgment.

An applicant for a construction permit should present a design approach together with data and analysis sufficient to give assurance that the design can reasonably be expected to fulfill the criteria.

FACILITY

CRITERION 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for nonnuclear applications may not be adequate.

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, and other natural phenomena anticipated at the proposed site.

CRITERION 2

Provisions must be included to limit the extent and the consequences of credible chemical reactions that could cause or materially augment the release of significant amounts of fission products from the facility.

CRITERION 3

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

REACTOR

CRITERION 4

The reactor must be designed to accommodate, without fuel failure or primary system damage, deviations from steady state norm that might be occasioned by abnormal yet anticipated transient events such as tripping of the turbine-generator and loss of power to the reactor recirculation system pumps.

CRITERION 5

The reactor must be designed so that power or process variable oscillations or transients that could cause fuel failure or primary system damage are not possible or can be readily suppressed.

CRITERION 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

CRITERION 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

CRITERION 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

CRITERION 9

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating condition.

CRITERION 10

Heat removal systems must be provided which are capable of accommodating core decay heat under all anticipated abnormal and credible accident conditions, such as isolation from the main condenser and complete or partial loss of primary coolant from the reactor.

CRITERION 11

Components of the primary coolant and containment systems must be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless the temperatures are well above the nil-ductility temperatures. For ferritic materials of the coolant envelope and the containment, minimum temperatures are NDT + 60°F and NDT + 30°F, respectively.

CRITERION 12

Capability for control rod insertion under abnormal conditions must be provided.

CRITERION 13

The reactor facility must be provided with a control room from which all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times. The control room must be provided with adequate protection to permit occupancy under the conditions described in Criterion 17 below, and with the means to shut down the plant and maintain it in a safe condition if such accident were to be experienced.

CRITERION 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

CRITERION 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no component or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

CRITERION 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can intefere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (heat from loss of instrument cooling, extreme cold, fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

ENGINEERED SAFEGUARDS

CRITERION 17

The containment structure, including access openings and penetrations, must be designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with the largest credible energy release including the effects of credible metal-water or other chemical reactions uninhibited by active quenching systems. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards must be provided for that part if necessary, to protect the health and safety of the public, in case of an accidental rupture in that part of the system. The appropriateness of safeguards such as isolation valves, additional containment, etc., will depend on environmental and population conditions surrounding the site.

CRITERION 18

Provisions must be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under the conditions described in Criterion 17 above. If engineered safeguards are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems must be provided, preferably of different principles. Backup equipment (e. g., water and power systems) to such engineered safeguards must also be redundant.

CRITERION 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

CRITERION 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leak-tightness can be demonstrated at design pressure at any time throughout operating life of the reactor.

CRITERION 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

CRITERION 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent of each other. Capability must be provided for testing functional operability of these valves and associated equipment to determine that no failure has occurred and that leakage is within acceptable limits. Redundant valves and auxiliaries must be independent. Containment closure valves must be actuated by instrumentation, control circuits and energy sources which satisfy Criterion 15 and 16 above.

CRITERION 23

In determining the suitability of a facility for a proposed site the acceptance of the inherent and engineered safety afforded by the systems, materials and components, and the associated engineered safeguards built into the facility, will depend on their demonstrated performance capability and reliability and the extent to which the operability of such systems, materials, components, and engineered safeguards can be tested and inspected during the life of the plant.

RADIOACTIVITY CONTROL

CRITERION 24

All fuel storage and waste handling systems must be contained if necessary to prevent the accidental release of radioactivity in amounts which could affect the health and safety of the public.

CRITERION 25

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

CRITERION 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate hold-up capacity must be provided for retention of gaseous, liquid, or solid effluents.

CRITERION 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

The AEC received a considerable number of responses to its request for comment on the proposed criteria, largely from representatives of the nuclear industry and from members of national laboratories. Looking back, more than ten years later, we excerpt a few comments, in random order.

Wm. Cottrell (ORNL): "The ramifications of sabotage was evidently not considered when preparing the criteria. Should they be? Criteria 2 and 3 address themselves to two specific situations namely, chemical reactions and missiles, respectively. Although it is obvious that these two situations are of concern, other situations of comparable concern have apparently been omitted. These might include fires, floods, major electrical bus failures, recoil forces from water hammer, pressure vessel failure, etc."

Peter Morris (AEC): The repeated use of the terms 'anticipated' and 'credible' yield a document of minimal value to design engineers. Criteria, to be useful, should specify what contigencies <u>are</u> 'credible' or 'to be anticipated' or, lacking that, should at least say to <u>whom</u> these contingencies must appear credible.

R. J. Rickert (Combustion Engineering): Part (b) of Criterion 1 should consider the past history of the site for design purposes and not attempt to 'anticipate' future natural phenomena.

R. L. Junkins (Battelle Northwest): Part (b) of Criterion 1 is deficient in that while no one is likely to disagree with the principle expressed, the criterion does nothing to either provide a standard for judging, or at least some guidance as to determining what is meant by 'most severe natural phenomena anticipated'. It is suggested that an arbitrary judgment be exercised, i.e. the loadings of the Uniform Building Code be increased by some minimum factor; say 1.5, for example.

D. L. Crook (U.S. Dept. of Commerce): Criterion 1 (a). It would be interesting to see some examples of what is meant by 'commonly used design codes are not adequate'. Our experience is that most existing codes are adequate. Any inadequacy more likely lies in the application and enforcement of the code. We feel that nuclear power should not be set up as something infinitely more dangerous than other industrial processes, and that it should be determined that existing codes are inadequate before new codes are inaugurated.

R. J. Rickert: As it stands, criterion 2 leaves to the applicant the decision of what is a 'credible' reaction, and how much will 'materially' augment the release of 'significant' amounts of fission products.

> J. B. McCarty (U.S. Coast Guard): Criterion 13 - No mention is made of fire, which we believe is a most serious hazard. We would recommend certain minimums in structural fire protection and in fire fighting equipment be specified.
>
> E. P. Epler (ORNL): Criterion 15 - This very excellent criterion does not include two very vital points: a) We have found that failure to identify all portions of a protective system has led to failure. b) Separation of safety and control was omitted from Criterion 15.
>
> C. Starr (Atomics International): Criterion 17 - This criterion is the most important one in assuming protection of the public health and safety. It might be pointed out that the significant problem is overpressurization rather than overheating.
>
> Stone and Webster: Criterion 17 - We take issue with the premise that engineered safeguards are not or cannot be given credit for limiting the credible energy release.
>
> We take issue with the premise that two independent cooling systems of different principles are preferable to two similar systems selected on the basis of greatest reliability and effectiveness. We, therefore, suggest that the words 'preferably of different principles' be deleted.

Following receipt of the public comments on the proposed design criteria of November 22, 1965, considerable effort on redrafting the criteria ensued by the Regulatory Staff, with frequent interaction with the ACRS.

Other events also transpired. On November 24, 1965, two days after issuance of the draft criterion, the ACRS sent a letter report to Chairman Seaborg of the AEC concerning pressure vessels, which initiated major efforts by the AEC and the nuclear industry on improved pressure vessel design, fabrication, and in-service inspection. And, in the summer of 1966, the "China Syndrome" question arose during the construction permit review of Dresden 3 and Indian Point 2, leading to ACRS recommendations for major new efforts on improved primary system integrity and major emphasis on greatly improved ECCS, all as part of the recognition that containment was not an independent barrier.

Hence, there were several major changes in the revised set of proposed General Design Criteria issued for comment on July 10, 1967, which are reproduced on the following pages.

That the original 27 criteria had grown to seventy, of itself, was not especially significant, since some separation of the original criteria took place. However, there now was a separate cluster of criteria on ECCS (criteria 45-48). There were a large number related to primary system integirty (5, 9, 33-36). The containment design basis (49) did not explicitly include coping with full melt down of the core; however, there is a vague term - "including a considerable margin for effects

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FOR IMMEDIATE RELEASE
(Monday, July 10, 1967)

AEC PUBLISHES GENERAL DESIGN CRITERIA
FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The AEC is publishing for public comment a revised set of proposed General Design Criteria which have been developed to assist in the preparation of applications for nuclear power plant construction permits.

In November 1965, the AEC issued an announcement requesting comments on General Design Criteria developed by its regulatory staff. These criteria were statements of design principles and objectives which have evolved over the years in licensing nuclear power plants by the AEC.

It was recognized at the time the criteria were first issued for comment that further efforts were needed to develop them more fully. The revision being published today reflects extensive public comments received from twenty groups or individuals, suggestions made at meetings with the Atomic Industrial Forum, and review within the AEC.

The regulatory staff has worked closely with the Commission's Advisory Committee on Reactor Safeguards on the development of the criteria and the revision of the proposed criteria reflects ACRS review and comment.

The General Design Criteria reflect the predominating experience to date with water reactors, but they are considered to be generally applicable to all power reactors. The proposed criteria are intended to be used as guidance to an applicant in establishing the principal design criteria for a nuclear power plant. The framework within which the criteria are presented provides sufficient flexibility to permit applicants to establish design requirements using alternate and/or additional criteria. In particular, additional criteria will be needed for unusual sites and environmental conditions and for new or

(more)

advanced types of reactors. In each case an applicant will be required to identify its principal design criteria and provide assurance that they encompass all those facility design features required in the interest of public health and safety.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information has been needed at the construction permit stage for certain of the criteria; these have been designated as Category A.

Development of these criteria is part of a longer-range Commission program to develop criteria, standards, and codes for nuclear reactor plants. This includes codes and standards that industry is developing with AEC participation. The ultimate goal is the evolution of industry codes and standards based on accumulated knowledge and experience as has occurred in various fields of engineering and construction.

The provisions of the proposed amendment relating to General Design Criteria are expected to be useful as interim guidance until such time as the Commission takes further action on them.

The proposed criteria, which would become Appendix A to Part 50 of the AEC's regulations, will be published in the Federal Register on July 11, 1967. Interested persons may submit written comments or suggestions to the Secretary, U. S. Atomic Energy Commission, Washington, D. C. 20545, within 60 days. A copy of the proposed "General Design Criteria for Nuclear Power Plant Construction Permits" is attached.

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7/10/67

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

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INTRODUCTION

Every applicant for a construction permit is required by the provisions of §50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors.

Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Criteria should be justified.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for those in Category B.

I. OVERALL PLANT REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

CRITERION 3 - FIRE PROTECTION (Category A)

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

CRITERION 5 - RECORDS REQUIREMENTS (Category A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 6 - REACTOR CORE DESIGN (Category A)

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

CRITERION 10 - CONTAINMENT (Category A)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

III. NUCLEAR AND RADIATION CONTROLS

CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (Category B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

IV. RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

Multiple failures resulting from a single event shall be treated as a single failure.

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

V. REACTIVITY CONTROL

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

At least two independent reactivity control systems, preferably of different principles, shall be provided.

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (Category B)

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

VI. REACTOR COOLANT PRESSURE BOUNDARY

CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

VII. ENGINEERED SAFETY FEATURES

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (Category A)

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

CRITERION 40 - MISSILE PROTECTION (Category A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (Category A)

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

CRITERION 49 - CONTAINMENT DESIGN BASIS (Category A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (Category A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (Category A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems."

Criterion 44 calls for "at least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling."

The single failure criterion, which had appeared earlier in slightly different wording, was prominent in the July, 1977 proposed General Design Crieria.

The press release stated that the design criteria were intended to provide flexibility, and, with few exceptions, they did. Hence, they still served primarily to call out the general safety areas which needed to be addressed. They did not provide a quantitative safety goal to be reached, or some other quantitative basis for establishing the adequacy of any particular specific design. Hence, a commitment by an applicant to meet the General Design Criteria provided no basis for assessing the safety level he would seek to provide. Nor did the criteria, for the most part, establish the safety level that the Regulatory Staff required in order to approve construction and operation of a reactor. Not that this was good or bad. Quite consciously, the General Design Criteria left most matters up to "engineering judgment."

Except for the qualitative reference to a margin for chemical reactions in containment design (criterion 49), the General Design Criteria make no reference to problems arising from core melt or to methods of coping with or ameliorating the consequences of core melt.

The "proposed" criteria of July 10, 1967 provided "interim guidance" to the Regulatory Staff and the nuclear industry for several years, even though they were not formally adopted as an AEC Regulation.

On February 20, 1971 the AEC published a revised set of General Design Criteria which became Appendix A to Part 50 of the AEC's regulations 90 days thereafter. The press release and criteria are on the following pages.

The 1971 criteria, now 64 in number, were generally drafted to conform with the design features of LWR plants receiving construction permits during the previous few years. They still remained general in nature. The word "appropriate" is used very often, e.g., "Appropriate consideration of the most severe of the natural phenomena that have been historically reported" and "Fire detection and fighting systems of appropriate capacity and capability shall be provided," etc.).

Under Criterion 35, no longer are two ECCS's, each capable of providing abundant cooling called for; rather, the criterion now says " A system to provide abundant emergency core cooling shall be provided," and further on, the single failure criterion is imposed. This represented a weakening of the equivalent criterion (No. 44) in the 1967 version; it represented an acceptance of what was the actual situation on the reactors which were being approved.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

No. O-23
Tel. 973-3446 (Info.)
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FOR IMMEDIATE RELEASE

AEC PUBLISHES GENERAL DESIGN CRITERIA
FOR NUCLEAR POWER PLANTS

The AEC is publishing a revised set of general design criteria for use in establishing the principal design criteria for nuclear power plants.

In July 1967 AEC published in the Federal Register for public comment "General Design Criteria for Nuclear Power Plant Construction Permits" developed by its regulatory staff. The revision published today reflects extensive comment received from 21 groups or individuals, review within the AEC, and developments that have occurred in the nuclear industry since publication of the criteria in 1967.

The regulatory staff has worked closely with the Commission's Advisory Committee on Reactor Safeguards in developing the revised criteria.

The amendment to Part 50 of the Commission's regulations fixes minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units previously approved by the Commission for construction. It provides guidance, also, for establishing the principal design criteria for other types of nuclear power plants. Additional or different criteria are expected to be needed for unusual sites and environmental conditions, and for nuclear power plants of advanced design.

Development of these criteria is part of a longer range Commission program to develop criteria, codes, and standards applicable to nuclear power plants. This includes criteria, codes, and standards that industry is developing with AEC participation. The ultimate goal is the evolution of industry criteria, codes, and standards based on accumulated knowledge and experience in various fields of engineering and industry.

(more)

The criteria will become Appendix A to Part 50 of AEC's regulations 90 days after being published in the Federal Register on February 20, 1971. Interested persons may submit comments to the Secretary of the Commission, U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Chief, Public Proceedings Branch, within 45 days. The comments will be given consideration with the view to possible further amendments. A copy of the proposed "General Design Criteria for Nuclear Power Plants" may be obtained by writing to the Director, Division of Reactor Standards, U. S. Atomic Energy Commission, Washington, D. C. 20545.

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February 19, 1971

TITLE 10 - ATOMIC ENERGY

CHAPTER 1 - ATOMIC ENERGY COMMISSION

PART 50 - LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plants

The Atomic Energy Commission has adopted an amendment to its regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which adds an Appendix A, "General Design Criteria for Nuclear Power Plants."

Paragraph 50.34(a) of Part 50 requires that each application for a construction permit include the preliminary design of the facility. The following information is specified for inclusion as part of the preliminary design of the facility:

(i) The principal design criteria for the facility

(ii) The design bases and the relation of the design bases to the principal design criteria

(iii) Information relative to materials of construction, general arrangement, and the approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

The "General Design Criteria for Nuclear Power Plants" added as Appendix A to Part 50 establish the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. They also provide guidance in establishing the principal design criteria for other types of nuclear power plants. Principal design criteria established by an applicant and accepted by the Commission will be incorporated by reference in the construction permit. In considering the issuance of an operating license under Part 50, the Commission will require assurance that these criteria have been satisfied in the detailed design and construction of the facility and that any changes in such criteria are justified.

A proposed Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" to 10 CFR Part 50 was published in the FEDERAL REGISTER (32 FR 10213) on July 11, 1967. The comments and suggestions received in response to the notice of proposed rule making and subsequent developments in the technology and in the licensing process have been considered in developing the revised criteria which follow.

The revised criteria establish minimum requirements for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission, whereas the previously proposed criteria would have provided guidance for applicants for construction permits for all types of nuclear power plants. The revised criteria have been reduced to

55 in number, include definitions of important terms, and have been rearranged to increase their usefulness in the licensing process. Additional criteria describing specific requirements on matters covered in more general terms in the previously proposed criteria have been added to the criteria. The Categories A and B used to characterize the amount of information needed in Safety Analysis Reports concerning each criterion have been deleted since additional guidance on the amount and detail of information required to be submitted by applicants for facility licenses at the construction permit stage is now included in § 50.34 of Part 50. The term "engineered safety features" has been eliminated from the revised criteria and the requirements for "engineered safety features" incorporated in the criteria for individual systems.

Further revisions of these General Design Criteria are to be expected. In the course of the development of the revised criteria, important safety considerations were identified, but specific requirements related to some of these considerations have not as yet been sufficiently developed and uniformly applied in the licensing process to warrant their inclusion in the criteria at this time. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(i) Consideration of the need to design against single failures of passive components in fluid systems important to safety.

(ii) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem and the required interconnection and independence of the subsystems have not yet been developed or defined.

(iii) Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

(iv) Consideration of the possibility of systematic, non-random, concurrent failures of redundant elements in the design of the protection systems and reactivity control systems.

In addition, the Commission is giving consideration to the need for development of criteria relating to protection against industrial sabotage and protection against common mode failures in systems, other than the protection and reactivity control systems, that are important to safety and have extremely high reliability requirements.

It is expected that these criteria will be augmented or changed when specific requirements related to these and other considerations are suitably identified and developed.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification to be effective 90 days after publication in the FEDERAL REGISTER. The Commission invites all interested persons who desire to submit written comments or suggestions in connection with the amendment to send them to the Secretary, U. S. Atomic Energy Commission, Washington, D. C., 20545, Attention: Chief, Public Proceedings Branch, within 45 days after publication of this notice in the FEDERAL REGISTER. Such submissions will be given consideration with the view to possible further amendments. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C.

1. Subdivision 50.34(a)(3)(i) is amended to read as follows:

§ 50.34 Contents of applications; technical information.

(a) Preliminary safety analysis report. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility. Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units:

* * * * *

2. Footnote[2] to § 50.34 is amended to read as follows:

   [2]General design criteria for chemical processing facilities are being developed.

* * * * *

3. A new Appendix A is added to read as follows:

APPENDIX A

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

IV. FLUID SYSTEMS

V. REACTOR CONTAINMENT

VI. FUEL AND RADIOACTIVITY CONTROL

INTRODUCTION

Pursuant to the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26 and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

NUCLEAR POWER UNIT

A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

LOSS OF COOLANT ACCIDENTS

Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

SINGLE FAILURE

A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2]

ANTICIPATED OPERATIONAL OCCURRENCES

Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited

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[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] Single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERIA

I. OVERALL REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 10 - REACTOR DESIGN

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

CRITERION 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

CRITERION 13 - INSTRUMENTATION AND CONTROL

Instrumentation and control shall be provided to monitor variables and systems over their anticipated range for normal operation and accident conditions, and to maintain them within prescribed operating ranges, including those variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

CRITERION 17 - ELECTRICAL POWER SYSTEMS

An onsite electrical power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electrical power sources, including the batteries, and the onsite electrical distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights of way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. Two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power sources and the other offsite electrical power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electrical power from any of the remaining sources as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electrical power sources.

CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

Electrical power systems important to safety shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

CRITERION 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. PROTECTION AND REACTIVITY CONTROL SYSTEMS

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems of different design principles and preferably including a positive mechanical means for inserting control rods, shall be provided. Each system shall have the capability to control the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operations, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. FLUID SYSTEMS

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

CRITERION 33 - REACTOR COOLANT MAKEUP

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

CRITERION 34 - RESIDUAL HEAT REMOVAL

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

CRITERION 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

CRITERION 44 - COOLING WATER

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. REACTOR CONTAINMENT

CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit (1) inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment. or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment. or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment. or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment. or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. FUEL AND RADIOACTIVITY CONTROL

CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

<u>CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING</u>

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

<u>CRITERION 63 - MONITORING FUEL AND WASTE STORAGE</u>

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

<u>CRITERION 64 - MONITORING RADIOACTIVITY RELEASES</u>

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

(Secs. 161, 182, 68 Stat. 948, 953; 42 U.S.C. 2201, 2232.)

Dated at Washington, D. C. this 10th day of February 1971.

FOR THE ATOMIC ENERGY COMMISSION

W. B. McCool
Secretary of the Commission

None of the criteria related to accidents involving large scale core melt.

A good example of the detailed evolution of a specific general design criterion (GDC) is provided by GDC 17 "Electrical Power Systems."

A memorandum prepared by ACRS Staff Engineer, M. C. Gaske, dated January 28, 1972, gives a brief summary of the chronology of the development of GDC 17 requirements relative to separate incoming transmission lines. The summary follows:

GENERAL DESIGN CRITERION 17
CHRONOLOGY REGARDING SEPARATE INCOMING TRANSMISSION LINES

July 11, 1967 - AEC published the proposed General Design Criteria for public comment. Criterion 39 (later numbered 17) stated:

"Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system."

April 24, 1968 - ACRS and Regulatory Staff representatives met. It was reported that the original intent of Criterion 39 was to require two independent paths but that recent interpretations have qualified this intent to permit one path to be available 30 seconds, 1 hour, or 8 hours after shutdown of the plant.

March 28, 1969 - New Draft of the General Design Criteria made available.

May 26, 1969 - At the May 26, 1969 meeting, the above draft was discussed. DRS indicated that there was a question as to whether the ACRS wished to require two physically independent transmission lines for off-site power.

June 5-7, 1969 - It was agreed by the Committee that the Electrical Systems, Control and Instrumentation Subcommittee would assist in review of Criterion 17.

August 7-9, 1969 - The ACRS considered a July 15, 1969 draft of Criterion 17 which stated in part "Offsite electrical power shall be provided to the site preferably by two physically independent transmission lines". The Committee discussed deletion of the word "preferably" and decided not to propose such a revision.

October 9-11, 1969-At the full Committee meeting, Mr. Case presented the majority and minority Staff positions regarding the Criterion 17 requirement regarding offsite power "... preferably by two independent transmission lines". The majority of the Regulatory Staff believed that one offsite transmission line was sufficient. (This was apparently based primarily on a Staff study which indicated that incoming power lines are two orders of magnitude more reliable than the offsite power grid.) Mr. Case stated that a change in Criterion 17 was planned to require at least one separate incoming line at each facility in addition to the line exporting power. The Committee decided to refer the matter of Criterion 17 to Subcommittee for further review.

November 5, 1969 - The Codes, Standards, and Criteria Subcommittee met and made a number of suggestions regarding Criterion 17.

November 6-8, 1969-The Committee considered, and approved, except for a minor comment, a draft Criterion 17 that stated in part:

"Two physically independent transmission lines, each with the capability of supplying electrical power from the transmission network to the switchyard and two physically independent circuits from the switchyard to the onsite electrical distribution system, shall be provided. Each of these circuits shall be available in sufficient time following a loss of electrical power from all other alternating current sources including onsite electrical sources, to assure that specified acceptable fuel damage limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be available immediately following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained."

October 27, 1970 - The Regulatory Staff provided a new draft of the General Design Criteria, based on industry comments. This version included a number of changes in Criterion 17 including rewording to require "... two transmission lines designed and located so as to suitably minimize the likelihood of their simultaneous failure ..." The criteria were referred to the Codes, Standards, and Criteria Subcommittee for review.

December 4, 1970 - At the ensuing Subcommittee meeting to review the criteria, Mr. Etherington evidenced concern regarding the erosion of the requirements that had occurred relative to Criterion 17.

December 10-12, 1970 - The Committee recommended, and the Regulatory Staff agreed, that Criterion 17 be modified to require two physically independent transmission lines.

December 21, 1970 -As a result of the Subcommittee and Committee comments, Criterion 17 was revised to state in part:

"Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights of way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. Two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power sources and the other offsite electrical power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained."

February 19, 1971 -The Commission adopted the General Design Criteria as Appendix A to Part 50.

July 15, 1971 -Appendix A to Part 50 was reissued with Criterion 17 essentially as stated on February 19, 1971.

Actually, discussion of the reliability of off-site power supplies (and AC power in general) remained a continuing matter of discussion. At the 144th meeting, April 6-8, 1972, ACRS Member Palladino expressed concern that GDC 17 did not clearly substantiate the bases for non-redundance of the connecting lines (e.g., the lines could be run on the same right-of-way). On the other hand, an ACRS subcommittee at a meeting on April 5, 1972 had estimated that the probability of failure of the entire off-site grid was at least one order of magnitude greater than that of the lines leading into the plant. Hence, improvment of these lines appeared to yield only marginal benefit.

The matter of actual versus required relaibility of power supplies remained an issue, open or just below the surface, in the years that followed, until in 1977 the ACRS requested that the Regulatory Staff formally evaluate the probability of loss of AC power as a function of length of loss, and examine the adequacy of systems that must work to keep the core acceptably cool in the absence of AC power.

One other interesting sidelight of the discussion on GDC 17 is that it represented one of the few times (prior to 1977) when opposing viewpoints from within the Regulatory Staff were formally presented to the ACRS. The brief minutes of the discussion are reproduced below:

> <u>Meeting with the Division of Reactor Standards</u>
>
> <u>General Design Criteria - Criterion 17 (GDC-17)</u>
>
> Mr. Case presented the majority and minority Staff positions on the GDC-17 statement regarding off-site power requirements, "... preferably by two physically independent transmission lines." DRS has determined that most power stations have two transmission lines, but it was not clear whether these were importing or exporting power. An examination of the reliability gained by having two versus one off-site transmission line results in a reliability improvement factor of two. However, the unreliability of the off-site power grid is 10^{-4}/time; the unreliability of the incoming power lines is 10^{-6}/time; therefore, the majority of the Staff believes that one off-site transmission line is sufficient.
>
> Mr. Levine, speaking as the minority, stated that he believes the loss of all AC power for several hours can result in worse core melting than could occur from a (design basis accident) DBA. He questioned the on-site AC power capability. His survey reveals that 90% of the 51 nuclear power plants have more than two off-site power lines, and 70% have more than two rights-of-way. He mentioned, as examples, that Commonwealth Edison and Philadelphia Electric believe that there should be two transmission lines, two rights-of-way, and two start-up transformers. He believes that, if the off-site power reliability of 10^{-4} obtained by the Staff is valid, this is unacceptable to the utilities.
>
> Mr. Case noted that a change in Criterion 17 is planned to require at least one, separate <u>incoming</u> power line at each facility in addition to the line that is exporting power.

It is somewhat ironical that the majority viewpoint used probabilistic arguments to indicate that there was only a slight gain in reliability from the presence of an additional line, while the minority view, in favor of at least two incoming lines, was presented by Saul Levine, later to become Staff Director of the Reactor Safety Study, which was based on probabilistic techniques, including common cause failures.

An example of the problems in going from the General Design Criteria to specific implementation is available from Criterion 3 - Fire Protection. All of the reactors approved in the few years preceeding 1971, and for a few years after 1971, were evaluated by the Regulatory Staff as having met this criterion. However, when the Browns Ferry Fire occurred in 1975, a full-blown technical review of the adequacy of protection against fire ensued, and major, specific modifications were required, even on existing plants, without any change in the wording of Criterion 3.

The Statement of Considerations (or introduction) to the 1971 criteria calls out several safetgy considerations for which specific requirements "have not as yet been sufficiently developed and uniformly applied in the licensing process to warrant their inclusion in the criteria at this time."

Most of that list, including all those related to redundancy, common mode failures, systematic, non-random failures and passive failures have remained open issues for the ensuring years. In 1977, three years after publication of the Reactor Safety Study (WASH-1400), the Regulatory Staff did acknowledge a need to review the single failure criterion. (E. Case, 1977)

The development of detailed regulatory positions with regard to the matters covered in the General Design Criteria (and other safety matters) began rather actively around 1967-68. There was a considerable, continuing growth in size of the Regulatory Staff, and with this came increased breadth and depth of knowledge in specific technical areas. The less-than-satisfactory experience with operating license reviews for reactors where only general design information had been available at the construction permit stage, and the difficulty in implementing major design changes in an already constructed plant, automatically led to growing emphasis on getting more design detail at the construction permit stage.

Since a process of promulgating new AEC Regulations covering specific design criteria was expected to be slow and cumbersome, a more flexible approach was sought for the documentation of specific approaches acceptable to the Regulatory Staff (and the ACRS). The Regulatory Staff began to develop internal documents which specified acceptably detailed design approaches to specific problems. In 1970 the Regulatory Guide approach to public documentation was initiated with the issuance of the first such guide (which dealt with the requirements for net positive suction head for ECCS pumps, and arose as a way of implementation of a philosophic safety concern that the function of the ECCS not be compromised because of a loss in containment integrity which, were it to occur, could change a relatively moderate accident to one having severe effects).

The Regulatory Guide approach has flourished so that by 1977 more than a hundred such guides had been published.

3.2 PRESSURE VESSELS: INSPECTABILITY, INSPECTION AND INTEGRITY

As was discussed in Chapter 2, the ACRS report of November 24, 1965 on Pressure Vessels produced sharp verbal reactions from the nuclear industry and the AEC. With the next few cases, Brookwood, Millstone Point and Indian Point 2, the ACRS established a pattern of dealing with the question of "coping with failure" which has remained essentially unchanged in the U.S. More specifically, for the more rural sites, the ACRS accepted the reactors without measures to deal with vessel failure, while for Newbold Island, which was somewhat more populated than Indian Point 2, the ACRS asked for protection against pressure vessel failure of limited extent.

To some extent, the advent of the "China Syndrome" matter, with the recognition of the direct correlation between core melt (for any reason) and a loss of containment integrity, lessened emphasis on pressure vessel failure as possibly the most significant source of a large reactor accident. Nevertheless, the ACRS remained very interested in improving the reliability of pressure vessels.

Since neither the nuclear industry nor the ASME code committee responded with great rapidity to oral urging by the Committee for more stringent requirements during fabrication, and for a program of inservice inspection, the ACRS decided to initiate a program within the AEC Regulatory Groups.

A review of the minutes for the latter part of 1966 and early 1967 provides a record of the way in which the ACRS proceeded in this matter. For example, a brief excerpt from the Special Meeting, December 2-3, 1966 records the initiation of the joint ACRS-AEC effort.

> <u>Special Meeting - December 2-3, 1966</u>
>
> Dr. Okrent went on to say that the Committee's present approach of asking individual applicants what would be done in excess of code requirements no longer seemed fruitful. He suggested that the industry as a whole be asked what code changes were considered necessary. In addition, he suggested asking Mr. Shaw and/or the Naval Reactors Division for their opinion on desirable code changes. He also felt that the Committee could take the position that the present code requirements are not sufficient since fabricators universally exceed them. A set of requirements acceptable to the Committee is necessary.
>
> Mr. Mangelsdorf asked if there were techniques available which were not being routinely applied.
>
> Dr. Hanauer suggested that ultrasonic inspection of welds is such a technique and is favored by some companies and not by others.

Mr. Palladino suggested that for the immediate future the Committee might have to make some arbitrary decisions.

Dr. Okrent proposed that the Committee suggest to Mr. Price and Mr. Shaw that the Committee would like to develop a set of acceptable standards for pressure vessel fabrication within three or four months. The ORNL study has been going on for about a year under the auspices of DRD&T so that background material should be readily available.

Dr. Hendrie commented that considerable work has already been done for the ACRS in formulating such standards. He suggested having Dr. Bush discuss his recommendations with personnel in DRD&T and the Regulatory Staff and produce ACRS criteria in that way. He felt that if a group is assembled including vendors and fabricators; it would not produce requirements for anything in excess of Section III of the ASME Code.

The following motion was passed:

The Chairman shall ask Mr. Price and Mr. Shaw whether members of their groups can and will work with one or more ACRS Members to develop additional requirements for Section III pressure vessels; these are to be considered by the ACRS within two or three months.

There was also considerable discussion on inspectability of pressure vessels, and the lack of access for inspection in BWR's at the Special Meeting in December, 1966. The ACRS was agreed that such access needed to be provided in future reactors. The question was, "How soon could and should new requirements in this regard be imposed." The Browns Ferry BWR's were coming up for review for a construction permit shortly, and their major preliminary design features were fairly well established. The Committee finally adopted the following motion:

The ACRS considers access for inspection by visual or ultrasonic means or other suitable methods, to the inside or outside of 100% of the vessel to be highly desirable.

The Committee realizes that gradual achievement of this aim is necessary, but hopes to see it fully achieved by January 1, 1968.

These matters were discussed further at the regular ACRS meeting on December 8-10, 1966 (the 80th). For BWR's, there was also some concern about access to the bottom interior of the reactor vessel for inspection (a thing that was very difficult in existing designs). The following ACRS position was adopted at the 80th meeting.

With regard to the inspectability of reactor pressure vessels for pressurized and boiling water reactors, the following is the position of the ACRS.

1. The interior of the vessel, including the bottom head, should be accessible for general visual observation on a scheduled periodic basis. Such observation has as its objective detection of mechanical damage or structural failure of reactor internals.

2. Practical means of access should be provided to essentially 100% of a reactor pressure vessel surface, either from the inside or outside or a combination thereof; the purpose of this access is to permit thorough inspection of the vessel at appropriate intervals by visual means and ultrasonic or other suitable methods.

3. The ACRS realizes that it may take time to achieve these aims, but expects to see them fully achieved in plants for which construction permit applications are filed more than one year after announcement of this position.

4. The foregoing should be announced formally to the nuclear industry within the next several months. The target date for the announcement should be selected at the February, 1967 ACRS Meeting.

   In the interim, the ACRS Chairman is to discuss this position with Mr. Price, and to explore with him possible means for informally advising the industry in advance of the date to be selected.

The minutes of the 81st meeting, January 12-14, 1967 and the 82nd meeting, February 8-11, 1967, record discussion and/or progress on pressure vessel research, or additional requirements and on inspectability.

I. Executive Session

A. <u>Proposed PVRC Research Program</u>

Dr. Bush noted that some months ago the PVRC had proposed to the AEC that a long-range experimental program be carried out to investigate heavy-walled steel sections used in pressure vessels. DRD&T has decided to take responsibility for eight of the subjects to be investigated, PVRC for the remaining three or four. Expenditures will total 8 million dollars over the next five years. The Committee had expressed its interest and Mr. Shaw has asked for ACRS comments on the proposal.

ORNL has been selected to administer the AEC part of the program and ACRS comments should be provided by next week at the latest, since a meeting has been planned to consider comments on the proposal.

The comments submitted by Dr. Bush and Dr. Okrent were discussed. It was decided that these should be provided as individual comments, with the information that the Committee has discussed them.

B. Special Working Group on Pressure Vessel Requirements

Dr. Okrent recalled that, at its December meeting, the Committee voted to ask Mr. Kavanagh and Mr. Price if they would be willing to work with the ACRS in forming a group to decide what should be required in pressure vessel fabrication, over and above Section 8 of the ASME Code.

Dr. Bush reported that the Working Group's first meeting had been held starting on January 10, 1967. The Group is comprised of Dr. Bush, Mr. Booth and Mr. Case. They have been provided with the services of several technical experts in DRD&T and DRL for consultation.

Four sub groups were established in the areas of design, materials, fabrication, and quality assurance, including inspection. The subgroups were to have prepared drafts in code language by March. In addition, longer-range goals were established for valve, pump, piping and pressurizer codes.

The group also heard a talk by Compliance personnel on the differences between field and shop fabrication and between various shop practices. There seemed to be areas which could be highlighted in field fabrication. Dr. Bush thought that a draft might be developed for use in connection with the Monticello review.

II. Meeting with Commissioner Ramey and the Director of Regulation

B. Inspectability of Reactor Pressure Vessels

Mr. Price thought that the idea of the Committee as embodied in the Committee's draft standard on inspectability was a good one but also that the best way of promulgating it needed consideration. He thought this might be partly included in the work of the special group which has been formed to consider pressure vessel requirements in addition to those imposed by the ASME Code.

He also thought that issuing a guide or a requirement which will not be made effective for some time would raise questions among the uninitiated.

Mr. Case said that the Department also had specific comments which it would like to make in the event a letter was written by the ACRS.

Mr. Palladino pointed out that there is a problem in too much delay. BWR's will have difficulty in meeting this requirement as presently designed, but they are not working toward increased inspectability and future plants will have the same lack in this regard unless the position is made known.

Mr. Price suggested that one way might be to include this item in the general design criteria and put them out for industry comment.

Mr. Palladino closed by saying that the Committee would have to consider the suggestions and arrive at a position.

D. Promulgation of Proposed Criterion on Access for Inspection (82nd Meeting)

The Committee discussed the Regulatory Staff proposal on promulgation of the proposed criterion. The Committee agreed to change the reference to access for "periodic" inspection to inspection at "appropriate intervals" in order to avoid the implication that annual inspections were being contemplated. The Comittee also agreed to include an additional paragraph stating the Commission's intention to effect the rule as soon as possible.

The Committee then unanimously voted to adopt the Staff proposal, with such other minor changes as the Chairman might care to make.

Mr. Palladino thought that no acceptable definition could be arrived at immediately and suggested that the original draft, with Mr. Price's revision, be sent. The Commission would have to be consulted, because of the reference to them.

The Committee adopted this suggestion.

Mr. Plaine was asked to continue to attempt a draft of a definition of undue risk, for the use of the Committee.

D. Promulgation of ACRS Pressure Vessel Inspectability Criterion

Mr. Palladino opened the discussion by noting that Mr. Price had suggested, during his discussion with the Committee, that the criterion be promulgated through the AEC's normal rule-making process.

Dr. Hanauer thought that he had not been particularly concerned about the time involved, however.

Dr. Okrent asked if Mr. Plaine shared Mr. Price's concern over issuance of a criterion which was not to be applied until after a stated interval.

Mr. Plaine did not. He pointed out that the whole system established for promulgating regulations is just such a system.

Dr. Zabel felt that Mr. Price was opposed to the issuance of formal requirements to the industry outside of the machinery of the Department of Regulation.

Dr. Hanauer observed that ACRS letters are final as they are issued, but that the regulations issued by the Department of Regulation are commented on by industrial groups and the ACRS before they become final. He felt that the criterion should be made known, but thought the Committee should recognize the usefulness of the lengthier procedure.

Mr. Mangelsdorf thought there were two reasons for issuing a regulation for comment, (1) the process might be educational and (2) industry resistance would be reduced thereby.

The Committee voted to have the Chairman find out, from Mr. Price, what specific comments he had on the draft criterion and to get from him a specific proposal for an appropriate method of promulgation, including timing.

Dr. Bush agreed that access provision would be included in the draft standards of the three-man Working Group considering additional requirements to those imposed by the ASME Code (See I.b).

Both initiatives, on more stringent fabrication requirements and on provision for inspectability moved surprisingly rapidly through the regulatory system;* and the code committee responded quickly, and began to develop similar though not identical requirements in the nuclear pressure vessel code. (Later, a new portion of the pressure vessel code, Section XI on Inservice Inspection, was developed.**

Another relatively prompt response to the November 24, 1965 ACRS report on pressure vessels lay in the initiation of new research and development programs. The AEC initiated a major safety research program on heavy section steels, which was to continue for more than a decade. And the Pressure Vessel Research Council took various initiatives to improve knowledge and eventually codes and standards relating to pressure vessel behavior.

One of the ACRS consultants, Dr. Paul Paris, had expressed reservations about the state of knowledge concerning the ductility after extensive irradiation of thick-walled pressure vessel sections, in the absence of good experimental informaiton. Considerable theoretical and experimental effort was devoted to this subject, and, while Paris' worst fears were not confirmed, a considerable effect of wall thickness on the fracture toughness (or nil-ductility transition temperature) was found, leading to a significant change in the requirements concerning an acceptable relationship between pressure and metal temperature. The minutes of the 103rd meeting, October 3-5, 1968 record this discussion.

Generally speaking, most of the results arising from the Heavy Section Steel Test Program indicated that at operating temperatures, pressure vessel steels were very tough, and not inclined to undergo rapid fracture.

Within the U.S., there was only one modest effort reported to the ACRS on efforts to cope with the possibility of gross pressure vessel failure. The Consolidated Edison Company informally submitted a report on a double steel walled pressure vessel design prepared by Combustion Engineering. Consolidated Edison had previously indicated it thought it would need to be able to cope with pressure vessel failure in order to obtain approval for constructing a reactor at a site much more populated than Indian Point 2. The concept was reviewed only briefly by the Regulatory Staff. The proposal was not part of a formal application. And the Staff did

* In August, 1967 the AEC issued tentative regulatory supplementary criteria of ASME nuclear pressure vessels for comment by code groups and the nuclear industry. In early 1968, the AEC formally requested public comment on such supplementary criteria. This prompt action not only improved pressure vessel integrity, it established the principle that the AEC could and would impose requirements over and above the code, if deemed necessary.

** The AEC-ACRS Primary System Review Group had developed draft requirements on pre-operational and in-service inspection by October, 1967.

not choose to try to examine the potential merits of the preliminary design in great detail to see if it offered significant promise of enhanced safety. Nothing ever came of this proposal.

Outside the U.S., at least two approaches to protection against pressure vessel failure have been considered. In Sweden, a fairly detailed conceptual design study was performed on an undergound reactor design intended for a possible urban site (the so-called Vartan reactor). However, this proposal was not pursued.

In Germany, a proposal was made to build a large PWR in the center of a heavily populated industrial complex, the so-called BASF reactor. After considerable review, the German safety groups accepted in principle the possibility of constructing a reactor at the site, but with very considerable additional safety features, going far beyond any proposed for Newbold Island, for example. The intent was to reduce the probability of a serious accident by at least a factor of ten, to balance against the increased numbers of people at risk and the fact that the larger population densities made evacuation much less feasible. One of the proposed requirements was a pre-stressed concrete, burst-proof container for the reactor pressure vessel. Considerable research and development effort was devoted to this new engineered safety feature. About the time final evaluation of the acceptability of this and other new features was to be made, the application for construction of the BASF facility was withdrawn.

In early 1971, the ACRS initiated a detailed re-evaluation of the reactor pressure vessel matter. It was initially proposed by some Committee members that the study consider both the status of pressure vessel reliability and the matter of possible measures to cope with vessel failure. However, it was decided to restrict the study to an evaluation of pressure vessel reliability. A fairly large Subcommittee was established with H. Etherington as its Chairman, and the subcommittee obtained the services of several consultants who were expert in specific facets of vessel integrity. The Subcommittee labored hard and long. After very many meetings the Subcommittee completed a rather long report in late 1973, which the full ACRS adopted as a Committee report and forwarded to AEC Chairman Dixy Lee Ray in a letter dated January 14, 1974. The principal conclusions of the report, which the AEC published soon thereafter as WASH-1285. "Report on Integrity of Reactor Vessels for Light Water Reactors" were the following:

WASH-1285, "Report on Integrity of Reactor Vessels for Light Water Reactors" were the following:

6. CONCLUSIONS AND RECOMMENDATIONS

The report reviews current reactor vessel construction practices, possible mode of failure, and operating practices pertinent to vessel integrity. It also reviews failure statistics of non-nuclear pressure vessels and gives an assessment of the disruptive failure probability of reactor vessels. Important conclusions and recommendations are recapitulated in this section.

6.1 Some Limitations in Scope of Report

Two important limitations have been placed by the Committee on the scope of the report:

6.1.1 Accident Conditions. The disruptive failure probability determined for reactor vessels is for vessels designed, constructed and operated in accordance with Code Sections III and XI. For reasons stated in Sec. 4.1, the probability does not include any contribution attributable to failures which conceivably could result from overstressing of the vessel by system accidents not contemplated in design.

6.1.2 Radiation Damage. The effect of irradiation is a matter requiring continuing attention. The Committee believes that, during the next few years, no large reactor vessel constructed to Section III of the Code will have been sufficiently irradiated for this matter to become a problem. The effects of irradiation will require careful review when larger fluences have been accrued and more data have become available. The material surveillance programs and research programs now being carried out should provide such data (Sec. 1.3, Sec. 3.3, Sec. 3.4, Sec. 5.8.4).

6.2 Current Practice

The technology of reactor vessels has been advanced greatly in recent years by marked improvement in steel quality, design methods, inspection techniques, and quality assurance programs. Some of these improvements are summarized here as background for recommendations recapitulated in Sec. 6.4.

6.2.1 Materials. The properties of steels used for the reactor vessel pressure boundary, SA-533 and SA-508, have been intensively studied for thick sections in the HSST programs. These steels provide a good balance of strength and toughness properties, with good metallurgical stability against changes occurring in service, including resistance to irradiation embrittlement (Sec. 2.1).

6.2.2 Design. Design rules under Section III of the Code differ substantially from those applied to conventional pressure vessels. The design requirements for nuclear vessels include extensive analytical stress calculations as well as fatigue analyses for combined mechanical and thermal stresses. Assurance of safe design of nuclear vessels does not rely primarily upon empirical design conservatisms, simple code rules, and established design details as does that of non-nuclear code vessels (sec. 2.2). On the other hand, higher design stress levels are permitted, thereby allowing thinner sections for the same application.

Fatigue analyses and inservice examinations give reasonable assurance that a fatigue induced crack developing during the service lifetime will not grow to a size sufficient to propagate rapidly, and that it will be detected and monitored if growth occurs (Sec. 2.2.1).

Section III rules impose design control measures on both the Owner and Manufacturer of the reactor vessel by requiring a Vessel Design Specification prepared by the Owner or his agent, a Stress Report prepared by the Manufacturer, and certification of these documents (Sec. 2.2.3).

6.2.3 Fabrication. Section III of the Code provides fabrication rules that are directed primarily toward controls, nondestructive examinations, and inspections performed during each stage of each of the major fabrication processes (forming, welding, heat treating, etc.) (Sec. 2.3).

The nondestructive examination requirements of Section I and Section III are compared in Table 2-1.

6.2.4 Preservice and Inservice Inspection. Section XI of the Code requires an enhanced program of preservice inspection, over and above the inspection programs required for fossil-fueled steam boilers, including ultrasonic mapping of all areas subject to periodic volumetric examination over the service lifetime, i.e., essentially 100 percent of the pressure-retaining welds. The preservice examinations also serve as a final confirmation of the structural quality of the vessel before it is placed in service (Sec. 4.3.1).

Section XI specifies periodic inservice examination requirements, including volumetric inspection of representative portions of pressure retaining welds. These examinations, in conjunction with the preservice mapping are intended to monitor extension during service of flaws in areas examined (Sec. 4.3.4).

6.3 Probability of Disruptive Failure

A disruptive vessel failure is defined for purposes of this report as a breaching of the vessel by failure of the shell, head, nozzles, or bolting, accompanied by rapid release of a large volume of the contained pressurized fluid (Sec. 5.1).

As assessment of the disruptive failure probability of nuclear reactor vessels has been made, based on:

(1) Consideration of the available failure data for non-nuclear types of vessels, such as boiler drums and unfired pressure vessels, and determination of the disruptive failure probability for such vessels; and,

(2) Comparison of the design, fabrication, materials, operating conditions, and inspection practices used for non-nuclear vessels with those used for reactor vessels, and estimation of the effect of the differences on the relative probability of disruptive failure of the two classes of vessels.

6.3.1 Probability of Disruptive Failure of Non-Nuclear Vessels. The Committee has reviewed available statistics of boiler drums and other non-nuclear pressure vessels and concludes that 99 percent confidence upper bound for the probability of disruptive failure (pertinent to reactor vessels) of such vessels is less than $1x10^{-5}$ per vessel-year (Sec. 5.7).

6.3.2 Probability of Disruptive Failure of a Reactor Vessel. The Committee believes that the disruptive failure probability of reactor vessels designed, constructed, and operated in accordance with Code Sections III and XI is at least one order of magnitude lower than that of the non-nuclear vessels evaluated. Accordingly, the Committee concludes that there is reasonable assurance that the disruptive failure probability of such reactor vessels is less than $1x10^{-6}$ per vessel-year (Sec. 5.9).

6.3.3 Probability of Disruptive Failure Beyond the Capability of the Engineered Safety Features. As defined, disruptive failures of reactor vessels include failures of various magnitudes and descriptions, not all of which would exceed the capability of the engineered safety features. Accordingly, the probability of reactor vessel disruptive failure beyond the capability of Engineered Safety Features is lower than the probability for disruptive failures of all kinds addressed in Sec. 6.3.2 (Sec. 5.10).

6.4 Recommended Supplementary Requirements for Reactor Vessels

Current practice gives a high degree of assurance against nuclear vessel failure. However, the Committee believes that reactor vessels, because of their greater importance to safety, should be considered in a class above other Class 1 vessels. The following recommendations are made.

6.4.1 Materials.

1. Explicit requirements for limits on residual elements, such as copper and phosphorous, which are believed to increase irradiation embrittlement, should be set forth for

materials subject to neutron fluences exceeding 10^{18} nvt (Sec. 2.1.3, Sec. 2.7, Item 3).

2. Maximum strength levels should be included in specifications of all materials, including SA-508 (Sec. 2.1.3).

3. New and high strength materials permitted by the Code such as SA-542 and SA-543 should not be used until their predicted behavior in nuclear service is well understood and documented and the ability to control their properties has been fully established by extensive test programs (sec. 2.1.4).

4. The Committee recommends that an investigation be made in the US into sensitivity of SA-533 and SA-508 welds to hydrogen damage as a result of deviation from optimum conditions (Sec. 3.1.2).

6.4.2 Design.

1. More definitive material toughness requirements for emergency and faulted conditions should be stated in the Code rules, taking account of the variable relationship between vessel temperature, level of stress, and fracture toughness (Sec. 2.7, Item 1).

2. First-of-a-kind vessel designs should be subject to confirmatory design review by the owner or his agent (Sec. 2.7, Item 5).

3. Present Code requirements for the Vessel Design Specification, Vessel Stress Report, and Certification of Stress Report for Compatibility with Design Specification should be interpreted as involving at least two independent organizations in the preparation or review of documents (Sec. 2.7, Item 4).

4. For a vessel that may be exposed to a neutron fluence of 10^{18} or more (> 1 Mev), the owner should assure himself that the system design permits a practical procedure for annealing the vessel in service in case this should become necessary (Sec. 3.4).

5. The Committee recommends that the biological shield (reactor cavity walls in PWR's) and other structures be designed to withstand the effects of a failure of the thin-walled part of the nozzle adjacent to the safe-end (Sec. 3.6.4).

6.4.3 Nondestructive Examination. The potential exists for further improvement in vessel defect characterization by the application of newly developed techniques, such as acoustic emission, acoustic holography, and acoustic spectroscopy. These techniques should provide a basic tool for determining, more precisely, flaw locations and growth characteristics in reactor vessels and may make an important additional contribution to vessel reliability. Attention has recently turned to supplementing current practice by acoustic emission techniques capable of showing the location of flaws by acoustic response during structural loading. These methods, when developed to the stage where they can quantify defect location and significance, should be applied to the manufacturing and preoperational testing phases, and possibly to the operating phase (Sec. 2.7, Item 6 and Sec. 4.3.4).

6.4.4 Inservice Inspection and Surveillance. Periodic visual inspection and timely repair of cracks has contributed much to the good safety record of fossil-fueled boilers. Although most of the cracks in boiler drums have resulted from thermal cycling and peak stresses that would not be permitted by Section III, it is important that good inservice inspectability be maintained for reactor vessels. Section XI of the Code requires visual examination of some critical areas, and 100 percent volumetric examination of other representative areas. Sensitive leak detection systems are also installed to monitor leakage from the primary pressure boundary, including the vessel.

The Committee offers the following additional observations and recommendations with respect to inservice inspection and surveillance.

1. The Committee believes that reactor designers should give greater attention to providing accessibility for inservice inspection. This applies to volumetric inspection and to visual inspection by optical means (of both interior and exterior surfaces), especially at regions of severe constraint (Sec. 5.8.4).

2. The Committee believes that experience and further development will show that inservice ultrasonic examination of welds is not a very time consuming or difficult operation. If this is demonstrated, consideration should be given to requiring more extensive vessel inservice ultrasonic examination than presently required by Section XI (Sec. 4.3.3).

3. The Committee believes that, although the "leak-before-break" criterion cannot be relied on, a potential failure might be averted by advance warning of leakage through a crack of subcritical size. More sensitive systems for detection and location of possible reactor vessel leaks should, therefore, be provided (Sec. 3.3.4).

4. Periodic overpressure testing is not required for reactor vessels; however, the Committee believes elimination of periodic tests should not be regarded as a firmly established practice and that further consideration should be given to this matter (Sec. 4.3.6).

5. The Committee recommends that periodic bolting examination required by Section XI be interpreted as requiring surface examination of both threaded ends of main closure studbolts (Sec. 3.6.5).

6.4.5 Operation.. The Committee recommends that Technical Specifications specify heatup and cooldown pressure-temperature limits that can be shown to be as conservative as practical with respect to 10 CFR Part 50, Appendix G (Sec. 3.3.3).

6.4.6 Research and Development. The adequacy of the Research and Development (R&D) pertaining to reactor pressure vessels has not been examined fully by the Committee as part of this review. The information provided from the HSST, EEI-TVS, and other programs has been valuable in assessing vessel reliability; however, the data have not yet been analyzed completely. The status of the R&D effort requires further study to determine what work should be extended, modified, or initiated to support the reliability requirements of nuclear reactor vessels. The necessary work may include further investigation of vessel failure modes, material properties, and inservice acoustic emission techniques. Sufficient work should be carried on to assure the continuing availability of expert current knowledge of reactor vessel technology to the AEC.

6.4.7 General.

1. In order to increase the assurance that QA programs are adequate, and thereby to minimize the probability that defects will remain undetected, the Committee recommends that the individual responsibilities of the principal organizations that establish QA procedures and controls be subject to audit by the owner and by the AEC (Sec. 2.7, Item 7).

2. Where owner's or manufacturer's quality requirements ordinarily exceed minimum Code requirements, consideration should be given to upgrading the Code to conform to practice (Sec. 2.7, Item 2).

3. The Committee believes further study of possible design changes to protect against vessel failure should be performed (Sec. 4.4.4).

6.5 Reactor Vessels Not Covered by Report

This report applied primarily to vessels construted of SA-533 and SA-508 steels, designed and constructed to Section III of the Code, and operated in accordance with Section XI. The report further stipulates that future consideration should be given to the effect of increasing irradiation.

The Committee recognizes that some older vessels are constructed of other steels, that they were designed to Sections I and/or VIII of the Code, and that only limited conformance to Section XI is practical.

Moreover, the belt zones of some of these vessels have been significantly irradiated. The provisions of Appendix G to 10 CFR 50 should give reasonable assurance against failure, but it may become increasingly difficult to apply the conservatism recommended by the Committee (Sec. 3.3.2) in applying Appendix G.

The Committee recognizes that these older vessels are under continuing surveillance by AEC, but recommends that a documented review be made of their present status and of the Commission's rules governing their operation.

Not too long after issuance of WASH-1285, the Regulatory Staff issued its own report on pressure vessel integiry, (USAEC, 1974) in which it basically agreed with the ACRS conclusions. And in WASH-1400, the median failure probability of pressure vessels was taken to be 10^{-7} per reactor year with resulting conclusion being drawn that pressure vessel failure was not a principal contributor to risk from LWR's.

However, there have remained a number of events and situations which have kept the matter of pressure vessel integrity from disappearing from sight. The ACRS report did not conclude what the effects of transients might be on pressure vessel integrity. The matter of Anticipated Transients without Scram (ATWS) remained to be resolved and ATWS provided a possible mechansim for vessel failure. In 1973, when WASH-1285 was completed, there had been a few incidents of reactor overpressurization while the primary system was cold. By 1977, this had grown to be a large number of incidents (approx. 20), and the Regulatory Staff took generic action with all PWR operators, requesting that short-term measures be taken to correct the breakdown in administrative controls (and other human errors) which was leading to overpressure events, and that longer term measures be considered to reduce the reliance on administrative controls.

W. Vesely, et al., of the NRC Safety Research Staff presented a paper in 1978 (Vesely, 1978) in which they estimated the probability that such overpressurization might lead to vessel failure. In a new vessel which had the originally specified fracture toughness they calculated the probability to be small. But, near the end of life, when some PWR's have received a substantial integrated neutron dose to the reactor vessel belt, they estimated a substantial probability of vessel failure due to cold overpressurization, roughly 10^{-5} per reactor year.

A concern raised in the ACRS report related to older vessels, for which no provision for in-service inspection had been made and for which many parts of the vessel might be inaccessible. Also, there was poorer knowlege of the original status of the vessel quality.

The Regulatory Staff reviewed this matter and in 1976 issued a report, NUREG-0081 (NRC 1976) which basically stated the existing situation was acceptable (with the exception possibly of one reactor). This conclusion was largely a matter of judgment, since a quantitative estimate of the vessel reliability is difficult to obtain.

Among the other matters which have kept the matter of pressure vessels in sight was the controversy in the United Kingdom. Professor Alan Cottrell questioned that pressure vessels for LWR's could be made with acceptable integrity. After some considerable argument and study, a report was issued on October 1, 1976 by a Committee headed by Dr. D. W. Marshall (Marshall, 1976), which generally endorsed the integrity of LWR pressure vessels, although it made several recommendations for improved quality assurance.

Other questions which hae arisen include the finding of multiple small cracks in nozzle weld regions by one of the German research groups, and the potential for multiple concurrent failure in the large number of small instrument lines entering the bottom of a PWR.

One of the most controversial aspects of the first draft related to a curve based on a report under preparation by M.G. Bonilla, of the U.S. Geological Survey, which related surface displacement to magnitude of earthquake on a surface fault. Also controversial was the choice of the minimum distance from the center line of a fault beyond which a reactor might be located. Another item which received much scrutiny and comment was the definition of a "capable fault," meaning a fault having surface expression and which was deemed capable of exhibiting permanent relative displacement on the two sides of the fault as the result of an earthquake.

It is to be expected that with time, as more and more vessels undergo in-service inspection, relatively large flaws will be found. And extremely difficult decisions will have to be made concerning the acceptability of continued operation, since the costs of repair may be very large. The Pilgrim reactor has already provided a very small example of the problem.

A very large number of meetings were held between the ACRS and the Regulatory Staff, and many revised drafts were prepared. In the latter half of the 1960's, the Regulatory Staff seemed to have the requirement that it obtain comment and preferably concurrence from the AEC Division of Reactor Development and Technology on such criteria, although the latter represented the AEC "promotional side."* And, during the period when the Bolsa Island project was active, the seismic criteria were held in a state of abeyance by Mr. Price while their potential impact on Bolsa Island (a project important to the AEC) was assessed.

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*See following excerpt from the 98th meeting, June 5-8, 1968, and the 105th meeting, January 1969.

98th Meeting, June 5-8, 1968 - Seismic Design Criteria

Dr. Okrent inquired into the status of the seismic design criteria. Mr. Price reported that a draft of the proposed criteria had been sent to DRD&T. A copy of the draft had been reviewed with Dr. Lieberman, and then it had been sent to Mr. Shaw about six weeks ago. Mr. Price said that he did not want any more unmanageable comments. He stated that a couple of days prior to this meeting he received a draft memo and comments, however the date established for that meeting was the day prior to this meeting, however it did not materialize. Mr. Price said that he expected such a meeting to be held within the next week or so. He said his present mood was to submit a draft seismic design criteria to the Commissioners with or without Mr. Shaw's input, however he said he was trying to reach agreement with DRD&T so as to approach the Commission without conflict.

Dr. Okrent noted that even if Mr. Shaw didn't have a conflict of interest, that DRD&T should not control the regulation of criteria. Dr. Okrent said that he would like a copy of the draft that Mr. Shaw is looking at now. He suggested that if there is a serious disagreement between the Division of Regulation and RDT that the disagreement should be taken to the Commission. He pointed out that it was a Committée privilege to recommend criteria. Mr. Price said that he would like to know the areas of disagreement between Regulation and RD&T, and he would be pleased to provide Dr. Okrent with a copy of the present draft. He pointed out that there seemed to be some technical difficulties between Regulation and DRD&T, and he is not sure how this criteria would

3.3 OYSTER CREEK: QUALITY CONTROL AND BACKFITTING

By letter dated March 26, 1964, the Jersey Central Power and Light Company submitted an application for authorization to construct a 1600 MW BWR at the Oyster Creek, New Jersey site. Within five months, the Regulatory Staff and the ACRS had completed the construction permit review, and on August 25, 1964 the Committee issued a report favorable to construction of the reactor, as is reproduced below.

The Subcommittee minutes for the Constrution Permit review note that the Preliminary Safety Analysis Report for Oyster Creek was about "the simpiest" the ACRS had been asked to review, and the Committee letter notes that many details of the design had not been completed.

On January 25, 1967, Jersey Central subumitted a final safety analysis report in support of its request for an operating license. It also requested a priority review of the emergency core cooling system (ECCS), this in response to a letter sent to the utility by the Regulatory Staff on October 20, 1966 which outlined the problem of "backfitting" for improved core cooling capability. The following excerpt from the minutes of the 81st meeting, January 12-14, 1967 provides a little background.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

August 28, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON OYSTER CREEK NUCLEAR POWER PLANT OF JERSEY CENTRAL POWER AND LIGHT COMPANY

Dear Dr. Seaborg:

At its fifty-seventh meeting, on August 24-26, 1964, the Advisory Committee on Reactor Safeguards considered the proposal of the Jersey Central Power and Light Company to construct and operate a nuclear power plant on Oyster Creek in New Jersey. This will be a 1600 MW(t) boiling-water type reactor with pressure absorption containment.

The Committee had the benefit of an oral presentation by representatives of the applicant and consultants and contractors, advice by the AEC Staff, and the reports cited. A Subcommittee meeting was held at the site on May 1, 1964, and a further Subcommittee meeting was held in Washington, D. C. on August 7, 1964.

Many details of the proposed design have not yet been completed. The applicant is continuing to study the limitation of maximum reactivity of individual control rods and the design of the reactor protection system. The following additional points should be given examination and consideration:

(1) Under some credible accident conditions, the dry well and absorption pool may require provisions for additional heat removal.

(2) In the unlikely event of a melt-down accident, a zirconium-water reaction may produce hydrogen. Provision should be made to prevent any hydrogen-oxygen reaction that would disrupt the containment.

(3) The adequacy of the reactor protection system when operating at partial recirculation flow rates should be established.

Estimates made by the applicant on halogen retention by absorption in water and by plate-out are based on limited data, and the consequences of the unlikely accident may be more severe than estimated. However, the Committee believes that more conservative assumptions would not make the proposal unacceptable.

With due regard to the above comments, the ACRS believes that the proposed reactor can be constructed at the proposed location with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

References:

1. Part B, Preliminary Safeguards Summary Report, Application to the United States Atomic Energy Commission for Construction Permit and Operating License, Oyster Creek Nuclear Power Plant Unit No. 1, Jersey Central Power and Light Company, undated, received April 2, 1964.

2. Amendment No. 2, Application Reactor Construction Permit and Operating License, Oyster Creek Nuclear Power Plant Unit No. 1, Jersey Central Power and Light Company, dated June 26, 1964, with enclosures.

81st Meeting, January 12-14, 1967

Backfitting

Mr. Price stated that letters had been sent to all companies with operating reactors, or reactors under construction, with power levels in excess of 50 MW. The letters asked for a review of the core cooling question. Because of the response it became evident that the Commissioner should first have been informed. The reaction was generally one of concern over the costs which might be involved. Mr. Price suggested that the present standards be articulated so that one has an idea of what backfitting would involve.

Dr. Morris observed that in the case of ECCS, at least an industry/advisory group has been established to provide guidance. In other areas, the ACRS, DRL and their consultants could determine present criteria. The operating plants can then be reviewed to see what could be done by way of improvement.

Commissioner Ramey advised the Committee that during the past few months, utilities and equipment manufacturers have held discussions with the Commission and have expressed their concern over snowballing safeguards requirements leading to increased costs. They have also indicated their feeling that increasing redundancy leads to the point where overall safety is degraded. They feel some balance is necessary.

The concern of the Commission was with the sense of urgency in the letters, which indicated that it was a matter of the highest priority that the information be gathered. This indicated to some people that major changes were to be instituted with some urgency.

Dr. Hanauer observed that in the case of plants in the early stages of construction, there might well be some cause for urgency.

In Report No. 1 to the ACRS, dated June 16, 1967, the Regulatory Staff reviewed the background of the problem, provided several conclusions on required changes in ECCS, and identified several areas needing further study. We excerpt from this report below.

2.0 Background

The Oyster Creek (OC) plant utilizes the non-jet pump type reactor vessel design which is significantly different from, and predecessor to, the current class of GE-BWR jet pump plants. There are five (5) external recirculation loops (26" dia. piping) with "below core" vessel penetrations. Thus, core flooding capability following a major loss-of-coolant accident (LOCA) is not inherent in the design as it is with the jet pump plants; i.e., on jet pump plants the two recirculation lines do not enter the vessel below the core. A low pressure flooding system has been proposed for all the current GE-BWR plants reviewed at the construction permit stage. Jersey Central has not proposed such a system for Oyster Creek.

At the time OC was reviewed for a construction permit the emphasis was placed on the primary containment with the assumption that it represented the "true final boundary." Since that time, consideration of the "core melt problem" caused a shift in emphasis to include core cooling capability. This has created a sizable gap in the equipment provided for the OC-ECCS as compared to present standards.

The ECCS proposed for the OC plant at the construction permit stage consisted of two single core spray loops (with no internal active component redundancy) and the feedwater system. Since then, 100% redundancy as well as booster pumps for higher pressure operation in each core spray loop have been added. In addition, a semi-automatic depressurization system (ADS) was incorporated to protect the core for small breaks in the event off-site power was lost. This system reduces reactor pressure to that level to allow the low pressure core spray system to operate and cool the core. These changes have been made voluntarily by Jersey Central.

These additions, however, do not update the ECCS capability to that now proposed for Dresden 2 class (jet pump) reactor plants. This raises the question of "backfitting." Other areas, including earthquake criterion, missile and pipe whiplash problems, tornado design basis, to mention a few, have been recognized as critical design areas and will be considered during our evaluation of the entire plant.

The interactions of the ECCS with the remainder of the plant are discussed in section 7.0 of this report.

We have addressed the problem of ECCS backfitting in our review. The positions stated in section 6.0, when implemented, would, in our opinion, be a significant advance toward backfitting the OC plant. Further study and evaluation are, however, required to establish the extent of backfitting necessary for OC-ECCS.

A comparison of the ECCS equipment proposed for OC with current GE-BWR's (at the construction permit stage) is shown in the table below.

TABLE 2.0

ECCS COMPARISON OF OYSTER CREEK AND CURRENT GE-BWR PLANTS

System	Oyster Creek	Current GE-BWR (Jet Pump) CP Application
Feedwater	Yes	Yes
High Pressure Injection System	No	Yes (HPCIS)
Semi-Automatic Depressurization System	Yes	Yes
Low Pressure Injection System	No	Yes (LPCIS)
Core Spray	Yes	Yes
Redundant On-site Power	No	Yes

3.0 Evaluation Approach

Our review of the functional aspects of the OC-ECCS was directed towards the performance capability to maintain the core in a geometry which will ensure continuous core cooling following a design basis loss-of-coolant accident. Meeting this objective requires that the core remain in an intact and definable geometry such that the ECCS would reverse any temperature transients before exceeding the melting temperature of the fuel cladding, and further to limit any metal-water reaction to the order of 1%. We consider that the ECCS redundancy should be such that a single failure of any active component will not decrease the ECCS performance below minimum requirements. In addition, failure of any ECCS sub-system or component (active or passive) should not result in a loss of core cooling and the ECCS minimum performance should not be affected by any single failure in the on-site electrical systems, AC or DC.

8.0 Conclusions

On the basis of our review of the proposed OC-ECCS (in terms of the functional performance) we have reached the following conclusions:

8.1 ECCS Changes Required:

- provisions necessary for redundant on-site AC and DC power sources

- multiple core spray system initiation and actuation of ECCS

- addition of a high pressure injection system

- ECCS capability to be immune from passive element failures

- program all four relief valves to open on initiation signal.

8.2 Areas Requiring Further Study:

- containment flooding

- diversification in ECCS initiation signals

- pressure relief capability for reactor vessel

- low pressure flooding or alternate cooling system

The Regulatory Staff pursued the ECCS matter with high priority, and in their second report, dated April 18, 1967, were able to make fairly specific findings, as shown in the report abstract which is reproduced below.

> This is the second report to the Committee concerning our review of the Oyster Creek reactor plant. The review and evaluation of the emergency core cooling system presented in this report supersedes that given in our first report and represents our evaluation of all aspects of the system necessary to approve it for the operating license.
>
> The functional aspects were reviewed in terms of the ECCS capability to cope with any design basis coolant loss accident over the entire primary system piping break spectrum. The mechanical aspects of the core spray system components located inside and outside the reactor vessel were investigated to determine the effects of seismic and blowdown forces and thermal shock on mechanical integrity. Our review of the instrumentation, control and emergency power systems has resulted in numerous changes and modifications that need to be made before plant operation. These would increase the protection against the effects of single failures.
>
> On the basis of our review, we make the following findings:
>
> - Additional emergency core cooling protection is required for the small break region to prevent reactor vessel blowdown via the auto relief system for very small breaks in the primary system. This added protection would provide greater assurance of preventing fuel clad heat up and fuel rod perforations in

the event of such breaks. The additional protection required need not be implemented before initial plant operation, however, we believe that the preliminary design must be provided for our evaluation prior to issuance of a provisional operational license.

- With additional emergency core cooling protection for the small break region, the results of our evaluation of the ECCS proposed for Oyster Creek provides reasonable assurance that the ECCS would prevent a core meltdown in the unlikely event of a loss-of-coolant accident.

- Adequate protection has been provided by the applicant to prevent loss of core cooling capability in the unlikely event of excessive water leakage from the ECCS ring header complex.

- There is reasonable assurance that the core spray and sparger inside the reactor pressure vessel will withstand design basis accident loads. Our review of the ECCS piping system external to the reactor vessel is pending receipt of the analysis on this portion of the piping system from the applicant.

- Provisions have been incorporated in the ECCS design that provide a basis for a reasonable testing and surveillance program. Details of the program will be developed during our review of the Technical Specifications.

- Sufficient redundancy has been provided so that no single failure of an active or passive component will decrease the ECCS performance below design requirements.

- The instrumentation, control and emergency power systems are acceptable pending resolution of fuse coordination test and level sensor error to our satisfaction.

- The three day on-site diesel fuel storage capacity in conjunction with the available off-site make-up sources is acceptable for the Oyster Creek Plant.

- An interlock should be provided to prevent blowdown whenever there is a complete loss of AC power.

We are attempting, by means of extraordinary effort to complete our safety evaluation of the entire facility by the December 1967 ACRS meeting. Achieving this objective is dependent upon the adequacy of the additional information provided by the applicant and the acceptability of the proposed Technical Specifications.

The ACRS supported the Staff position on ECCS, and after considerable prodding the reactor vendor proposed additional high pressure emergency cooling adequate to meet the Staff requirements. It was impractical to provide an ECC system to reflood the core in addition to core sprays, as was done for Dresden 2 and the succeeding BWRs, because of the different reactor design of Oyster Creek in which "bottom" piping breaks could empty the reactor vessel. However, redundant on-site D.C. and A.C. power sources were possible and were backfitted.

Many other safety aspects differed in Oyster Creek from what the Regulatory Staff was requesting of new plants under consideration for construction in 1967. The following is a partial list of items which were reviewed but not "backfitted" to meet the new criteria:

1) The two isolation valves in each steam line were both outside containment, instead of one in and one out, creating the possibility of a non-isolable steam line rupture outside containment.

2) Tornado protection was much less than the 1967 requirement.

3) Whipping of large pipes within the drywell under certain postulated rupture conditions could damage containment integrity.

4) Access for periodic inspection of the pressure vessel and primary system was limited.

Thus, while many backfitting questions arose during the review, it was found to be impractical to make major hardware changes. And even changes in electrical circuitry were required only when a very strong safety connotation was involved.

The experience with backfitting questions during the Oyster Creek review was similar, though probably more difficult, to that encountered with the other operating license reviews undertaken after the summer of 1966. Between the new emphasis on ECCS and the development of draft General Design Criteria, many differences were apparent between as-built plants and the new requirements. The problem remained a very thorny one, with decisions made on each reactor on a judgmental basis which involved both the safety significance and the ease of making modifications. The experience led the Regulatory Groups to require more detailed design information at the construction permit stage. It also brought large protests from the nuclear industry which eventually resulted in the adoption in 1970 of the so-called "Backfitting Rule" which required that, in order to require a backfit, the Regulatory Staff had to make a finding that "such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security." The "Backfitting Rule" unquestionably exercised an inhibiting effect on the Regulatory Staff in this regard, and perhaps led to a tendency for several years for the Staff not to "look" for backfit questions on older plants.

The first ACRS Subcommittee meeting during its operating license review for Oyster Creek was held on July 28, 1967, and included a tour of the facility, as usual. During that meeting various site-related safety issues, as well as matters pertaining to ECCS, were discussed. But the most significant thing to arise from the Subcommittee vist, however, was a considerable concern about the quality of the field construction and fabrication.

The minutes of the 91st meeting, November 2-3, 1967, record the ACRS concern with the quality control of welding, with the sloppiness of the construction site, and with the lack of an adequate quality assurance program, in general. In fact, between the July Subcommittee visit and the November meeting, a specific major quality control problem had turned up. During a hydrostatic test of the reactor pressure vessel (prior to reactor operation) on September 29, 1967, a small leak was noted near one of the control rod drive housings. Examination revealed that the vast majority of all the control rod housings, over one hundred, had suffered cracks in certain welds, and that there was a generic deficiency in the combination of design and welding. A major program was required to define an acceptable repair procedure and to reweld all the deficient joints.

This very major set of flaws turned out to be only a part of a long story. The Regulatory Staff, on being made aware of the ACRS concern with the overall quality assurance picture, decided to focus special inspection attention on Oyster Creek. The problem of inadequate quality assurance turned out to be so severe that startup of the reactor was delayed for at least a year beyond the date which the utility had been insisting was vital.

In a report to the ACRS dated November 29, 1968, (a year later), the Regulatory Staff documented a large number of deficiencies in the quality assurance program, some of which still remained to be resolved. The same report lists many deficiencies in the installation of circuits for instrumentation and power, involving violation of the criterion requiring separation of redundant circuits, most of which could be remedied in a practical fashion, fortunately.

Further deficiencies in quality assurance turned up in 1969, when some cracked valves were investigated, and it was found that at one point the quality assurance program had lapsed sufficiently to permit the purchase and installation of second-hand valves of unknown condition.

The findings on Oyster Creek provided a very great impetus for the generation of a structured quality assurance program and for a greater amount of inspection audit by the Staff. High priority was given both to the development and adoption of an AEC Rule on Quality Assurance, and to pressing the industry, particularly the utilities, to see to it that improved quality assurance programs were instituted. The Commissioners decided that the Regulatory Staff (and not the ACRS) would have the responsibility for determining that adequate quality assurance was actually employed on each reactor.

The question of quality assurance has remained troublesome. The Browns Ferry Fire (Hanauer, S. H., 1976) and the fabrication problems arising at North Anna (USNRC) are two examples from a rather long list.

The Browns Ferry fire illustrates a failure of the utility to institute adequate quality assurance practices, the Staff or the industry to develop adequate fire protection codes or regulations, and the Regulatory Staff to act in timely fashion on an identified problem. It is a perfect example of the thorny decision-making process which enters where backfitting is involved.

A small sidelight on the history of the evolution of quality assurance dates back to 1963. The minutes of the 46th meeting, January 31, February 1, 2, 1963 record a meeting between the ACRS and the Commission as follows:

> Chairman Seaborg, Commissioners Palfrey, Ramey, Haworth and Wilson joined the Committee with members of the AEC Staff.
>
> Dr. Hall (ACRS Chairman) presented a draft of a letter to the Commission recommending improvement in the quality and reliability of materials, equipment and fabricated facilities toward better engineering safeguards for reactors; Mr. Price, who had not seen the draft, preferred not to comment. Dr. Hall pointed to the review of reactors at the site approval, construction permit, and operating stage, and he noted that surveillance to insure continued vigilance toward satisfactory condition of the plant and safety is difficult. Although some of the Commissioners believed such a formal letter might be useful, it was concluded, following Dr. Wilson's comments, that bringing the matter to the attention of the Commission orally was sufficient; he feared the letter might be misunderstood by the public and reflect unfavorably on the reactor program.

Six years after Oyster Creek, quality assurance became a major aspect of the AEC's effort to improve both the reliability and safety of reactors.

3.4 PRAIRIE ISLAND AND THE STEAMLINE BREAK ACCIDENT

The ACRS had completed its review of the construction permit for Prairie Island Units 1 and 2 (1650 Mwt PWRs) at its 95th meeting, March 7-9, 1968. That review was known most for the so-called "Prairie Island position" on separation of control and protection instrumentation, which is discussed in Chapter 4 in the section entitled "Separation of Protection and Control."

The construction permit review of the Kewaunee reactors, which were very similar to those at Prairie Island, was completed by the ACRS at its 97th meeting, May 9-11, 1968.

The Regulatory Staff report to the ACRS of February 27, 1968, on Prairie Island included the usual analysis of the off-site consequences of a postulated steamline break accident, and with a transmittal memorandum dated September 28, 1972, the Regulatory Staff forwarded to the ACRS its Safety Evaluation Report concerning the proposed operation of the Prairie Island plant.

The Staff report pointed out no major problems or controversies between it and the applicant, and it appeared it would be a relatively routine ACRS review. The Staff report stated that Prairie Island was designed and constructed to meet the AEC's General Design Criteria, as proposed in July, 1967. The applicant had not been asked to reanalyze the plant against the General Design Criteria adopted in 1971; however, the Staff said "our technical review did assess the plant against the General Design Criteria now in effect and we are satisfied that the plant design generally conforms to the intent of these criteria."

The Staff report includes short sections on the steam and feedwater lines and an analyses of the off-site consequences of a main steamline break.

The ACRS Subcommittee on Prairie Island met on October 24, 1972, and the case was scheduled for full Committee review at a special ACRS meeting held October 26-28, 1972.

On October 25, 1972, the ACRS office provided to ACRS members the following two pages received from sources unknown, in a letter postmarked Philadelphia, Pa.

> There are several safety related unresolved items between DRL and the applicant on both Kewaunee and Prairie Island Projects. Some of these are:

1. Non-fulfillment of old criterion 20, 21, 40, 41 and 42.

2. A rupture anywhere (everywhere) outside the containment in in the steam line including the rupture of a relief header produces intolerable consequences in that the walls, floors and ceilings will collapse in less than 60 mili seconds with loss of all electrical and mechanical equipment. A massive destruction of the Aux Building or the control room will result and, therefore, safe shutdown of the reactor is jeopardized.

3. Containment over power protection systems are very different from other two loop plants and indeed reduce the safety of the plant in the event of a LOCA or steam line break inside the containment.

4. DRL has not done any independent containment pressure transient analysis nor have they performed any compartment pressure analysis.

5. Several Electrical and Instrument & Control Changes affecting safety systems have not been received by DRL e.g..
   Accumulator & RHR Valve interlocks
   Boric Acid System which is full of holes

6. Loss of a station Battery (Single Failure) past LOCA will prevent both diesels from startup and also prevent several 2 out of 3 logic from performing properly.

7. Turbine overspeed missile will go right through the containment dome, the Aux Bldg. Roof, Spent Fuel Pool Roof etc..

8. Changes in the Westinghouse ECCS Model especially Reflood and Blowdown have not been reviewed extensively by the Staff.

9. Seismic Analysis of Critical components have not been satisfactorily completed.

10. Primary Containment leakage of 0.1%/day is very easily attainable from present day technology and, therefore there is no need to allow higher leak rate and thereby expose the public to a higher off-site dose.

11. Several potential leak paths have no Iodine Filtration and go directly to the atmosphere. These paths should be modified to fall within the Secondary Containment boundary. This path alone contributes to more than 50% of the total off-site dose.

12. Containment pressures of 60 psig (75 psia) are very real in an accident situation and, therefore, over pressure test at 80 or 85 psia should be performed to assure containment integrity.

13. The Containment Free Volume has never been verified by DRL and is definitely well below the assumed values.

With just these many items alone it is not in the public interest to allow further actions until complete review by DRL.

The anonymous memorandum was discussed briefly following the ACRS Subcommittee report, both in Executive Session, and it was decided that members would try to interpret the significance of the listed items and explore them during the upcoming discussions with the Regulatory Staff and the Applicant.

Some of the items listed, such as the effect of a turbine overspeed missile, represented known effects or phenomena which were already under study on a generic basis as to their potential for causing serious accidents and the possibility of requiring changes in current or future plants. Some appeared to have been met acceptably, or represented areas where it was reasonable to expect the Regulatory Staff to handle the matter prior to their actual issuance of an operating license; and this was confirmed by questions to the Staff or the Applicant.

ACRS member Etherington made some crude calculations concerning the possible pressure buildup in the auxiliary building from a gross steam line rupture, and judged that the point raised in the anonymous memorandum might have some validity. When this issue was raised with the Regulatory Staff and Applicant, it quickly became clear that the Applicant had not evaluated such a rupture in his safety design, nor had the Regulatory Staff reported on the issue in their review. It also confirmed that a potentially serious safety problem could be associated with the rupture of steam lines and other high energy process line outside containment.

The ACRS decided not to complete action on its operating license review for Prairie Island, and forwarded the anonymous memorandum to the Regulatory Staff for review and comment. The Staff met with the Applicant on November 3, 1972, and on November 8, 1972, the Staff forwarded an information report to the ACRS in which they concluded that the overall matter of steam and feedwater line breaks required further evaluation and advised that they were requesting considerable additional information from the Applicant.

At the 151st meeting, November 9-11, 1972, the Regulatory Staff reported to the ACRS on the steam line break matter for Prairie Island, as follows:

> "Many modifications to the existing Auxiliary Building will be required to provide pressure relief in the event of steam line failure, and the Applicant, Northern States Power Company, has not yet completed the evaluation of the consequences of a doubled ended break of the main steam line in the building as requested by the Regulatory Staff. His current analysis identifies a single-ended rupture of the largest (6 inch) main steam line branch connection as the design basis accident. The Committee was informed that engineered safety features and other critical equipment in compartments exposed to the steam atmosphere following a break would have to be qualified for service in such an environment."

When other PWRs besides Prairie Island and Kewaunee were examined with regard to this safety issue, the concern was found to apply to most of the reactors under construction or in operation to a greater or lesser degree. Each reactor was, therefore, reviewed and evaluated specifically and in detail for adverse effects from pressure, forces, and the high temperature steam environment which would accompany large ruptures of such lines.

The event itself should be one of fairly low probability; however, the probability was not thought to be so low that the potential effects could be neglected. Some of the reactors required large changes at considerable expense in money and time in order to reach an acceptable level of protection against such breaks.

Why the matter had slipped through the regulatory process is obscure. The ACRS assumed that the General Design Criteria required consideration of such postulated failures. Apparently, the architect-engineers had been using a design basis for such postulated ruptures which had been accepted prior to 1966. And, while the Regulatory Staff was reviewing in increasing depth the effects of postulated ruptures of the largest pipe in the primary system, they had not extrapolated this approach to the secondary system of a PWR.

By March, 1973, the Prairie Island Applicant had established what design changes and other measures were needed to cope with process line breaks outside containment. Another Subcommittee meeting was held March 31, 1973, and at its 156th meeting, April 12-14, 1973, the ACRS wrote a report favorable to the operation of the Prairie Island Reactors.

All in all, the main steam line break issue was confronted and resolved in a relatively expeditious manner, even though some refinements in the Regulatory Staff position took time to evolve.

4. ANTICIPATED TRANSIENTS WITHOUT SCRAM

4.1 SOME EXCERPTS FROM WASH-1270 "ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR WATER-COOLED POWER REACTORS", SEPTEMBER, 1973

I. Background and Summary

. "ATWS" is an acronym for "anticipated transients without scram." It is a subject that has been mentioned in reports of the regulatory staff and the Advisory Committee on Reactor Safeguards (ACRS) on nuclear power plants. This report by the regulatory staff is intended to provide a general explanation of the nature of ATWS, to outline the results of the staff evaluations, and to provide guidance for applicants and licensees.

The first part of ATWS, "anticipated transients," is concerned with various events that may happen during the operation of a water-cooled reactor power plant. These deviations from normal operating conditions are called "anticipated transients," and might occur one or more times during the service life of a plant. They are thus distinguished from "accidents," which have a much lower likelihood of occurrence. There are a number of anticipated transients, some of quite trivial nature and others that are more significant in terms of the demands imposed on plant equipment. Anticipated transients include such events as a loss of electrical load that leads to closing of the turbine stop valves, a load increase such as the opening of a condenser bypass valve, a loss of feedwater flow, and a loss of reactor coolant flow. Nuclear power plants are designed with various safety and control systems to preclude adverse effects from these and other anticipated transients.

The other part of ATWS, "without scram", is concerned with the reactor protection system. The reactor protection system, or shutdown system, involves numerous instruments, cables, amplifiers, switching devices, alarms, trips, control rods and drive mechanisms, etc. The protection system is arranged to detect off-normal conditions in the plant and to institute automatically whatever safety action is needed. If plant conditions indicate there is a potentially damaging situation, the automatic reaction of the protection system is to cause the control rods to move rapidly into the reactor core to shut down the nuclear reaction. This most drastic form of automatic response of the protection system, which results in a very rapid shutting down of the reactor, is called the "scram." In some of the anticipated transients, shutting down the nuclear reaction and hence rapidly reducing the amount of heat being generated by the reactor core, is an important step in assuring that no damage to the plant or risk of accident occurs. If such a transient should occur and if, in spite of all the care built into the reactor shutdown system, a scram should not result, then an ATWS event would have occurred.

The background of ATWS as a possible source of concern in nuclear power plants goes back some years to discussions of the ACRS, the regulatory staff, and reactor instrument designers about the safety implications of interactions between normal control system circuitry and protection system circuitry in the instrument systems of power plants. After considerable discussion, and some design changes, it was determined that separation of control and protection functions was being achieved to a reasonable degree, either by physical separation or by electrical isolation. The focus of interest with regard to instrument systems then shifted to the ability of the shutdown system to function with the needed reliability. Reactor instrument designers carried out analyses of various kinds of failures. Reports on these analyses have been published and are available in the Commission's Public Document Room (1717 H Street, N.W., Washington, D.C.).[1,2,3,4,5,6*]

It became clear that failures caused by equipment wear-out or failure occurring on a random basis in protection systems, would not cause appreciable deterioration of reliability because of the redundancy of the systems. The staff concluded that for random component failures or malfunctions, protection systems of current design were adequate. It was not so clear, however, that these systems were sufficiently invulnerable to what are called "common mode failures."

In the simplest form, common mode failures could be a result of environmental conditions that affect all of the instruments of a particular type. More subtle forms of common mode failure have to do with design or maintenance errors that might be made for similar redundant independent portions of a protection system. One of the difficult aspects of deciding whether or not common mode failures were being adequately accounted for in shutdown system design was that techniques to analyze a system for common mode failures were not as well-developed as techniques to analyze a system for random failures.

In February 1969, an ACRS consultant pointed out that if common mode failures could reduce the reliability of protection systems in such a way that the system might not function properly in the event of an anticipated transient, there was the possibility of a safety problem with current designs. Since early 1969, the ACRS and the regulatory staff have continued their discussions and analyses of possible ATWS events. These efforts have taken two general directions. The first was concerned with attempting to evaluate the likelihood of common mode or other failures of reactor protection systems that might lead to ATWS events if the appropriate transients should occur. The second was to assume, simply as a basis for discussion, that ATWS was possible and to examine the consequences of various postulated ATWS events.

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* References are from the WASH-1270 Study and given at the end of this section.

The staff's preliminary results on ATWS were discussed with the ACRS in September 1970, and possible new requirements for instrument system designs were discussed. Analyses of the consequences of postulated ATWS events were requested of reactor designers after the September 1970 discussions, and all of the designers made these analyses. Reports of this work have been published and are available in the Commission's Public Document Room.[7,8,9,10,11] Some analysees submitted to the staff contain proprietary information.[12,13,14]

After reviewing these analyses of postulated ATWS events, the staff concluded that several anticipated transients in boiling water reactors would require prompt action to shut down the reactor in order to avoid serious plant damage and possible offsite effects. In pressurized water reactors, several anticipated transients would require rapid shutdown of the reactor to avoid pressure surges in the primary system that might, in some cases, exceed allowable limits. The staff found that the great majority of postulated ATWS events did not lead to serious consequences, but that design changes to improve protection against ATWS would be appropriate in anticipation of the large numbers of plants expected in the future.

In August 1971, the ACRS and the regulatory staff concluded that a design change to the proposed Newbold Island boiling water reactor units was appropriate to limit the possible consequences of ATWS. The same design change has been made for a number of boiling water reactor plants.

In April 1972, the staff transmitted to the ACRS a proposed set of positions and actions to be taken to implement the conclusions of the staff and ACRS studies on ATWS.[15] The ACRS replied in May 1972,[16] noting that:

"The ACRS recognizes ATWS as a low probability event. Nevertheless, it believes that, in consideration of the large number of water-cooled power reactors expected eventually to be in operation, and in view of the expected occurrence rate of anticipated transients (collectively, on the order of one per reactor year), experience with scram systems of current design is insufficient to give assurance of an adequately low probability of occurrence of an ATWS event of possibly serious consequence. Accordingly, the Committee agrees with the intent of the ATWS position recommended...."

In January 1973, as a result of further review and discussion of the various aspects of ATWS, the staff transmitted to the ACRS an amended proposed position on the need for protection against ATWS for water-cooled power reactors.[17] The amended position extended and elaborated the earlier staff position forwarded to the Committee. The ACRS responded in April 1973, agreeing with the amended position.[18] This double exchange of letters between the regulatory staff and ACRS has been made public and is available in the Commission's Public Document Room.

The thrust of the staff's amended position is that in view of the difficulty of verifying the needed reliability of reactor shutdown systems, and since larger safety margins are appropriate as increasing numbers of power reactors are built and operated, design improvements should be made to reduce the probability of ATWS in new plants to a negligible level, and to make the consequences of ATWS acceptable in plants now being constructed and placed in operation. It is fair to note that reactor instrument designers and plant operators believe that current designs of reactor shutdown systems are adequate for the purpose and that no upgrading in design needs to be done to deal with possible ATWS events.

The staff intends to implement the amended licensing position set forth in its January 1973 letter, setting October 1, 1973, as the effective date of the position. Analyses for older operating plants should be provided by October 1, 1974, and the need for any changes will be considered by the staff on a case-by-case basis. Plants recently started in operation, now under construction, or for which applications for construction permits are filed before October 1, 1976, should have any equipment provided and any changes made that are necessary to make the consequences of ATWS acceptable. Analyses of the effects of ATWS and plans and schedules for any changes found necessary should be provided for these plants by October 1, 1974, or at the time of submission of an application for a construction permit, whichever is later. Plants for which applications for construction permits are filed after October 1, 1976, should have improvements in the protection system design that make the chance of an ATWS event negligibly small.

The staff believes that the measures indicated for protection against ATWS and the schedule for implementing them take reasonable account of present and future needs for adequate safety margins, and that these measures would provide substantial additional protection of the public health and safety. The present likelihood of a severe ATWS event is considered by the staff to be acceptably small, in view of the limited number of plants now in operation, the reliability of current protection system designs, and the expected occurrence rate of anticipated transients of potential safety significance. As more plants are built, however, the overall chance of ATWS will increase and the staff believes that design improvements are appropriate to maintain and to improve further the safety margins provided for the protection of the public.

II. General Safety Basis

In establishing the boundary between accident sequences that are to be within the design basis envelope, and hence for which engineered safety features are provided, and accidents that reasonably may be assigned to that small residuum for which no further protective features are considered necessary, the regulatory staff uses the safety objective that the risk to the public from all reactor accidents should be very small compared to other risks of life such as disease or natural catastrophes. The staff believes this safety objective is met by requiring a design basis accident envelope that extends to very unlikely postulated accidents, and by establishing the further objective that accidents not included in the design basis envelope should have an average recurrence interval of at least a thousand years for all nuclear plants combined.

For an anticipated population of about one thousand nuclear plants in the United States by the end of the century, the safety objective will require that there be no greater than one chance in one million per year for an individual plant of an accident with potential consequences greater than the Part 100 guidelines. Since plants now being designed and constructed are expected to have service lives approaching 40 years, and may thus be part of the century-end population, the staff believes it appropriate to consider their designs in the light of this future requirement. In view of the difficulty of determining such a low probability, the staff regards this number as an "aiming point", or design objective, rather than as a fixed number that must be demonstrated for a given plant design.

III. Reliability Requirements, Failure Rates, and Test Intervals

The essential questions with regard to ATWS are the possibility of occurrence of ATWS and the nature of the consequences if it should occur. This section deals with part of the possibility question, namely the reliability required of the protection or shutdown system to reduce ATWS to an acceptable risk, and the relations between protection system reliability, failure rate, and testing interval. Subsequent sections discuss design aspects of protection systems and the failure analyses that have been made, the available experience record, and the various anticipated transients and their consequences if the protection system does not respond adequately.

The design objective for ATWS events that have significant consequences is to achieve a probability of occurrence consistent with the overall safety objective for an individual plant, considered on the basis of the large number of nuclear plants anticipated to be in operation in a few decades, of less than one chance in one million per year of a serious accident. The probability of an ATWS event is equal, for all practical purposes, to the product of two other probabilities. The first of these is the probability of an anticipated transient that may have serious consequences if the reactor is not shut down promptly. The second factor of the product is the conditional probability that the protection system will not respond adequately if challenged. The conditional probability that the protection system will not respond if challenged, i.e., that the protection system is in a failed condition at the moment of challenge, is also called the "unreliability" of the system. Letting A be the probability per year of the ATWS event, P_A the probability per year of an anticipated transient of potentially serious consequences, and U the unreliability of the protection system, we have

$$A = P_A \, U.$$

The safety objective is that the likelihood of all accidents with significant consequences not included in the design basis envelope should not be greater than one chance in one million per year, i.e., should not occur with a failure rate greater than 10^{-6} per year. For the particular potential failure path of ATWS, the staff believes that a failure rate of the order of one tenth of the overall safety objective is an appropriate objective. Thus,

$$A \sim 10^{-7},$$

so we require

$$P_A \, U \sim 10^{-7}.$$

The probability per year of anticipated transients varies with the type of nuclear plant and with the type of transient. For pressurized water reactor plants the transients having the greatest apparent potential for severe consequences are loss of all feedwater flow and, in some plants, loss of condenser vacuum. The staff believes these transients may be expected to occur once in five to ten years for a plant. For boiling water reactor plants, turbine trip, or other events causing the main steam line valves to close, may be expected to occur once in two to four years. In net, the annual likelihood of an anticipated transient of significance for ATWS is believed to be in the range 0.1 to 0.5, and it appears prudent to assign unit annual probability for such occurrences.

VI. ATWS Analyses

At the request of the staff, the designers of water-cooled nuclear plants have performed analyses of various anticipated transients with the arbitrary assumption that control and safety rods do not move into the core during or after the transients. No prejudgment was made as to the likelihood of these events; rather, the aim was to determine whether the consequences of ATWS were potentially severe enough to require further measures should the reliability of reactor shutdown systems be judged to be less than the design objective. In analyzing each transient, the designers were asked to assume that all other systems would react normally unless the consequences of the transient would make them inoperative. The reactor operating conditions and parameters of the transient, e.g., power level, flow rate, pressure, power distributions, and feedback coefficients, were to be those normally anticipated for the reactor state under consideration. The course of each transient was to be followed in the analysis until terminated, a condition defined as one with the reactor at essentially zero power in a coolable geometry, with normal afterheat removal systems in operation and containment pressure within design limits.

The anticipated transients analyzed for pressurized water reactor plants were as follows.

1. Loss of Electrical Load. These transients included generator trip, turbine trip, and loss of condenser vacuum.

2. Load Increase. These transients covered the accidental opening of the largest secondary valve that could increase steam flow, i.g., a turbine bypass valve.

3. Loss of Feedwater. These transients included loss of one feedwater pump or the closing of one feedwater valve, and loss of all feedwater pumps or the closing of all feedwater system valves.

4. Loss of Primary Flow. These transients included the loss of one primary coolant pump and the loss of all pumps.

5. Loss of Normal Electrical Power. This transient covered the simultaneous loss of power from the unit generator and from the offsite transmission grid, leaving the reactor with the onsite emergency diesel generator sets functioning as the only source of electrical power.

6. Inactive Primary Loop Start-Up. This transient was to be analyzed unless start-up of an inactive loop was precluded by protection system-grade interlocks.

7. Rod Withdrawal. These transients included control rod withdrawal from zero power, hot critical condition, and from full power.

8. Primary System Depressurization. This transient covered the opening of the largest single safety or relief valve in the primary system or any combination of such valves that could open together due to a single fault.

9. Boron Dilution. This transient involved malfunction of the boron concentration control system, with the most rapid possible dilution of boron in the primary system.

10. Small Line Break. This transient covered the failure of an instrument, drain, or sampling line connected to the primary system.

The corresponding list of transients considered for boiling water reactor plants is as follows.

1. Primary Pressure Increase. These transients included loss of load events such as generator trip, turbine trip, and loss of condenser vacuum. Also considered were such transients as closure of one or all of the main steam line isolation valves, and malfunction of the reactor primary system pressure regulator causing increasing pressure.

2. Reactor Water Inventory Decrease. These transients included events leading to a decrease in the inventory of reactor primary coolant such as loss of auxiliary power, loss of feedwater, pressure regulator failure in a direction to cause decreasing reactor system pressure, inadvertent opening of a safety or relief valve, and opening of condenser bypass valves.

3. Reactor Coolant Flow Increase. These transients included events that might increase the recirculation flow and thus induce a positive reactivity increment. They included a malfunction of the recirculation flow controller in a manner to cause increasing primary coolant flow and the start-up of a recirculation pump that had been on standby.

4. Reactor Water Temperature Decrease. These transients included events that might cause a power surge by reduction of the reactor primary coolant water temperature. They included malfunction of the feedwater control in a direction to increase feedwater flow, loss of a feedwater heater, shutdown cooling malfunction, and inadvertent activation of auxiliary cold water systems.

5. Reactivity insertions. These transients included rod withdrawal transients from zero reactor power, hot critical condition, and from full power; fuel assembly insertion, control rod removal and control curtain removal errors during refueling.

6. Primary Coolant Flow Decrease. These transients included failure of one or more recirculation pumps or malfunction of the recirculation flow control in a direction to cause decreasing flow.

The analyses performed by the reactor designers show that for transients in which the heat removal systems of the plant are not greatly affected, the consequences of the transients without scram occurring are not particularly severe.[7,8,9,10,11,12,13,14] After some period of off-normal operation, the plant stabilizes and can be shut down without damage. For those transients where the heat removal systems are affected, however, the potential exists for significant damage. This is not a very surprising conclusion. The primary assumption of the analysis, that the rods do not move into the core in the event of the transient, means that if the reactor is at full power, it will continue to generate substantial power during the transient. If the transient involves the interruption of the normal process of energy removal from the reactor, then the energy being generated in the core must appear as increased temperature and pressure in the reactor coolant system. In some cases the pressure increase may be great enough to raise questions as to the integrity of the reactor coolant system.

For pressurized water reactor plants the transients with the greatest potential for damage are the loss of feedwater and certain loss of load transients occurring with the reactor at full power. Loss of feedwater flow could occur as the result of malfunctions of the interlock and supervisory circuitry controlling the feedwater or condensate pumps or valves. The sequence of events for a typical pressurized water reactor plant in the event of a loss of feedwater transient without reactor scram may be summarized as follows. An accidental trip of the feedwater or condensate pumps or valves would cause a rapid reduction of feedwater flow. Low feedwater flow compared to steam flow, in coincidence with low steam generator water level, would initiate a reactor scram signal. This scram signal is ignored as part of the assumptions in these analyses, as are three or more subsequent reactor scram signals generated as the transient proceeds. The loss of feedwater flow to the steam generator secondary side would result in a drop in water level in the steam generator. A falling water level in the steam generator results in reduced heat transfer from the primary system. The primary coolant temperature would begin to increase since reactor power would remain high, and this, in turn, would cause the primary pressure to increase. The auxiliary feedwater pumps would be started automatically after the main feedwater pumps or condensate pumps were tripped. However, the auxiliary feedwater pump capacity is not large enough to remove all of the heat being generated in the core; consequently, the steam generator would boil dry. The primary system temperature and pressure would continue to increase and the primary safety valves in the surge volume of the pressurizer vessel would open and discharge steam. The increasing temperature of the primary coolant causes expansion of the coolant and the water level would rise in the pressurizer. When the pressurizer vessel became filled completely with water, the safety valves would discharge water instead of steam, but at a rate less than required to keep the primary system pressure from rising sharply. The reactor power would decrease throughout the transient because of the negative reactivity feedback arising from increased water temperature and reduced density. This effect, combined with heat removal by the auxiliary feedwater system and with the discharge of water through the pressurizer safety valves would reduce the pressure. The pressurizer safety valves would then close and steam would reappear in the pressurizer dome. If the primary system survives the pressure peak, which is estimated by the various analyses of the designers to reach values between 3000 and 7000 psi, heat generation in the core would be reduced and the heat removal capacity of the auxiliary feedwater system on the secondary side of the plant would cool the core and prevent further pressure increase. Lower pressure in the primary system would allow boron solution injection into the primary system initiated by a safety injection signal generated by low pressure in the secondary steam line or by manual actuation. When boron solution reaches the core, enough negative reactivity is provided to shut the plant down.

The loss of electrical load transient could occur from a generator trip, a turbine trip, or loss of the turbine condenser vacuum. Generally, the most severe transient would be caused by the loss of condenser vacuum. The main feedwater pumps in many plants are steam turbine-driven and exhaust to the main condenser. Thus, loss of condenser vacuum also could cause a loss of the main feedwater pumps. In this case the sequence of events would be similar to the loss of feedwater transient. The severest effect of the transient, the pressure surge in the primary system, would be of about the same magnitude as in the loss of feedwater flow transient.

For boiling water reactor plants, the transients having the greatest potential for significant damage are those leading to a reactor primary coolant system pressure increase. The most severe of these are the loss of condenser vacuum and the closure of all main steamline isolation valves. A loss of condenser vacuum causes automatic closure of the turbine stop valves and the turbine bypass valves. The turbine stop valves are fast-acting valves, so that there is an abrupt interruption of steam flow from the reactor. The main steamline isolation valves are slower in closing, but in this case the large steamline volume is not avaliable to buffer the pressure rise. The result in either case would be an increase in primary system pressure and temperature. The pressure increase would decrease the volume of steam bubbles in the reactor core and this, in turn, would increase the reactivity and cause a surge in reactor power. The power surge would cause a further increase in system temperature and pressure, with the pressure rising to values above acceptable limits. The other transients that lead to primary system pressure increase are less severe. Generator or turbine trips are less severe because the turbine bypass valves can be assumed to open and the condenser to be operative. Although the transient proceeds more slowly in these cases, the result still would be an excessively high reactor coolant system pressure.

4.2 SEPARATION OF PROTECTION & CONTROL

S. H. Hanauer, who became an ACRS member in 1965, was the first member to have a strong personal background of experience in reactor instrumentation and control. He had worked in the subject for many years at Oak Ridge National Laboratory (ORNL), prior to becoming a Professor at the University of Tennessee, and brought to the ACRS not only much practical knowledge about the difficulty of getting reliability from components like motors and valves which are actuated by instrumentation, but also a strong personal conviction that in good reactor design the safety systems which were provided to shut the reactor down should not be interconnected to the systems used to control the reactor. This was a philosophic approach, like "separation of chruch and state", which Hanauer and others at ORNL* had developed as the sound way to avoid unexpected, hidden interactions which might negate safe reactor shutdown, if some common usage or interconnection between the control instrumentation and safety instrumentation was part of the design approach. There existed a history of failures to support this point of view, and hanauer began raising the matter in connection with specific reactor projects under review by the ACRS in 1966 and 1967. The ACRS specifically identified the issue in its report on Point Beach in May, 1967.

The four light water reactor vendors each had adopted their own approach to instrumentation design, and it was especially the Westinghouse practice of using the same sensors and subsystems for both control and protection (safety) purposes that troubled Hanauer.

It had been customary on many of the earlier reactors for the reactor instrumentation design to be unavailable for review at the construction permit stage, and the Regulatory Staff and the ACRS had accepted this practice.

In connection with review of the Diablo Canyon Unit 1 PWR in the fall of 1967, Hanauer noted that Westinghouse had still not submitted an instrumentation design, although they proposed to submit reports giving some details by the end of 1967. He felt it would not be wise to approve still another reactor on this basis. Westinghouse was therefore prompted to provide partial information on the design during the ACRS review of Diablo Canyon. At the 92nd meeting, December 7-9, 1967, the ACRS tried to complete its construction permit review for Diablo Canyon. The site was relatively remote, and, at that time there were no known nearby large faults, so that the seismic design basis was not a matter of controversy. The Diablo Canyon reactor was to be one of the first of the highest power PWRs built, however, which made it a logical reactor on which not only to look for new, previously unanticipated issues, but to resolve some that had been ongoing.

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*Epler remarked to this effect in his comments on the proposed General Design Criteria.

At the 92nd meeting, Dr. Hanauer noted that Westinghouse did not meet the General Design Criterion which asked for two emergency cooling systems, preferably diverse. More importantly, he also noted that the control and safety instrumentation were intermingled. Westinghouse claimed their design met the proposed new IEEE - 279 standard criteria for Nuclear Power Plant Protection Systems; but Hanauer had major reservations concerning the adequacy of the proposed Westinghouse protection system design, and in the Committee caucus did not think a letter favorable to construction of Diablo Canyon could be written.

The matter received considerable discussion at the 92nd meeting, and the ACRS finally wrote a report (dated Dec. 7-9, 1967) which included the following paragraph:

> The Committee believes that control and protection instrumentation should be separated to the fullest extent practicable. The Committee believes that the present design is unsatisfactory in this regard, but that a satisfactory protection system can be designed during the construction of the reactor. The Committee wishes to review an improved design prior to installation of the protection system.

The Prairie Island case represented the next Westinghouse PWRs to be reviewed for a construction permit. The Regulatory Staff, who had previously accepted the proposed Westinghouse design, asked the Pririe Island Applicant to respond to the ACRS paragraph on Diablo Canyon. Only one minor design change was proposed by Westinghouse.

In their safety evaluation report to the ACRS on Prairie Island, the Regulatory Staff proposed the following criterion.

> As an absolute minimum, each variable monitored for protection should be instrumented by sufficient channels independent of control to meet the single failure criterion (generally this means three channels connected in two of three coincidence). Furthermore, the applicant may elect to provide additional channels of protection which are not independent of control. If he elects to follow this route, the applicant should provide a rigorous failure mode analysis to show that there can be no interaction between the control system and the independent protection channels through the shared channel(s).

There was considerable discussion with the Staff and Westinghouse about the matter, following which the ACRS provided the following comment in its report of March 12, 1968 on Prairie Island.

> The applicant has proposed using signals from protection instruments for control purposes. The Committee continues to believe that control and protection instrumentation should be separated to the fullest extent practicable. The Committee believes that the proposed protection system can and should be modified to eliminate or reduce to a minimum the interconnection of control and protection instrumentation. The modified system should be reviewed by the Regulatory Staff.

In the next few months, the ACRS wrote construction permit reports on other Westinghouse reactors, including Surry and Kewaunee, with a similar recommendation on separation of protection and control.

In the meantime, Westinghouse continued to argue that their design met the proposed IEEE criteria and was adequate. However, they prepared a somewhat modified design which they posed as a possible alternate at a meeting with the Regulatory Staff on May 3, 1968. They said they would offer their customers a choice of the original design and the modified design, but they would recommend the original as providing better control. In a letter dated May 30, 1968 to the ACRS, Committee consultant Epler found the modified Westinghouse design to still pose problems of interconnection of safety and control, and recommended further steps to reduce such interconnection. He appended a draft article entitled "Identical Systems for Protection and Control" in which he reviewed a bad previous history with such systems.

On May 8, 1968 the ACRS held a briefing on Control and Safety Instrumentation, at which experts from Argonne, Brookhaven, Los Alamos and Oak Ridge National Laboratories gave their opinions. The ORNL representatives explained their reasons for feeling that separation of control and protection was important, particularly in the neutron flux (power) systems. They acknowledged, nevertheless, that some mixing was tolerated under certain conditions even in the ORNL designs.

All the experts agreed that system designs which purposely intermix control and protection must be carefully designed and analyzed. However, there ws not a unanimous position in favor of full separation; nor was there agreement that adequate analysis of a mixed design could be accomplished.

One problem facing the ACRS was the implied threat by Westinghouse that if forced to go to complete separation, they would go to a single channel for control, which would lead to less reliable control and more plant transients with possibly adverse safety connotations.

In a memorandum to ACRS members dated May 22, 1968, Hanauer tried to present the problem as he saw it in summary fashion. The memo is reproduced on the following pages.

At its May, 1968 meeting, on the urging of member Mangelsdorf, the ACRS requested the Regulatory Staff to provide the Committee with a background paper on the overall subject, including the Staff point of view.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 22, 1968

MEMORANDUM

To : ACRS Members

From : S. H. Hanauer

Subject: INTERACTION OF PROTECTION INSTRUMENTATION WITH CONTROL

In this paper, I have tried to set down concisely my reasoning in the current discussions on Westinghouse designs.

1. The basic problem is to avoid having an event requiring protection action somehow inhibit the needed protection. In the classic example (HTRE-3), design defects in the neutron-flux instruments prevented an increase in current when the reactor power increased; the too-small current both told the servo to withdraw the rods and blinded the protection system to the resulting power increase. The core was destroyed.

It is for this reason that common use of a detector signal for both control and protection is automatically suspect.

2. The situation is complicated by the fact that three or four channels of instrumentation are provided for each protection variable, rather than just one. Random single failures do not fail such a system, because of the redundancy. However, such duplication does not prevent system failure due to design deficiencies, since all can be expected to fail for such a reason.

The use for control of signals from one, or two, or all the redundant protection channels is the point at issue.

3. If the only postulated faults are random component failures, it is easy to show, using statistical analysis, that control use of protection signals won't hurt you. A single failed channel causes an excursion; the other channels, assumed unaffected by the fault, provide the needed protection. The channels must be truly independent for this assumption to be valid.

This reasoning is the origin of the IEEE criterion (Section 4.7) that for such a failure the unfailed channels must meet the single-failure criterion. (The present Westinghouse designs do not meet Section 4.7 in every respect.)

4. In my opinion, random component failures are not the only postulated faults against which a protection system should have a defense. We learn from the history of various accidents, incidents, and near-misses that mistakes are made in design, execution, and operation of instrumentation systems. Such non-random (systematic) failures are not amenable to statistical analysis. They are the reason for employing diversity in protection-system design; one hopes that if a mistake is made, it will not involve dissimilar types of equipment.

The use of redundancy doesn't reduce the probability of occurrence of systematic failures, so the redundant nature of the protection signals does not in this instance justify their use for control.

5. As a minimum position, the regulatory staff has proposed requiring enough protection channels completely independent of control to satisfy the single-failure criterion; usually, this number is three. Additional channels in which control and protection could be mixed would also be allowed, provided they could be demonstrated independent of the sacred three.

This is a strict interpretation of the meaning of IEEE Section 4.7. Westinghouse does not agree that the staff position is the correct interpretation.

My own feeling is that mixing control and protection by using the same signal for both functions is wrong in principle, but that the proposed staff position is probably as far as one can go in requiring separation of a recalcitrant applicant.

6. "Protective overrides" are <u>control</u> functions, not protection, put in for the laudable purpose of forestalling the protection action by initiating a milder action first. Examples are (a) blocking rod withdrawal on high flux to inhibit a potential increase which, if continued, would require a scram; (b) shutting off feedwater on high boiler water level to avoid carrying a slug of water to the turbine, which could cause a turbine trip and scram. Overrides fall into two classes: blocking rod withdrawal can't cause an excursion requiring protection-system function, but shutting off the feedwater surely can. Thus it should be allowed to use redundant signals from the protection system to block rod withdrawal, but not to shut off feedwater.

7. Let there be no illusion; the protection system cannot be made "completely independent" of control. The two systems related to the same reactor plant, are located in the same control room, are watched and manipulated by the same operator, are serviced by the same maintenance men, have cables in the same containment. What I am objecting to is the deliberate use of signals from protection instruments for control purposes, leading to the potential that faults could initiate an excursion and consequentially cripple the needed protection for that excursion, to the detriment to health and safety of the public.

8. As a postscript, it seems worth pointing out that concern should also exist where identical devices, albeit independent, are used for both protection and control. A design, maintenance, or operating error in such identical components, or their exposure to a deleterious, unusual environment, would have the same effect as their interconnection. Thus, diversity as well as independence should be required of protection and control instrumentation. This matter is in urgent need of further study.

cc: R. F. Fraley
E. P. Epler

Such a paper was prepared and the subject was discussed in some considerable length at the 98th meeting, June 5-8, 1968.* The Staff report, which provided an excellent review of the matter, concluded as follows:

> In summary, we believe the determination of which approach is preferable depends on a judgment as to whether protection channels independent of control are required to protect against subtle control and protection system interactions.
>
> We have been asked what the DRL position on control and protection interaction would now be if the ACRS comment on Diablo Canyon had not been made. Since we recommended approval of a number of designs based on the IEEE criterion during the year before the Diablo Canyon letter, we would probably still be using the IEEE criterion. Our deliberations since the Diablo Canyon letter have, however, made us question one of our recommendations on Diablo Canyon. We recommended acceptance of certain scram channels which did not meet IEEE-279 based on their not being required for safety. This gives rise to what have been called first class and second class scrams. This could cause confusion as to which protection channels should be relied on for safety and which should not be relied on. The present DRL criterion takes care of this situation. If however, we returned to using IEEE-279, we would attempt to develop criteria for channels used for equipment protection rather than safety. We also will have to develop criteria for protective

Subsequent to the meeting with the Regulatory Staff, the ACRS considered three alternative positions.

1. The IEEE single failure position plus overrides.

2. The Regulatory Staff position on Prairie Island where some channels are permitted to be interconnected while others are independent.

3. Separation to the maximum extent practical.

Interestingly, members Hanauer, Hendrie and Zabel, who had the most experience with such systems, all indicated they would prefer position 3, that is separation to the maximum extent practical if starting anew, but that due to other considerations, they did not recommend that choice for the Committee position.

The ACRS finally agreed to a modified position, as follows:

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*It was noted that the British and Canadians require separation of protection and control, also, that General Electric employs it.

> The Committee believes that systematic, non-random, concurrent failure of redundant elements should be considered in the design and review of reactor instrumentation system. Systematic failure of a protection system and, particularly, common failure modes of control and protection system, are of concern. The Committee suggests that the Staff consider supplementation of the IEEE criteria for use in review of instrumentation, and propose criteria which take the possibility of systematic failure into account.

On July 5, 1968 the Regulatory Staff provided a short discussion paper in which they agreed that additional design objectives needed to be developed which treat the design of protection systems in regard to potential sources of systematic failures more adequately than IEEE-279. The Staff outlined an approach they planned to follow in order to pursue the matter.

At the 100th meeting, August 8-10, 1968, the Regulatory Staff provided another status report with some specific proposed courses of action. ACRS members Hanauer and Mangelsdorf took strongly opposing views on the Staff proposals, and the minutes of the 100th meeting record the following Committee action (Note that DRL refers to the Division of Reactor Licensing in the AEC).

<u>EXCERPT FROM SUMMARY OF 100TH ACRS MEETING</u>

<u>AUGUST 8-10, 1968</u>

<u>EXECUTIVE SESSIONS</u>

4. <u>Separation of Protection and Control Functions of the Instrumentation and Control Systems</u>

   DRL requested ACRS endorsement or comments on their status report, received August 7, 1968. The Committee agreed that at this time it could only endorse that portion which says, "The basis of the approach proposed in our (DRL) July 1968 report is that the potential for systematic failures should be considered in protecting against all accidents and transients, not just those arising from control system action. Thus we (DRL) plan to examine the benefits of functional diversity for all accidents and transients."

   Dr. Morris was informed of this ACRS conclusion by telecon on August 12, 1968.

   An ad hoc Subcommittee consisting of a neutral Chairman, both protagonists and such other members as the Chairman of the ACRS may select will be appointed to review the DRL position and make a recommendation to the full Committee. This item will appear again on the September agenda.

An Ad-Hoc Subcommittee consisting of Hanauer, Mangelsdorf and Hendrie (serving as Chairman) met on September 4, 1968 to review the matter. The Russelville plant was on the ACRS agenda for September, so the discussion was undertaken with that in mind.

The minutes record that several possible alternative positions were identified, as follows:

ALTERNATE ACRS POSITIONS REGARDING SEPARATION OF CONTROL AND SAFETY

Russellville	Tell Staff
1. Accept applicant's proposal	1-a. Interconnection per IEEE 279 is all right.
	1-b. B&W interconnection is all right; still studying the more complicated W system.
2. Repeat indecisive paragraph	2. Committee still considering the problem.
3. State that flux instruments must be separated at least as far as Prairie Island DRL position.	3-a. B&W must separate; still studying the W system.
4. State that separate flux instruments are required for control and protection.	4. Separate instruments must be provided.

Mr. Mangelsdorf inquired if interconnected systems, with required redundancy, would improve or detract from overall plant safety. Dr. Hanauer replied that there are advantages and disadvantages which must be evaluated for interconnected or separated systems. He noted that one arrangement (Figure 1) (proposed by the Regulatory Staff for Prairie Island) improves plant operability. Its effect on safety is not clear however. Other interconnections (Figure 2) however actually reduce plant safety.

As one example of an interconnection that has reduced safety he described an incident at MTR where two diverse and independent liquid level indications were provided for the head tank level. A head tank position indicator and a "bubbler" were provided with one controlling the plant pumps and the other providing reactor scram. When one indicator failed, both the control feature and the safety feature were connected to the same indicator. When this indicator subsequently failed, the safety and control functions were both disabled simultaneously.

The minutes record that the personal preferences of the Ad-Hoc Subcommittee members was as follows:

Member	Favored Position
Mangelsdorf	1
Hanauer	3
Hendrie	4

The summary minutes of the 101st meeting, September 5-7, 1968, show that with regard to Russelville, the Committee recommended that "the instrumentation system should be reviewed for common failure modes not considered in the single failure criteria. The applicant should show that the proposed interconnection of control and safety instrumentation will not adversely affect plant safety in a significant manner, considering the possibility of systematic component failure."

The minutes also record adoption of the following compromise recommendation to the Regulatory Staff.

101st Meeting, September 5-7, 1968

Separation of Protection and Control Functions of the Instrumentation and Control Systems

The Ad-Hoc Subcommittee presented several alternate positions representing the various schools of thought within the Committee with respect to separation of protection and control functions of the instrumentation and control systems. Dr. Okrent offered a compromise statement that was discussed and eventually adopted.

The Committee advised the Regulatory Staff:

a. For reactors under construction, and approved prior to Diablo Canyon, the ACRS believes it reasonable to accept designs in accordance with IEEE-279, modified as necessary for protective overrides and channels used only for equipment protection. In view of the desirability of having similar instrumentation systems in the units of a multi-reactor plant, Point Beach No. 2 may be considered on the above basis.

b. For Diablo Canyon, and reactors approved subsequently, the applicant should be required to show that any interconnection of control and safety instrumentation will not adversely affect plant safety in a significant manner when considering the possibility of systematic component failure.

c. The Committee reiterates its belief that systematic, non-random, concurrent failure of redundant elements should be considered in the design and review of reactor instrumentation systems. Systematic failure of a protection system and, particularly, common failure modes of control and protection systems are of concern. The Committee suggests that the Staff continue its program to supplement the IEEE criteria for use in review of instrumentation, and to develop criteria which take the possibility of systematic failure into account.

This remained the position of the ACRS for the next several meetings. The emphasis had shifted from a request for as much separation of protection and control as practical to conscious effort to account for systematic (common mode or common cause) failures by appropriate design approaches.

712 Florida Avenue
Oak Ridge, Tennessee

January 21, 1969

RECEIVED

Mr. R. F. Fraley
Executive Secretary to the Advisory
Committee on Reactor Safeguards
U.S. Atomic Energy Commission
1717 H Street
Washington, D. C. 20545

Dear Sir:

It has recently been disclosed that the public is in jeopardy but for the protection afforded by the reactor shutdown system. I am convinced that this single line of defense is inadequate and therefore request that you bring this to the attention of the Committee.

At one time it was reasonable to consider that a serious reactor accident, occurring simultaneously with a breach in containment, would require a series of highly improbable events. This is no longer true; it is necessary only, in a BWR, that loss of electric load, a routine operating event, be followed by failure to scram.

G.E. told us in an Oyster Creek subcommittee meeting that loss of electric load occurs approximately once per year. Failure to scram had not been analyzed as such an analysis would be too "hairy," i.e., containment would not be possible. G.E. spokesmen further agreed that once per year the public is placed in jeopardy but for the prompt intervention of the reactor shutdown system.

In defense G.E. cited BWR operating experience which turned out to be no more than 20 reactor years or 20 successful operations of the shutdown system when challenged. G.E. had not considered whether 20 successes constituted a large enough number to demonstrate adequacy.

The AEC has not adopted a number for an acceptable uncontained failure rate. The accompanying paper, however, shows that the public is otherwise being led to believe that the rate will be 10^{-6} to 10^{-7} per year. To support this belief we should experience 10^6 to 10^7 successful interventions to demonstrate adequacy.

The industry by not attempting to mitigate the "China Syndrome" has placed the entire burden of protecting the public on the reactor shutdown system.. An informed segment of protection system specialists believes that in this area the very best practice would fall short by a factor of 10^2 to 10^3.

Ergen has stated, however, that in his opinion, the containment of a slumped core is "not entirely hopeless."

The attached paper, which was prepared as part of another program, leads to the conclusion that if conditions continue unchanged, we will experience an uncontained meltdown once per year or, at best, once per 10 years should 1000 reactors be placed in operation.

Very truly yours,

E. P. Epler

EPE:bls

4.3 ATWS - Part 1

In a letter dated January 21, 1969 to R. F. Fraley, Executive Secretary of the ACRS, Committee consultant E. P. Epler advanced the thesis that the reliance being placed on reactor safety systems in BWRs provides inadequate protection for the health and safety of the public. More specifically, Epler argued that reactor scram was needed to prevent core meltdown and a loss of containment integrity following a routine operating event such as loss of electric load, which might occur about once a year. Epler argued that a scram system unreliability of less than 10^{-4} per demand could not be expected because of systematic failures.

Epler's letter is reproduced on the following pages.

In the attachment to the letter, Epler mentions that public figures (Alvin Weinberg, Director of ORNL and Chauncey Starr, then Dean of Engineering at UCLA and former President of Atomics International) had publically indicated that the probability of a serious reactor accident was similar to that of a jet airliner plunging into Yankee Stadium during a World Series game, which Epler estimates s roughly 10^{-7}/year. However, because of the lack of measures to cope with the "China Syndrome," and because of his own estimate of a relatively high scram failure rate, Epler feels that the actual probability of a serious accident may be a factor of 1000 higher. He refers to a recent article authored by himself for the journal Nuclear Safety (Epler, 1968) and two papers by well known Canadian and English engineers in the field of reliability.

The matter raised by consultant Epler was placed on the agenda of the next ACRS meeting, the 106th held February 6-8, 1969, and Epler was present to discuss his concern. The Committee decided to refer the matter to a Subcommittee and provided copies of Epler's letter to the Regulatory Staff.

At the 108th meeting, April 10-12, 1969, the ACRS reviewed the construction permit applications for the Hatch Unit 1 BWR. The Committee took advantage of the occasion to have General Electric make a presentation on the matter of anticipated transients without scram (ATWS). The meeting summary does little more than record that a discussion took place and that General Electric felt their design was acceptable but that they had not factored systematic errors into a reliability analysis.

The ACRS did not complete action on Hatch Unit 1 at its April, 1969 meeting. At a Special Meeting, May 2, 1969 various matters on the Hatch application were discussed, and an Ad-Hoc Subcommittee Meeting on ATWS was arranged to be held with General Electric on May 7, 1969, immediately prior to the regular May full Committee meeting. The summary of the Subcommittee meeting, together with excerpts from the minutes are reproduced below:

Purpose

The Subcommittee met to hear GE's story on the consequences of a failure to scram and possible fixes if the consequences are unacceptable.

Summary

GE presented some preliminary information on their evaluation of present BWR designs and alternate designs, with regard to the consequences of expected transients and a failure of the reactor to scram. GE has not yet reached a conclusion on these matters and will continue their study and present the results in a topical report.

GE described the events that would ensue from a turbine trip if the reactor failed to scram for the present designs, with and without bypass, and then describe the events that would occur if; (1) the safety valve capacity was increased, (2) the bypass flow was increased, (3) the recirculation pumps were tripped, (4) the relief valves are opened in time to augment the bypass system. In all cases it was assumed that the turbine valve closed in 1/10 of a second. GE pointed out that scram of the reactor rods is initiated by, first, a 10% closure of the turbine valves, second, by 120% power level, and third, by high reactor pressure. To reach the condition of no scram, all three of these scrams must fail. In most cases the neutron flux went up over 600% immediately following the turbine trip, but was turned around by doppler broadening, and the power leveled out at some value higher than 100%. For each of these cases GE displayed curves showing the neutron flux, peak fuel temperature, fuel surface heat flux, water level, steam line pressure, vessel pressure, safety valve flow, and relief valve flow, all with respect to time. However, these were preliminary values and GE refused to leave copies for the Committee to review.

Mr. Levy assured the Regulatory Staff and the Ad Hoc Group that this information would be available as a topical report in a couple of weeks.

Executive Session

Dr. Okrent opened by stating the purpose of this discussion, and pointing out that in the current draft of the Hatch letter there is a paragraph referring to the need for action in the case of a turbine trip. He said that studies were needed to see if the reliability of these plants could be improved. He noted that common-mode failures or system failures have occurred that would prevent the reactor from scramming if called upon to do so. He also pointed out that there are known transients that result in a need to scram in order to prevent the release of fission products. He said that he felt the probability of an accident that released fission products in the range of Part 100 limits should be down to 10^{-6} or 10^{-7} in accordance with Mr. Farmer's views in the United Kingdom.

Dr. Hanauer stated that the turbine trip question was only an example. The main question was that of an anticipated initiating event that requires a scram to protect the public. He was looking at a class of events and the potential for a control failure. Until recently he had never considered bypassing as a safety function.

Meeting with the Regulatory Staff

Dr. Morris reported that the Regulatory Staff had planned and had scheduled a meeting with GE to discuss this problem on May 12. He agreed that it needed study and that the designer ought to look at the consequences and potential fixes for this type of accident. He said that GE felt that they should concentrate on improving the reliability of the scram system, however, GE has started to look at the consequences of a failure to scram. Dr. Morris stated that Mr. Price had advised GE that protection against failure to scram would not be required on the Hatch and Brunswick plants, that is, the current plants being considered.

Dr. Okrent asked if this would not be a bypassing of the intent of the Atomic Energy Act when the Regulatory Staff knew of a condition that could present a hazard to the health and safety of the public and did not require the applicant and designer to correct it.

Dr. Beck said that a rigorous interpretation of this would mean shutting down a number of operating reactors.

Dr. Okrent said that the Committee frequently identified problem areas but permitted the reactors to continue to operate.

Dr. Hanauer asked if there was a middle ground between giving this applicant a construction permit and not identifying this turbine trip failure to scram problem, or if the problem was identified would that preclude the issuance of a construction permit.

Dr. Morris said that there should be some middle ground in this area. Dr. Morris reported that GE was currently estimating the cost of backfit items at approximately $50 million per plant.

At the 109th meeting, May 8-10, 1969, the ACRS completed action and wrote letter reports on the Hatch Unit 1 BWR and on the application to build two BWRs at the Brunswick Station. In each report the ACRS included a paragraph as follows:

"A study should be made by the applicant of further means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram during anticipated transients."

Also, at the May, 1969 meeting the ACRS began asking questions concerning the effects of a ATWS on the PWR that happened to be in for review (Ginna, formelry called Brookwood). Thus, Epler's original question had broadened into a class of transients for BWRs, and was furthermore being examined for PWRs.

The broadening of the ATWS matter is illustrated by the results of the 111th meeting, July 10-12, 1969. The ACRS reviewed the construction permit application for Three Mile Island 2, a PWR. The Committee report of July 17, 1969, includes the "customary" paragraph with regard to common mode failures in protection systems; however, it also requests that a study be made of the possible consequences of anticipated transients without scram for the PWR.

The Regulatory Staff had been engaged for almost a year in studies of systematic failures in reactor protection systems, in response to the issue of separation of protection and control first brought into full view with the ACRS Diablo Canyon letter of December, 1967 (which later evolved into the concern with systematic failure as enunciated in the letter on Russelville of September, 1968). In a letter to ACRS Chairman Hanauer, dated August 4, 1969, Peter Morris, the Director of the Division of Reactor Licensing of the Regulatory Staff, provided a status report on the Staff studies of systematic failres. He indicated a preference for separating the study of systematic failures from that of ATWS in order to not delay the former, but noted that at a Subcommittee meeting some ACRS members and consultants indicated that they saw the two matters as strongly connected. Morris requested comment on his proposed plan of action, which involved having each vendor do separate (but related) studies on the two topics.

In a memorandum from Fraley to Harold Price, dated December 15, 1969, (reproduced on the following page) the ACRS accepted Morris' proposal but expressed concern with the "time required to resolve the safety questions."

We shall see that this memorandum was prophetic, and that time stretched out to roughly a decade without resolution of the matter.

While the Regulatory Staff began its study of ATWS, the ACRS continued to conduct relatively short reviews of the matter with each applicant and to insert a paragraph concerning the matter in each ACRS case letter. Thus, at the 113th meeting, September 4-6, 1969, the Dresden 2 operating license review included a comment to the effect that the Applicant was studying "further means of preventing common mode failures from negating scram action and of design features to make tolerable the consequences of failure to scram during anticipated transients."

At the 119th meeting, March 5-7, 1970, Combustion Engineering, the vendor for the Hutchinson Island (St. Lucie) reactor, promised to report to the AEC on common failure modes by June, 1970 and on ATWS by November, 1970.

At the 121st meeting, May 7-9, 1970, Babcock and Wilcox, the vendor for the Midland reactor, stated that "they had recently discussed with the Regulatory Staff the results of the B&W studies of various systematic failures. B&W added that the Staff had additional requirements regarding the studies, and, therefore, B&W would have to wait for a clarification of these requirements before continuing the studies. B&W had not

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 15, 1969

Mr. Harold L. Price
Director of Regulation

ACRS POSITION RELATIVE TO SYSTEMATIC FAILURES IN REACTOR PROTECTION SYSTEMS AND FAILURE TO SCRAM FOLLOWING ANTICIPATED TRANSIENTS

At the December 1969 ACRS meeting, the Committee adopted the following position relative to the matters of systematic failures in reactor protection systems and failure to scram following anticipated transients:

> "The ACRS agrees with the future direction of the study regarding systematic failures in reactor protection systems and failure to scram following anticipated transients, as outlined in the August 4, 1969 DRL report, "Systematic Failures Study Status Report." The Committee is concerned, however, regarding the amount of time which will be required to resolve the safety questions which may be involved. The problem of available manpower is recognized, and the Committee suggests that it may be necessary to obtain assistance from outside the regulatory group, although the best manner in which to obtain such assistance has not been determined."

R. F. Fraley
R. F. Fraley
Executive Secretary

analyzed a number of anticipated transeints, e.g., loss of primary pumps or loss of off-site power. The possible benefit of additional safety valves and of rapid injection of boron into the reactor moderator were mentioned by the Committee."

This unwillingness by B&W to exercise initiative in deciding what transients needed study and to rapidly develop the relevant information was generally characteristic of all the vendors, and contributed greatly to delay in the first few years of consideration of the ATWS.

At the 122nd meeting, June 11-13, 1970, during the review of Midland 1 and 2, B&W stated they would have a report on their analysis of failure to scram on anticipated transients in early 1971. The ACRS recommended in its report that "the applicant accelerate his study of means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram during anticipated transients." The Committee also noted that the applicant stated that the engineering design would maintain flexibility with regard to relief capacity of the primary system and of a diverse means of reducing reactivity.

At the 122nd meeting, the ACRS also held a general discussion of ATWS. The summary minutes are excerpted below.

> Failure to Scram During Anticipated Transients
>
> Dr. Hanauer summarized the information presented by the four NSSS vendors to the Instrumentation and Control Subcommittee on June 3, 1970. He stated that the consequences of a failure to scram on anticipated transients appears to be as follows for each of the NSSS vendors: (a) Westinghouse already has more pressure relieving capacity than is required by the pressure vessel code. It appears they can resolve this problem by adding slightly more relieving capacity. Another possibility is to provide a boron injection system; (b) B & W may have to provide more significant measures than Westinghouse because of features such as single-pass boilers vs. multi-pass boilers; (c) CE has not provided any information which can be evaluated as possible fixes; (d) GE has only a few seconds in which to mitigate a failure to scram. (GE believes that tripping out the recirculation pumps, when the anticipated transient occurs, would solve their problem.)
>
> The Committee decided to accept the "guide" (list of transients and associated conditions to be considered) proposed by DRL on anticipated transients for PWRs, dated June 1, 1970).
>
> Discussion was held of the following motion: "It is the Committee's position that failure of the single rod-scram-system, as presently provided, to scram on anticipated transients should not result in an accident worse than Part 100 for light water-cooled power reactors.

The Committee decided to consider the question of Anticipated Transients Without Scram (ATWS) with high priority ("case-like" review). To this end, the Committee intends to schedule discussion of the subject with the Regulatory Staff at the July meeting (one hour). Accordingly, the Regulatory Staff was requested to furnish a report to the Committee on this matter before the July meeting. Subjects should include:

1. Whether protection is required for ATWS.

2. Status and schedules of studies underway. (PWRs and BWRs)

3. Scram reliability required for ATWS protection not to be necessary.

4. Estimate of reliability of scram systems presently installed or under construction (expected reliability of current rod safety systems).

5. Possible fixes if ATWS protection is necessary; likelihood of success of fixes; advantages and disadvantages of fixes. (Discussion of possible measures which might make such transients lead to conditions not exceeding Part 100 under realistic, though conservative, assumptions of activity release; also the likelihood of success of such measures, and possible advantages and disadvantages.)

It is perhaps of interest to note that on May 4, 1970, ACRS consultant Epler, who had first raised the question for BWRs, wrote a letter to the ACRS in which he concluded that the PWR may also have a safety problem with ATWS. In this letter he recommended that criteria be developed for an additional reactor shutdown system for both BWRs and PWRs.

At the 123rd meeting, July 9-11, 1970, during the operating license review for Dresden 3, General Electric stated that they "need AEC criteria if more is required of them regarding ATWS." They found it "a problem to have to document their studies in the public record."

At the same meeting, during the Trojan construction permit review, Westinghouse stated that they "will issue a report which contains the results of their studies of ATWS before the end of the year. The Applicant can provide flexibility in his design into 1971 to accommodate fixes for ATWS.

The Committee report noted that the design would maintain flexibility regarding fixes for ATWS.

Dr. S. Hanauer, who had been an ACRS member for about five years while on the faculty of the University of Tennessee, was asked by Mr. Price, the Director of Regulation, to join the AEC Regulatory Staff on a full-time basis in mid-1970. Thus, the minutes of the July, 1970 meeting show Dr. Hanauer reporting on behalf of the Regulatory Staff to the ACRS of which he was no longer a member. Dr. Hanauer commented that "some information which is needed to resolve the ATWS matter is still lacking, but it is questionable how much more will be forthcoming." He stated that Westinghouse believes they need no fix and General Electric believes they have a fix (turn off recirculation pumps).

At the 124th meeting, August 13-15, 1970, during the construction permit review for North Anna, the ACRS expressed its concern over the slow pace of the common mode failure study and of the ATWS study. Westinghouse stated they did not believe that additional measures will be required; they agreed to incorporate plant changes, however, if the requirements were established by early 1971.

At the same meeting, during the Davis Besse review, B&W again said they were awaiting further guidance from the Regulatory Staff as to the transients to be considered.

At the 125th meeting, September 17-19, 1970 the ACRS discussed ATWS with the Regulatory Staff. The Staff had submitted a report in September on ATWS, which included the following recommendations:

(a) The reliability of scram systems of current design demonstrated to date and the occurrence rate of anticipated transients leads to the conclusion that anticipated transients without scram having serious consequences will occur at an unacceptably high rate when there are a large number of reactors in operation.

(b) The consequences of anticipated transients without scram should be shown to be acceptable or design changes should be made.

(c) Where changes are needed, they can be provided by improving the reactivity reduction systems or by modifying the plant so the consequences of not reducing reactivity are acceptable."

The Staff also recommended that:

(a) Applicants should be required either (1) to demonstrate that with their present designs the consequences of anticipated transients without scram are acceptable, or (2) to make design changes to improve significantly the reliability of the scram system.

(b) The BWR and PWR manufacturers should be informed of this decision.

(c) The analysis of ATWS in accordance with the guide (Appendix D) originally prepared for use by the reactor designers should be pursued with applicants. This should be done with applicants on current and future construction permit cases and with holders of construction permits for which the AEC safety evaluation and the ACRS letter identified this problem. Backfitting of other cases should be considered on a case by case basis.

However, at the September, 1970 meeting, the Committee apparently was not ready to endorse the Regulatory Staff recommendations, and decided it needed more discussion on the subject.

At the 126th meeting, October 27, 1970 there was a long discussion with representatives of North Anna, Trojon and Westinghouse, in which they "clarified" what they had meant by maintaining flexibility. Specifically, they said they were not maintaining flexibility for hardware changes; that were these to be required, the AEC would have to make a finding that such changes will provide substantial additional protection which is required for the public health and safety, in accordance with the AEC Regulation 10 CFR 50.109 - Backfitting. So that, except for possible changes which affect the reliability of the instrumentation and control system, the reactors were being proposed (and implicitly or explicitly accepted) as is, despite the lack of resolution of ATWS.

At the 126th meeting, the ACRS had available written comments from several consultants in the field of reliability who had attended an earlier Subcommittee meeting on scram system reliability. The comments were varied; when specific, they tended to support Epler's original thesis.

By this time, at least some ACRS members were getting concerned about the delay in resolving ATWS and the apparent loss of real flexibility to make future changes on plants currently receiving favorable construction permit reports. The minutes of the 127th meeting, November 12-14, 1970 record a Committee discussion on how to accelerate the pace. As of that time the Regulatory Staff had still not sent the list of questions to be answered to the four reactor vendors.

At the request of the Committee, ACRS member David Okrent prepared a short discussion paper on ATWS, which is reproduced on the following pages.

The minutes of the 127th meeting also note that member Okrent attached additional remarks to an ACRS report concerning a power level increase in the Oyster Creek BWR, one of them relating to ATWS. (The Committee had discussed this and some other "backfit-type" matters and chose to omit any reference to the matter in its report).

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 30, 1970

ACRS Members

A DISCUSSION OF ATWS*, PER REQUEST OF THE COMMITTEE AT THE NOVEMBER, 1970 MEETING

1. Safety Objective - The objective of a total probability of less than 10^{-7} per reactor year for a very bad accident (an accident much worse than Part 100) is postulated. For 1,000 reactors, this yields a probability of 1 in a 100 each century, and the U.S. now anticipates about 1,000 reactors by the year 2000. If this objective is missed by a factor of 100, there will probably be several very bad accidents in the world during the next century.

   There are a variety of ways (more than 10) whereby a very bad accident may occur. Therefore each of these must have a probability much less than 10^{-7} (say 10^{-8}) per reactor year, if the combined probability is not to exceed the objective of 10^{-7}.

   Thus, an objective of 10^{-8} per reactor year for a very bad accident from ATWS is suggested, with a need to meet this objective within a factor of ten.

2. The Staff report references various experts who have estimated an unreliability of scram between 10^{-3} and 4×10^{-4} per demand. At the ACRS Subcommittee meeting of August 26, 1970, General Electric stated that experience with GE reactors led to a failure probability of 8×10^{-4} with a 95% probability. It was stated that to demonstrate empirically an unreliability of 10^{-7}, approximately 300,000 reactor years with a zero failure history would be required.

   Frequent testing can improve the failure probability somewhat, say a factor of ten, for some common mode failures, but not all can be or have been detected by testing.

*The Staff report of September, 1970 on ATWS merits re-reading. This memo's major function is to call it (and the subject) again to your attention.

The Regulatory Staff concludes that the failure probability is much larger than 10^{-7}.

All of the ACRS consultants offering opinions agreed with the Staff and with the previously published opinions that the unreliability of current systems probably falls in the range of 10^{-3} to 10^{-4}.

3. Historically, this is now an old issue which represents, in part, the evolution of the issue of separation of safety and control which was first raised in an ACRS letter in December, 1967. The question of common mode failures followed, and ATWS, per se, was called directly to ACRS attention about two years ago in a letter from Dr. Epler, and soon thereafter in a publication in the Nuclear Safety journal.

   The first Subcommittee meeting with GE on ATWS was held 18 months ago.

   On or around June, 1970 the ACRS took the position that ATWS was a high priority item requiring resolution and asked the Staff for a report with recommendations. A draft Staff report was provided in early July. At that time questions were raised concerning how the Staff was handling probabilities - so a Committee decision was put off, and a Subcommittee meeting held in August.

   In September a formal Staff analysis was received, reaffirming the July draft. It concludes that scram reliability is far from adequate and that positive steps should be taken.

   The Committee took no action in September. In October the Committee approved the Staff's sending out a list of questions to vendors, but took no decision. The matter was referred to the vendor subcommittees to continue work on the ATWS question.

4. What information does the ACRS seek? Is each vendor subcommittee supposed to get new information which will provide a different basis on which to judge scram unreliability? If so, how? For example, what empirical information is there on which to judge B&W or CE power reactors, neither of which have yet operated?

   It appears that no vital source of new information which can contradict the Staff position on reliability has been identified. In fact, during the recent past, another failure has actually been experienced at Hanford and a partial failure at SEFOR thereby reenforcing the Staff position.

5. I would like to propose the following course of action to achieve what can be accomplished in a practical manner on a reasonable time scale:

A) The ACRS vendor subcommittees should not spend any significant amount of time in discussion of the reliability or unreliability of scram systems. Instead their efforts should be directed toward:

1) Establishing requirements for and possible improvements in the methods for initial and periodic testing of safety systems.

2) Establishing procedures for maintenance and periodic adjustment of safety systems that will minimize the introduction of common mode failures (e.g., adjust only a limited number of channels during any single shutdown, replace components in a limited fraction of installed channels until performance of the new components is demonstrated).*

3) Consideration of improvements in systems design that will further reduce the possibility of common mode failures such as:

a) Use of diverse components, functional signals, physical separation, etc. (Note: Westinghouse, B&W and GE reports regarding signal diversity have already been received. A report by CE has been promised).

b) Separation of control and safety systems.

B) In addition to A above, the vendors should be required to (1) demonstrate that with their present designs the consequences of anticipated transients without scram are acceptable or (2) describe design changes which render the consequences of ATWS acceptable.

C) The ACRS vendor subcommittees should plan to report to the full Committee within 6 months with respect to A & B above and the Committee should then recommend implementation of appropriate procedures and design changes as are considered necessary.

R Fraley
for D. Okrent

*This matter should also be reviewed with representative utilities.

At the 128th meeting, December 10-12, 1970, the ACRS discussed the ATWS matter further in Executive Session. The minutes of this discussion are reproduced below.

EXCERPTS FROM 128TH ACRS MTG. SUMMARY - 12/10-12/70

Executive Session

ATWS - The Committee discussed various ways to attempt to resolve the outstanding concerns about ATWS. The Committee did not believe it was necessary at this time to make plant changes to reduce the probability of failure to scram in the event of anticipated transients and to increase significantly, when appropriate, the probability that the consequences of ATWS will be tolerable.

The Committee decided to send a memo to Mr. Price which indicated the Committee position regarding ATWS:

1) Reliability of present protection (scram) systems cannot be accurately established at this time.

2) The Staff is urged to send out the ATWS questions to vendors and get answers expeditiously.

3) The Committee and Regulatory Staff should maintain a detailed and continuing review of present plants, including exploration of what fixes are possible if they are needed.

At the 129th meeting, January 7-9, 1971, the ACRS decided that an Ad hoc group should be formed to pursue the ATWS matter. The Committee also decided that the ATWS matter should be considered generically by the Regulatory Staff but that the Staff should attempt to get reasonable assurance that the critical areas of ATWS can be answered for Newbold Island prior to completing its construction permit review.

At the 130th meeting, February 4-6, 1971, during the first full Committee meeting on Newbold Island, General Electric proposed implementation of a recirculating pump trip as a backup to reactor scram. However, at the 131st meeting, March 4-6, 1971, Commonwealth Edison and General Electric maintained that no modification (such as pump trip) was needed for Quad Cities 1 and 2, which were receiving an operating license review.

In May, 1971, the ACRS received copies of a General Electric report, NEDO-10349 entitled "Analysis of Anticipated Transients without Scram," which showed the calculated behavior of a BWR with pump trip in the event of an ATWS.

On July 30, 1971, the Regulatory Staff forwarded a report to the ACRS in which they concluded that, specifically with Newbold Island in mind, the recirculation pump trip provides a substantial increase in the probability that the facility could withstand an ATWS event. The Staff noted that further analyses were required concerning the diversity of components in the pump trip, and criteria were required with regard to provision of a poison injection capability such that the suppression pool temperature is maintained below about 170°F. At the 136th ACRS meeting, August 5-7, 1971, both the Newbold Island and Limerick Stations are listed as committed to the use of the pump trip.

Beginning in early 1971, the ATWS Subcommittee, chaired by H. Monson, met frequently. First a series of meetings were scheduled with each of the four LWR vendors. All of the vendors argued that their scram systems were much more reliable than Epler or the Regulatory Staff suggested. Unreliabilities less than 10^{-6} or 10^{-7} per demand were obtained by the vendors on performing reliability analysis of their respective systems. The matter of how to include systematic or common mode failures in such analyses was admittedly troublesome.

It turned out that each of the PWR vendors had to develop new computer code models and systems in order to analyze ATWS events with the high degree of sophistication required. "Bounding" calculations led to unacceptably high pressures for some transients. And the different vendors exhibited varying degrees of cooperation and speed in analyzing the set of anticipated transients agreed upon as appropriate by the Regulatory Staff and the ACRS.

During the Executive Session of the Subcommittee meeting held with Combustion Engineering on March 30, 1971, Chairman Monson reviewed the results of the previous meeting with Westinghouse. He said it appeared that Westinghouse would be able to cope with ATWS through the possible use of one of the following:

1. Enough safety valves to prevent excessively high pressure.

2. The removal of power from some or all of the primary system pumps.

3. Use of a fast-acting injection system.

At the Subcommittee meeting held May 5, 1971 with Babcock and Wilcox (B&W), it was noted that both B&W and Combustion Engineering were still developing the needed analytical tools; also, both seemed to have reactor designs which might get to higher pressures than Westinghouse in the more severe ATWS events because of differences in system design.

The ATWS Subcommittee met again on September 8, 1971. Some excerpts from the meeting minutes follow:

> The ATWS Subcommittee met on September 8, 1971 to continue discussions regarding resolution of the ATWS problem. There appeared to be agreement between the Regulatory Staff and the Subcommittee that an improvement in reliability of approximately 10^2 to 10^4 is needed either in scram reliability or the ability to cope with an ATWS situation or that a combination of these two is needed to improve the ATWS situation by a factor of 10^2 to 10^4.

Dr. Monson said that the vendors appear to have been dragging their feet regarding the ATWS problem. The Regulatory Staff had also been slow in taking action. They have not been able to assign sufficient manpower to the problem. Dr. Monson said that he had wanted to have a Subcommittee meeting two months previous but that the Regulatory Staff was not ready at that time. He also stated that the Regulatory Staff was reportedly not ready for the present Subcommittee meeting but that he desired to proceed with the Subcommittee meeting anyway. Westinghouse and GE are the only two reactor vendors that have performed a significant amount of work regarding analysis of the ATWS problem.

B&W and CE reactors have relatively less relief capacity and less room for level swell in the pressurizer than Westinghouse reactors. Dr. Monson stated that the Committee has not accepted GE's proposed solution as a final resolution of the ATWS problem for GE reactors, but the Committee noted with satisfaction that GE is incorporating a pump trip for the Newbold Island reactor. It was stated that the proposed GE and Westinghouse changes involving tripping of pumps are inexpensive.

Dr. Monson said that the ACRS has been talking in terms of an acceptable frequency of a very severe accident of approximately 10^{-6} to 10^{-7}/reactor-year. He believed the Regulatory Staff was earlier stating a value of 10^{-7} to 10^{-8}/year but is now saying a value of 10^{-7}/year. There would be an average of less than 200 large power reactors in operation in the U.S. during the next 20 years, with a minimum total of approximately 3300 reactor-years of operation occurring during this 20-year period. With a probability of a severe accident of 10^{-6}/reactor-year, there would be a probability of .003 that a severe accident would occur within the next 20 years. He thought that this was an acceptable risk in the light of other hazards, such as a 10^{-5} probability of death each time a person takes a commercial airplane flight. The probability of an anticipated transient is being assumed to be once per reactor-year; however, the actual value might be higher. A value between 10^{-3} and 10^{-4}/reactor-year is the range that has been considered as the probability of failure to scram during an ATWS event. Dr. Monson said that, considering the number of types of scram systems and different operating and maintenance personnel at the various reactor sites, he favors using the value of 4×10^{-4} which has been proposed by the Regulatory Staff. In any event, based on an assumed acceptable severe accident probability of 10^{-6} or 10^{-7}/year and a failure to scram frequency of 10^{-3} or 10^{-4}/demand, improvement by a factor of 10^2 to 10^4 appears needed either in scram reliability or the ability to cope with an ATWS situation, or a combination of the two.

The normal operating pressure of a Westinghouse reactor is 2250 psi and the design pressure 2500 psi. An ATWS event reportedly may result in a pressure of 4391 psi in a Westinghouse reactor. Dr. Monson said that he was not certain that the reactor pressure vessel would fail at 4391 psi, but that he did not think that the probability of failure is as low as 10^{-3} for that pressure. Dr. Monson stated that there is obviously some peak pressure that would be acceptable for an ATWS event for a given reactor type and indicated the desirability of determining this value.

Mr. Epler said he believes there is a need to improve the scram system as well as a need to provide greater ability to cope with the consequences of an ATWS event.

Dr. Monson inquired as to what degree of improvement might be obtained in the reliability of the scram system. Mr. Epler replied that his associates at Oak Ridge believe that reliability could be increased by a factor of approximately 10. He thinks that changes could be made to improve the reliability by as much as a factor of 10^3. Westinghouse calculates a random failure rate of 2×10^{-7}/reactor-year. The common mode rate is an order of magnitude higher. Westinghouse feels confident that there is a probability of approximately 10^{-6} for failure of the scram breakers to open on scram demand. Mr. Epler thought that the probability of failure to scram on demand is approximately 10^{-3} for present day water cooled power reactors and that this might be improved to 10^{-4}/demand as experience is gained. Dr. Monson pointed out that 2/3rds of the control rods could fail to scram and the situation may still be acceptable. Mr. Epler thought that it is more likely that all of the rods will fail to scram than it is that one-half will fail to be inserted. Dr. Monson said that there may be common mode failures that are not common to all control rods and that the fact that only a portion of the rods need to be inserted to avoid a catastrophic situation is a significant advantage.

Regulatory Staff

Mr. Moore reported he understood from conversations with Westinghouse representatives that they have now changed their position from one of attempting to drag out the ATWS review to one of wanting to settle the matter with the Regulatory authorities. This is a result of a paragraph in the ACRS letter regarding the Newbold Island construction permit. Westinghouse construes this paragraph as possibly indicating that the GE proposal of tripping pumps has been accepted as a resolution of the ATWS question for GE reactors. Dr. Monson quoted the paragraph from the Newbold Island letter, which does not indicate that the GE proposal has been accepted as a final resolution to the ATWS question regarding GE reactors. Dr. Lipinski stated that, if the same parameters are used to scram a reactor and trip recirculation pumps, he is not sure how much reliability is improved.

Mr. Moore stated that Westinghouse has calculated that random failures have a 2×10^{-7} probability of preventing the trip breakers from opening on scram demand. GE obtained a 2×10^{-15}

value for failure to initiate scram based on a similar study. Mr. Moore said he believes that common mode failures will predominate over random failures. Mr. Moore stated that Westinghouse had omitted manufacturer deficiencies in their probability considerations. There have been two examples where such deficiencies have caused instrumentation failure. These were at the San Onofre and the Monticello reactors.

When the ATWS Subcommittee met on March 1, 1972, it had available a draft report of ATWS for Westinghouse PWRs which had been prepared by Voss Moore of the Regulatory Staff and had not been reviewed by Staff management. The report summarized the previous history of technical submissions by Westinghouse, noting that they had not provided recommendations for design changes although this had been requested on more than one occasion by the Staff.

After considerable discussion of various technical aspects, the draft report concluded that

(1) Westinghouse be informed that for future designs the consequence of ATWS should be shown to be acceptable or design changes should be made.

(2) That "emergency" conditions as specified in the ASME code, rather than "faulted" conditions as proposed by some vendors be acceptable limit for ATWS.

(3) and that other appropriate limits be put on fuel damage and containment conditions.

Excerpts from the minutes of this subcommittee meeting follow:

Executive Session

Dr. Monson stated that the matter of ATWS has been discussed for the last two or three years without much specific action being taken. It was not apparent, however, that action needed to be taken rapidly. BWRs seemed to be significantly worse regarding

ATWS than PWRs. Dr. Monson pointed out that the ACRS had gotten the applicants to perform analyses and suggest design changes relative to ATWS in the case of Newbold Island and of Limerick. In its letters regarding these reactors, the Committee recognizes the fact that design changes were being made but indicated that the ACRS did not think that these will necessarily be sufficient.

Mr. Mangelsdorf inquired regarding the effect of tripping the pumps for BWRs. Dr. Monson stated that shutting off of pumps slows the coolant flow and increases the voids in the core, thereby tending to shut the reactor down. Dr. Lipinski said that there may be a stability question with natural circulation and that there is a possibility that chugging might occur. Dr. Monson thought that the Committee would probably not want to take further action regarding changes to cope with ATWS situations until they are convinced that the existing situation is not adequate and that feasible solutions are available. Mr. Mangelsdorf inquired whether Dr. Monson believed it is possible that the ATWS problem is being overemphasized. Dr. Monson replied he believes it is possible both to overemphasize and underemphasize the problem. He indicated that the unreliability of scram systems is a debatable matter, but it appears that it is approximately 10^{-3} to 10^{-4} per demand. Mr. Epler pointed out that situations have existed where reactors were incapable of scramming at HTRE-3, at a Savannah River reactor, at a Hanford reactor, and at KAHL. Mr. Mangelsdorf indicated that he did not believe it credible that the unreliability is as high as 10^{-3} per demand for commercial water reactors. He wondered if the unreliability might not be smaller than 10^{-4} per demand. Dr. Monson believed that no system as complicated as reactor scram systems and which require maintenance and testing will have an unreliability much less than 10^{-4} per demand. It was pointed out that difficulty had been encountered with the Monticello scram system and the wrong thing was fixed three or four times. Mr. Epler said that, at the Hanford N-Reactor, three common-mode failures occurred. Filters failed, diodes failed, and all 88 rods failed to scram when one failed to scram. The Regulatory Staff is currently saying it is unacceptable to have a probability of occurrence greater than 10^{-7} per reactor-year for accidents worse than Part 100. Dr. Monson indicated he personally felt that this probability is a little on the low side and that he might be willing to accept a probability as high as 10^{-6} per reactor-year. Dr. Monson indicated that the Regulatory Staff believes that improvement is needed regarding the ATWS situation.

Regulatory Staff

Mr. Moore of DRS stated that the draft ATWS report which had been provided for use in conjunction with the Subcommittee meeting represented his own views. He said that Westinghouse had been asked to recommend a fix regarding ATWS, and they choose not to make such a recommendation. They were asked to provide details regarding the sequence of events during ATWS occurrences, and they have provided somewhat more information than had previously been submitted regarding this. The previous assumptions used by Westinghouse involved unfavorable conditions which may exist for only a small portion of time. Westinghouse was asked to provide additional information relative to this but choose not to do so. At Westinghouse's request, they were asked to discuss possible scram system improvements and then, for some reason, they choose not to answer the question.

Mr. Mangelsdorf inquired as to what Westinghouse claims is the unreliability of their scram system. Mr. Moore replied that they have indicated a value of 2×10^{-7} per demand but that they do not include common mode failures in their considerations. Dr. Monson pointed out that there have now been two clear cases where the scram systems of power reactors were inoperable, the scram systems for the KAHL and the Hanford-N reactors. At KAHL, the scram relays were replaced and were later found to have all failed simultaneously.

Mr. Moore said he believes the pressure vessel and not the primary system piping is more likely to be the limiting factor regarding resistance to pressure. The stresses for the DBE and the DBA are limited to the ASME emergency conditions and the stresses from a combination of the two are limited to the ASME faulted conditions. Mr. Moore indicated he believed that ATWS events should not result in situations worse than the ASME emergency conditions, unless the probability of an ATWS event resulting in catastrophic results has a probability of less than that of the simultaneous occurrence of the DBE and the DBA. The question was raised as to whether a single small break would be considered a faulted condition.

Mr. Moore presented Figure 1 entitled "Paths to Failure as a Result of ATWS." In this figure, the probability of failure to scram is assumed to be 10^{-4}/anticipated transient. In order to achieve a 10^{-7}/year probability that there will not be a catastrophic consequence from an ATWS occurrence, there needs to be a probability of approximately 10^{-3}/ATWS event that the primary system will not fail. Mr. Moore indicated that pressure vessel experts are not willing to state a probability for pressure vessel failure at faulted conditions. Mr. Moore said that he is inclined to believe that the pressure should not be allowed to exceed the ASME emergency value for ATWS events.

Another ATWS Subcommittee meeting was held with Combustion Engineering on March 16, 1972. Excerpts from the meeting follow:

The ATWS Subcommittee met to continue discussions, particularly regarding C.E. reactors. C.E. has provided less safety valve capacity for their reactors than Westinghouse. C.E. calculated a maximum pressure of 6000 psi for an ATWS event (loss of feedwater flow). They believe that this is an acceptable situation since 6000 psi would not result in more than 80% of the ultimate tensile strength being reached. The Regulatory Staff thought that a pressure as high as 6000 psi might damage valves which would later need to operate (e.g., ECCS values) and that the fuel cladding would collapse and possibly result in greater than Part 100 doses through leakage to the secondary system. Dr. Monson suggested that C.E. pursue investigation of possible fixes relative to the ATWS situation. He thought that their effort should be directed toward obtaining a substantial reduction in the peak pressure and peak fuel temperature for ATWS events.

Mr. Maccary indicated that the allowable stress is approximately 1/3 the ultimate strength. Upset conditions represent a stress which is approximately 110% of the design value. Emergency conditions correspond to the yield stress, and faulted conditions (as viewed by the Regulatory Staff) are those that result in stresses equal to the yield stress plus 1/3 the difference between the ultimate stress and the yield stress. The order of probability of failure due to high pressure was listed by Mr. Maccary as being (1) piping components, (2) valves, (3) pumps, and (4) pressure vessels. He thought that the probability of failure of the primary system is less than 10^{-3} for emergency conditions.

At the 144th meeting, April 6-8, 1972 the full Committee was briefed on the then current status of ATWS. The minutes follow:

Subcommittee Report on ATWS

Dr. Monson reviewed for the Committee in considerable detail the general status of ATWS knowledge in relation to water reactors, including the effects of various "fixes" on the magnitude of pressure and temperature transients accompanying the event in each type of plant.

In addition, he presented information to aid the Committee in forming opinions in regard to "acceptable" risk of a catastrophic accident.

In brief, all of the analyzed transients extend into or approach the regions of "emergency" or "faulted" condition allowances for the materials of construction. There is a spread of approximately a factor of 10^3 between the probability of ATWS per reactor year (10^{-4}) and the Staff proposed "acceptable" probability of a catastrophic accident per reactor year (10^{-7}). (The Subcommittee considers a probability of 10^{-6} to 10^{-7} for a catastrophic accident to be an acceptable figure.)

Apparent solutions would be:

- increase the reliability of scram systems by a factor of 10^{-3}, or

- ensure that the consequences of ATWS will be not greater than a Part 100 event.

Dr. Monson stated that the Committee needs to establish a position on the following considerations and that as many members as possible should plan to attend the April ATWS Subcommittee meeting in Denver, at which time these items will be taken up by the Subcommittee.

(1) Is ATWS a problem?

(2) Should the Committee write a letter?

(3) Does the Committee agree with the basic Staff position (see below)?

(4) What should be the criteria to make a non-protection system "fix" acceptable?

(a) $<$ Part 100 releases?

(b) pressure peak $<$ emergency σ?

(c) no melting of fuel?

(d) peak clad temperature $<$ design?

(5) What should be the requirements for an acceptable protection system fix?

(a) increase reliability by factor of (?)?

(b) separate, diverse system?

(c) apply requirements to "warned" plants (ACRS reports)?

(d) supply backfitting provisions?

REGULATORY STAFF POSITION ON ATWS
(Essentially unchanged since Sept. 16, 1970)

Applicants should be required either (1) to demonstrate that with their present designs the consequences of anticipated transients without scram are acceptable, or (2) to make design changes which render the consequences of anticipated transients without scram acceptable, or (3) to make design changes to improve significantly the reliability of the scram system.

The BWR and PWR manufacturers should be informed of this decision.

The analysis of ATWS in accordance with the guide (Appendix A or B) originally prepared for use by the reactor designers should be pursued with applicants. This should be done with

> applicants on current and future construction permit cases and with holders of construction permits for which the AEC safety evaluation and the ACRS letter identified this problem. Back-fitting of other cases should be considered on a case-by-case basis."

Still another ATWS Subcommittee meeting was held April 21, 1972 (the twelfth on the subject, to date). And on April 28, 1972, L. Manning Muntzing, the Director of Regulation, transmitted to the ACRS the Regulatory Staff Position on ATWS. This was reviewed by the full Committee at the 145th ACRS meeting, May 4-6, 1972, and on May 10, 1972, an ACRS letter giving general approval was dispatched to Mr. Muntzing.

Both Staff position and ACRS letter are reproduced below:

4-46

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

APR 28 1972

RECEIVED
1972 MAY 1 AM 9 00
U.S. ATOMIC [illegible] COMM.
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

Dr. Chester P. Siess
Chairman, Advisory Committee on
Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Siess:

I am enclosing eighteen copies of the requirements for protection against anticipated transients without scram (ATWS) which we plan to impose on applicants for construction permits.

In February 1969, Mr. E. P. Epler, an ACRS consultant, suggested that protection systems designed for highly unlikely events might not have the required reliability to protect against anticipated transients. Our review found that several anticipated transients require protection system action (scram) to prevent unacceptable consequences. We also investigated the reliability required of scram systems for these transients and have concluded that protection against ATWS should be provided in accordance with the enclosed requirements. Our conclusions were based on an investigation of scram system experience in power reactors which considered common mode failures as well as random failures. We reported the results of our studies to the committee in September 1970 with the recommendation for new requirements. It should be noted that the reactor manufacturers believe that the reliability of scram systems is so high that protection against ATWS should not be required. The staff disagrees on the basis of its studies referred to above, and believes in addition that information on scram reliability obtained since September 1970 reinforces its earlier conclusion.

Since our September 1970 recommendation, we have continued to study anticipated transients with the water reactor designers. In December 1970, the staff requested the four water reactor manufacturers to furnish detailed information regarding the consequences of ATWS and possible remedial measures. The staff has reviewed the information presented in response to this inquiry, and finds that its earlier conclusions are unchanged. An ACRS subcommittee has held meetings with the manufacturers and has been kept fully informed by the staff during its review.

Reproduced and distributed following [illegible] 145th ACRS Mtg.

In August 1971, the ACRS concurred in the staff recommendation that a design change (recirculation pump trip) was required to make the consequences of anticipated transients without scram acceptable for the Newbold Island reactors. The same design change has been required for subsequent boiling water reactors.

Based on the foregoing, we have concluded that it is advisable at this time to impose requirements on construction permit applicants for all water reactors in order to make further progress on the problem of anticipated transients without scram. The enclosed regulatory requirements are consistent with our recommendation of September 1970. More specific information, however, is provided to define acceptable consequences and to assist in implementing design changes. Studies are continuing to develop lists of required transients and assumptions, and acceptable evaluation models for the various designs.

ACRS comments on the proposed requirements that we receive soon after the May meeting would be most timely in our considerations.

Sincerely,

L. Manning Muntzing
Director of Regulation

Enclosure:
Requirements (18)

REGULATORY STAFF POSITION ON

ANTICIPATED TRANSIENTS WITHOUT SCRAM

April 28, 1972

I. Recommended Position

Applicants should be required to: (1) demonstrate that with their present designs the consequences of anticipated transients without scram (ATWS) are acceptable, or (2) make design changes which render the consequences of anticipated transients without scram acceptable, or (3) make design changes to improve significantly the reliability of the scram system.

II. Implementation

A. Definition of Acceptable Consequences

It is necessary to establish acceptable consequences of ATWS in order to implement either option 1 or option 2 of the recommended position. Acceptable conditions are defined as follows:

1. Radiological consequences

   The radiological consequences shall be within the guideline values set forth in 10 CFR Part 100.

2. Primary System Pressure

   The maximum acceptable transient primary system pressure shall be based on the primary system pressure boundary limit or the fuel element limit whichever is more restrictive:

   a. Primary Pressure Boundary Limits

      The transient pressure shall be limited to less than that resulting in a maximum stress anywhere in the reactor coolant pressure boundary of the "emergency conditions" as defined in the ASME Section III Nuclear Power Plant Components Code.

   b. Fuel Pressure Limits

      The transient pressure shall not exceed a value for which test and/or analysis demonstrate that there is no substantial safety problem with the fuel. The safety considerations include radiological consequences as well as hydraulic effects.

3. Fuel Thermal and Hydraulic Effects

   a. The average enthalpy of the peak pellet shall not exceed 280 calories per gram.

b. **A calculated critical heat flux event will not be acceptable unless the peak cladding temperature can be shown not to result in significant cladding degradation.**

b. Containment Conditions

Calculated containment pressures shall not exceed the design pressure of the containment structure. Equipment which is located within the containment and which is relied upon to mitigate the consequences of ATWS shall be qualified by testing in the combined pressure, temperature and humidity environment conservatively predicted to occur during the course of the event.

B. Analyses of Possible Detrimental Effects of Required Modifications

Any modifications made to comply with option 2 of the recommended position shall be shown not to result in violations of safety criteria for steady state, transient, or accident conditions and shall not substantially affect the operation of safety related systems.

C. Diversity Requirement for Implementing Option 2 of the Recommended Position

Design changes to make the consequences of ATWS acceptable should not rely on equipment or system designs which have a failure mode common with the scram system. The equipment involved in the design change shall, to the extent practical, operate on a different principle from equipment in the scram system. As an absolute minimum, the equipment relied on to render acceptable the consequences of the ATWS event shall not include equipment identical to equipment in the associated scram system.

D. Diversity Requirement for Implementing Option 3 of the Recommended Position

Improvements must reduce considerably the potential for common mode failure of the scram system. Failures of identical equipment from a common mode should not disable sensing circuits, logic, actuator circuits or control rods to the extent that scram is ineffective. The addition of a separate protection system utilizing principles diverse from the primary protection system is indicated in order to meet this requirement.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 10, 1972

Mr. L. Manning Muntzing
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Muntzing:

Your letter of April 28, 1972, describes requirements for protection against anticipated transients without scram (ATWS) which the Regulatory Staff plans to impose on applicants for construction permits, and notes that ACRS comments on the proposed requirements received soon after the ACRS May meeting would be timely. The Committee wishes to make the following comments.

I. The ACRS recognizes ATWS as a low probability event. Nevertheless, it believes that, in consideration of the large number of water-cooled power reactors expected eventually to be in operation, and in view of the expected occurrence rate of anticipated transients (collectively, on the order of one per reactor per year), experience with scram systems of current design is insufficient to give assurance of an adequately low probability of occurrence of an ATWS event of possibly serious consequence. Accordingly, the Committee agrees with the intent of the ATWS position recommended, viz:

> "Applicants should be required to: (1) demonstrate that with their present designs the consequences of anticipated transients without scram (ATWS) are acceptable, or (2) make design changes which render the consequences of anticipated transients without scram acceptable, or (3) make design changes to improve significantly the reliability of the scram system."

II. The Committee has the following comments on the criteria proposed to be used in implementation of the basic position.

A. In respect to the proposed definition of "acceptable consequences" for implementation of either option 1 or option 2:

> 1. Concerning radiological consequences, we agree with the proposed condition: "The radiological consequences shall be within the guideline values set forth in 10 CFR Part 100".

2. Concerning primary system pressure:

a. We agree with the intent of the proposed condition: "The transient pressure shall be limited to less than that resulting in a maximum stress anywhere in the reactor coolant pressure boundary of the 'emergency conditions' as defined in the ASME Section III Nuclear Power Plant Components Code". However, we believe the wording should be changed so as to read along the following lines: "The transient pressure shall not be greater than that which results in reactor coolant pressure boundary stress conditions corresponding to those of 'emergency conditions' as defined in ASME Section III Nuclear Power Plant Components, 1971". We believe it should be noted that the intent of this provision is to obviate the need to consider a loss-of-coolant accident (LOCA) in conjunction with an ATWS event.

b. We agree with the intent of the proposed condition as expressed in the first sentence: "The transient pressure shall not exceed a value for which test and/or analysis demonstrate that there is no substantial safety problem with the fuel" However, we recommend that the second sentence be deleted.

3. Concerning fuel thermal and hydraulic effects:

a. The Committee believes that the proposed limit on enthalpy of the peak pellet of 280 calories per gram should not be adopted at this time. A single limit for all cases may not even be desirable. It is recommended that the criterion be changed to indicate that, in an ATWS involving a power excursion, the effects of rapid increase in fuel enthalpy shall not result in significant cladding degradation or in significant melting of fuel even in the hottest fuel zones.

b. We agree with the intent of the proposed condition: "A calculated initial heat flux event will not be acceptable unless the peak cladding temperature can be shown not to result in significant cladding degradation".

4. Concerning containment conditions, we agree with the intent of the proposed condition: "Calculated containment pressure shall not exceed the design pressure of the containment structure. Equipment which is located within the containment and which is relied upon to mitigate the consequences ~~of~~ ATWS shall be qualified by testing in the combined pressure, temperature and humidity environment conservatively predicted to occur during the course of the event".

B. We agree with the intent of the proposed requirement B:

"Any modifications made to comply with option 2 of the recommended position shall be shown not to result in violations of safety criteria for steady state, transient, or accident conditions and shall not substantially affect the operation of safety related systems."

C. We agree with the intent of the proposed requirement C:

"Design changes to make the consequences of ATWS acceptable should not rely on equipment or system designs which have a failure mode common with the scram system. The equipment involved in the design change shall, to the extent practical, operate on a different principle from equipment in the scram system. As an absolute minimum, the equipment relied on to render acceptable the consequences of the ATWS event shall not include equipment identical to equipment in the associated scram system."

D. We agree with the intent of the proposed requirement D:

"Improvements must reduce considerably the potential for common mode failure of the scram system. Failures of identical equipment from a common mode should not disable sensing circuits, logic, actuator circuits or control rods to the extent that scram is ineffective. The addition of a separate protection system utilizing principles diverse from the primary protection system is indicated in order to meet this requirement."

III. In addition to the above comments on the proposed requirements, the Committee makes the following recommendations:

A. In any announcement of the basic position recommended, as well as in its implementation, care should be taken to make clear the fact that availability of options 1 and 2 is not to be construed as prejudicing in any way the importance of continuing effort to improve existing scram systems to the extent practical, irrespective of ATWS considerations.

B. As indicated in your letter, Regulatory Staff studies are continuing for the purpose of developing lists of transients to be considered, assumptions to be used, and acceptable evaluation models. We recommend that this effort be accelerated to the extent practical, in order that a maximum of guidance be available to the applicants upon commencement of implementation. It is also recommended that the list of transients to be treated be described as a minimum but not necessarily sufficient list, with the applicant responsible for identifying all relevant transients.

C. The Committee recommends that the proposed position, modified as above, also be implemented on a reasonable time scale in respect to those water-cooled power reactors under construction for which the ACRS letter and the Regulatory Staff safety evaluation associated with the construction permit identified the ATWS problem. The Committee assumes that, in due course, the Regulatory Staff will propose an appropriate course of action in connection with earlier plants.

IV. The Committee intends to continue working closely with the Regulatory Staff in the further study and development of criteria and procedures to be applied in the ATWS area.

Sincerely yours,

Original Signed by
C. P. Siess

C. P. Siess
Chairman

After more than three years, the matter appeared to have been resolved. Reactor designers would have to demonstrate an ability to tolerate ATWS, unless by changes in design they could provide convincing arguments that the scram systems were 100-1000 times more reliable. The anticipated alternative was to make the designs such that the consequences of ATWS would be tolerable.

However, the Regulatory Staff continued to hold internal discussions with regard to their position on ATWS. In a draft dated November 30, 1972, they reversed their stance with regard to the recommended fix for ATWS, namely, for reactors whose construction permit applications were to be filed a few years in the future. Improved reliability in the shutdown system (i.e., two independent systems) would be required, rather than an ability to tolerate the consequences of ATWS.

The ATWS Subcommittee met with the Regulatory Staff to discuss this revised position on January 10, 1973. The problem of how and whether to "backfit" ATWS to plants in operation or under construction was a particularly difficult one, in view of the considerable problems involved in making changes in valves, piping, etc. in an existing plant. This problem had been aggravated by the increased number of plants now in this situation, vis-a-vis 1969. The Subcommittee questioned the proposed use of "faulted conditions" as an acceptable pressure limit for backfitting.

On January 22, 1973, Mr. Muntzing formally forwarded a new recommended licensing position on ATWS to the ACRS. The Muntzing letter makes it appear that Newbold Island adopted the recirculation pump trip as a result of Staff initiative, rather than that of the ACRS. The letter and its attachment are reproduced (in large part) on the following pages.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

JAN 2 2 1973

Mr. Harold G. Mangelsdorf, Chairman
Advisory Committee on Reactor Safeguards
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Mangelsdorf:

I am enclosing 18 copies of a recommended licensing position on the need for protection against anticipated transients without scram (ATWS) for water-cooled power reactors. This position is a result of the Regulatory staff's continuing review and of discussions with the ACRS following our exchange of letters on ATWS in April and May 1972.

As noted in my earlier letter, it has been suggested that protection systems designed for highly unlikely events might not have the reliability to protect against anticipated transients that is required in view of the large number of water-cooled power reactors anticipated to be in operation in the future. Our review found that several anticipated transients require protection system action (scram) to prevent unacceptable consequences. We also investigated the reliability required of scram systems for these transients and have concluded that protection against ATWS should be provided. Our conclusions were based on an investigation of scram system experience in power reactors which considered common mode failures as well as random failures. We reported the results of our studies to the Committee in September 1970. It should be noted that the reactor manufacturers believe that the reliability of scram systems is so high that protection against ATWS should not be required. The Regulatory staff disagrees on the basis of its studies referred to above, and believes that additional information on scram reliability obtained since September 1970 supports its earlier conclusion. The staff has reviewed detailed information requested of the four water-cooled power reactor manufacturers in December 1970 on the consequences of ATWS and possible remedial measures, and found that its earlier conclusions were unchanged. An ACRS subcommittee held meetings with the manufacturers and was kept fully informed by the Regulatory staff during its review. In August 1971, the ACRS concurred in the staff recommendation that a design change (recirculation pump trip) was required to make the consequences of ATWS acceptable for the Newbold Island reactors. The same design change has been required for subsequent boiling water reactors.

We continue to believe, based on the foregoing, that it is advisable to begin to impose requirements on construction permit applicants for all water-cooled power reactors in order to maintain an appropriate level of safety with regard to possible ATWS events and the increasing number of power reactors. The enclosed recommended licensing position is consistent with our previous recommendations. Studies are continuing to develop lists of required transients and assumptions, and acceptable evaluation models for the various designs.

I would appreciate having any comment the ACRS might like to make on the enclosed licensing position.

Sincerely,

L. Manning Muntzing
Director of Regulation

Enclosure:
Licensing Position on ATWS (18)

RECOMMENDED LICENSING POSITION ON ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) FOR WATER-COOLED POWER REACTORS

I. Recommended Position

A. Applicants for construction permits, where application is made after __________*, should be required to incorporate design changes that improve significantly the reliability of the reactor shutdown systems, as compared with current designs.

B. Applicants and licensees with plants for which the need for provisions for ATWS is noted in the AEC construction permit-stage Safety Evaluation Report or Advisory Committee on Reactor Safeguards Report, and applicants for construction permits, where application is made before __________* and where the construction permit review has not been completed, should be required to incorporate any design changes necessary to assure that the consequences of anticipated transients would be acceptable in the event of a postulated failure to scram.

C. The need for backf·tting of plant changes to mitigate the consequences of ATWS in plants for which neither the AEC construction permit-stage Safety Evaluation Report nor the Advisory Committee on Reactor Safeguards Report identify ATWS as a continuing area of review should be considered on an individual case basis.

II. Implementation

A. Requirements for Improved Reactor Shutdown Systems. Applicable to plants for which construction permit applications are made after __________*.

1. Protection Against Common Mode Failures.

The required improvements in reactor shutdown systems must reduce significantly the potential for common mode failure of the shutdown system. Postulated common mode failures of identical equipment should not disable sensing circuits, logic circuits, actuator circuits, control rod mechanisms, or other shutdown system components to the extent that shutdown is ineffective. In order to meet this requirement, the provision of two separate shutdown systems utilizing diverse principles and equipment seems under present circumstances to be indicated.

*Date to be 3 years after promulgation of position.

2. Program for Implementation.

A comprehensive program leading to the design, manufacture, and installation of reactor shutdown systems improved significantly over present systems in reliability and freedom from common mode failures is required, together with such research and development as is needed to support the program. The research and development needed for this purpose must be accomplished in a time consistent with the requirements of Paragraph I.A. The Regulatory staff will require each reactor manufacturer to submit periodic reports of the progress of its program to develop an improved system and will review such progress reports with the manufacturers to assure that a program consistent with the new requirements is being pursued expeditiously. It is expected that the improved systems will be included in the design of plants for which applications for construction permits are made after __________*.

B. Requirements to Make ATWS Consequences Acceptable. Applicable to: (a) plants for which the need for provision for ATWS is noted in the AEC Safety Evaluation Report or the Advisory Committee on Reactor Safeguards Report at the construction permit stage; and (b) plants for which construction permit applications are or have been made prior to __________*, and for which the AEC Safety Evaluation Report is not yet issued.

1. Calculation of Consequences.

The calculated radiological consequences should be within the guideline values set forth in 10 CFR Part 100. In addition, the limits listed below on calculated system pressure, fuel performance, and containment conditions should be required.

a. Reactor Coolant System Pressure.

The maximum acceptable calculated transient reactor coolant system pressure should be based on the system boundary pressure limit or the fuel pressure limit, whichever is more restrictive:

*Date to be 3 years after promulgation of position.

(i) Reactor Coolant System Boundary Pressure Limit.

The calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the "emergency conditions" as defined in the ASME Nuclear Power Plant Components Code, Section III.

(ii) Fuel Pressure Limit.

The calculated reactor coolant system transient pressure should not exceed a value for which tests and analyses demonstrate that there is no significant safety problem with the fuel.

b. Fuel Thermal and Hydraulic Performance.

(i) The calculated average enthalpy of the hottest fuel pellet should not result in significant cladding degradation or significant fuel melting.

(ii) A calculated critical heat flux event will not be acceptable unless the calculated peak cladding temperature can be shown not to result in significant cladding degradation.

c. Containment Conditions.

Calculated maximum containment pressure should not exceed the design pressure of the containment structure. Equipment located within the containment that is relied upon to mitigate the consequences of ATWS should be qualified by testing in the combined pressure, temperature, and humidity environment conservatively predicted to occur during the course of the event.

2. Evaluation Techniques.

Analysis models and techniques, including computer codes, used for conservative evaluations of the consequences of postulated ATWS events, together with associated assumptions and parameters, should be described and justified in topical reports.

3. Review of Reactor Shutdown System Design.

A review of the reactor shutdown system design should be made with the aim of identifying and correcting areas that might be particularly vulnerable to common mode failures.

4. Diversity Requirements.

Design changes to make the calculated consequences of a postulated ATWS acceptable should not rely on equipment which has a failure mode common with the anticipated transient or the shutdown system. To the extent practical, the equipment involved in the design should operate on a different principle from equipment in the shutdown system. As a minimum, the equipment relied on to make the consequences of a postulated ATWS event acceptable should not include equipment identical to equipment in the associated shutdown system. Such designs should be shown not to result in violations of safety criteria for steady state, transient, or accident conditions and should not adversely affect the operation of any safety-related systems.

5. Program for Implementation.

For plants already under construction, or for which construction permit applications are made prior to ___________**, applicants should submit by ___________** evaluations of the consequences of ATWS, plans for any proposed plant changes required to make the consequences acceptable, and a schedule for implementation of the proposed changes. Applications for construction permits made after ___________** should include analyses to show that the consequences of ATWS are acceptable with the proposed plant design. The regulatory staff will require each applicant or his reactor manufacturer to submit periodic reports of the progress of his program to make ATWS consequences acceptable and will review such progress reports with applicants and manufacturers to assure that a program consistent with the above schedule is being pursued. The program for topical reports of evaluation models (Paragraph II.B.2) and the experimental basis for the models will be reviewed for consistency with the above schedule.

**Date to be 1 year after promulgation of position.

C. Requirements for Backfitting of Plant Changes. Applicable to plants for which neither the AEC construction permit-stage Safety Evaluation Report nor the Advisory Committee on Reactor Safeguards Report identifies ATWS as a continuing area of review.

1. Analysis of ATWS Consequences.

An analysis should be made of the consequences of anticipated plant transients in the event of a postulated failure to scram. The analysis should show whether

a. calculated reactor coolant system transient pressure exceeds a value such that the maximum primary stress in the system boundary is equal to that of the "emergency conditions" as defined in the ASME Nuclear Power Plant Components Code, Section III, or

b. effects of the ATWS event result in significant fuel cladding degradation or significant fuel melting, or

c. calculated containment pressure exceeds the design pressure of the containment structure.

2. Review of Reactor Shutdown System Design.

A review of the reactor shutdown system design should be made with the aim of identifying areas that might be particularly vulnerable to common mode failures.

3. Program for Implementation.

The analysis of ATWS consequences and the reactor shutdown system design review should be submitted by ____________** for review by the regulatory staff. The staff will evaluate the need for plant changes with the objective of achieving an appropriate resolution of the ATWS issue on an individual case basis.

**Date to be 1 year after promulgation of position.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 11, 1973

HIGHLIGHTS OF
ATWS-RELIABILITY ANALYSIS SUBCOMMITTEE MEETING
APRIL 7, 1973

The combined ATWS and Reliability Analysis Subcommittees met in Washington, D. C., on April 7, 1973, to discuss redundancy-diversity and ATWS. This was an internal meeting; the attendees were limited to ACRS members, ACRS Staff, and ACRS Consultants.

1) Epler feels that the unreliability of current scram systems is about 10^{-3} to 10^{-4}/demand. Lipinski and Vesely agree.

2) Monson suggests the possible use of 10^{-7}/reactor year as the acceptable probability of ATWS.

3) The vendors report that they meet this 10^{-7} requirement.

4) The ATWS is assumed to yield a major accident; that is, result in offsite whole body or thyroid doses of 25R and 300 Rem respectively.

5) Vesely feels that common mode failures in identical systems may increase the unreliability by a factor of 10^3 to 10^4.

6) Assuming 1., 2., and 5. above, redundant fast acting scram systems diverse in design may be required. This could mean that extra rods, as compared to the current designs, would be necessary.

7) Several consultants noted that a good data base is required in order to develop better reliability numbers. Reethof stated the recently described EEI data collection program does not meet the needs.

8) Kerr feels that the designer should be provided the reliability criteria and then it would be his option as to how to meet the criteria.

9) Kerr notes that diverse systems infer two sets of spare parts, separate maintenance procedures, and installation of one system inferior to the other. On balance, it is not clear to him that diversity is good.

10) The 10^{-7}/reactor year value discussed above appears to result in a probability of affecting a person offsite several orders of magnitude less probable than that person dying from accidents such as lightning, electrocution, etc.

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JEH

J. E. Hard

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 16, 1973

Mr. L. Manning Muntzing
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Muntzing:

Your letter of January 22, 1973, transmitted copies of a revised recommended licensing position on the need for protection against anticipated transients without scram (ATWS) for water-cooled power reactors, and invited ACRS comment. The Committee considers this position to be generally consistent with your earlier proposed position favorably commented on by Committee letter of May 10, 1972, and supports the position.

It is suggested that, in those cases of "backfit" plants for which an OL stage or post-OL stage ACRS report or AEC Safety Evaluation identifies ATWS as a continuing area of review, the philosophy used in evaluating the possible need for plant design changes should be similar to that for plants in Class I.B, recognizing the need for special consideration of the problems that may result if extensive modifications appear to be required.

The Committee also continues to believe that it is timely to begin implementation of the proposed ATWS position.

Sincerely yours,

H. G. Mangelsdorf
H. G. Mangelsdorf
Chairman

cc: J. M. Hendrie, L
J. F. O'Leary, L
P. C. Bender, SECY
A. Giambusso, L

In September, 1973 the Regulatory Staff issued WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors,"

4.4 ATWS - Part 2

With the issuance of WASH-1270 in September, 1973, the Regulatory Staff had taken a position on the matter and it was seemingly resolved except for implementation. The ACRS letter of February 13, 1974 on the status of generic items moved the ATWS matter into the resolved column on exactly this basis. In the period 1974-1975 all the reactor vendors submitted analyses on ATWS in general response to the requirements set forth in WASH-1270.

On July 3, 1974 an ACRS Subcommittee meeting was held with representatives of the Regulatory Staff to discuss the status of ATWS, and to permit a discussion of the ATWS criteria/guidelines being developed by the Staff for Class A, B and C plants. The Staff advised that no firm proposals had been submitted by the reactor vendors for Class A plants (those for which a construction permit application would be filed after October 1, 1976 and for which diverse shutdown systems of high reliability would be required.) Dr. Hanauer of the Regulatory Staff stated that several vendors had already submitted analyses which showed protection system unreliability values smaller than 10^{-7}, but that these were not considered acceptable by the Staff because they did not satisfy the intent of WASH-1270 or the Staff guidelines/criteria. The Staff agreed that the vendors did not consider the Staff guidelines/criteria to provide adequate guidance for the initiation of new designs.

A brief excerpt from the meeting minutes illustrates a small portion of the complex problems involved for "Class A" reactors.

ATWS Working Group Meeting

Class A Criteria/Guidelines - Supplemental Comments/Questions

1. With respect to the criteria/guidelines presented on Attachments ATWS "A" 3 and 4. DRL admitted the vendors do not consider this to be adequate guidance for the initiation of new designs.

2. Dr. Hanauer of DRL stated that several vendors have already submitted analyses which show protection system unreliability values smaller than 10^{-7}, but these are not considered acceptable by DRL because they do not satisfy the intent of the WASH-1270 requirements or the guidelines/criteria given on Attachments ATWS "A" 3 & 4.

3. With respect to boron injection as an alternate backup shutdown mechanism, there was much comment to the effect that it would be impractical to activate boron injection on every transient, particularly in BWRs, because of the difficulty of cleaning up the systems. Feed-and-bleed can be used in PWRs -- apparently there is no analogous simple routine procedure for BWR.

4. Dr. Monson raised the question of status of Prompt Relief Trip (PRT) in connection with Pump Trip fix proposal by GE, as indicated on Attachment ATWS "A"-5. He had observed in documentation forwarded to ACRS within recent months that an unanticipated shift in scram curves (reactivity addition vs. rod insertion), had been noted recently in tests following refuelings at some BWRs. Apparently consideration of void distribution in the BWR core and the anomalous shift in the scram curve indicate the possibility of overpressure following Pump Trip - GE had proposed PRT as a fix for this potential problem. DRL indicated that the Monticello docket will be the lead case for evaluation of the adequacy of PRT. Preliminary indications are that PRT will be an acceptable fix, but DRL has not yet seen the detailed analysis.

5. With respect to use of rods as an alternate type of backup shutdown system as indicated on Attachments ATWS "A"-6 and ATWS "A"-8, Dr. Monson expressed concern regarding the DRL position that diversity only need apply to the drive release mechanism (<u>not</u> to the rod itself). He felt that the diversity requirement should apply to the rods as well, at least to the extent that the clearances between rod and guide should be greater in one set/group of rods. (For information, it was noted that CE also shares the view that diversity should apply to the shutdown mechanism itself). In support of this view, he postulated a crud release as a possible common-mode failure mechanism for identical control rods.

   In this context Dr. Kerr cautioned that diversity itself could produce problems (e.g., in maintenance, spare parts, etc.). He considered it possible that thorough evaluation might show that some set of undetected non-common mode failures in diverse protection systems might be as bad (or worse) than undetected common-mode failures in redundant protection systems. He would support a requirement for diversity, only if an improvement in reliability can be demonstrated as a direct result.

   DRL admitted that Dr. Kerr's question/comment could not be properly addressed at this time... A strong feeling was expressed, however, that although it has not to date been analytically/definitively demonstrated, present knowledge and experience seem to indicate that diverse shutdown system design is the optimum approach.

   DRL expressed a feeling of disquiet regarding any significant design change (e.g., additional/changed penetrations in the reactor vessel to accommodate bottom/top/side rods) for ATWS, in consideration of all the associated as-yet-unknown problems which would likely result from such modifications.

Dr. Monson reiterated and emphasized during these discussions his concern that, in providing a diverse protection system for ATWS, extreme care should be exercised that the reliability of the existing reactor protection system should not be degraded.

6. DRL noted candidly that reactor vendors are not enthusiastic in their approach to solving the ATWS problem, because they feel that a problem does not exist. Further, they fear ratcheting [sic] if they do serious analysis and propose solutions. Apparently their approach is to await DRL criteria (licensing requirements), and will design to meet them when they are issued. In this context, DRL paraphrased a Westinghouse comment to the effect that one very effective protection against ATWS is a reactor design which can withstand ATWS -- in that sense the existing scram system is already a backup protection system.

There was also considerable discussion concerning the guidelines to be proposed for Class B plants (those whose design was such that the consequences of ATWS were tolerable as were reliability requirements for systems which must work in the event of an ATWS.)

The minutes show that the following summary was given by the Staff.

ATWS Working Group Meeting July 3, 1974

DRL supplemented the Class A/Class B status for the four major vendors as given in the Attachments as follows:

a. Westinghouse -- existing plants are amenable to ATWS solution for Class B because of large steam generator volumes and more relief capacity. This vendor seems to feel that may "go away" for Class A.

CE -- current designs are not amenable to solution for Class B -- This vendor appears to be concentrating current efforts on solution for Class A plants.

B&W -- this vendor appears to feel that their design is already optimized for all considerations by the rod runback feature.

GE -- this vendor has been relatively uncommunicative recently -- ATWS plus PRT plus reactivity scram curve shifts appear to be making engineering a solution difficult-to-impossible (to the extent that DRL considers it a possibility that GE may choose to fight WASH-1270 on the basis that it is not a bonafide licensing requirement, but only an "interesting technical document").

In September of 1975, the Regulatory Staff asked to meet with the ACRS concerning a possible major change in the approach to ATWS adopted in WASH-1270 for "Class A" plants. At a Subcommittee meeting held October 8, 1975, the Staff listed four alternatives, as follows:

1. Implement the original WASH-1270 position (i.e., no change in philosophy).

2. Make a re-evaluation of the positions in WASH-1270.

3. Accept partial conformance with the WASH-1270 position.

4. Apply the Class B position to Class A plants, namely that the consequences of an ATWS be tolerable.

According to the Staff, Alternative 1 had as an objective the elimination of ATWS as a design basis accident. The rationale is based on reduction of the probability of an ATWS to an acceptably low value. The Staff noted that if this were accomplished for Class A plants, Class B plants would become a controlling factor on risk, since there would be a large number of Class B plants, roughly 300.

Mr. Minners of the Staff, in discussing Alternative 2, said the objective here would be to determine if any new information changes the conclusions of WASH-1270; the Staff did not expect that a re-evaluation would reveal that the probability of an ATWS in plants of current design was significantly different than assigned in WASH-1270.

With regard to Alternative 3, Mr. Minners said that the PWR vendors had proposed additional shutdown systems which practically conform to the independence and diversity requested in WASH-1270. Such systems could reduce the probability that the system would not de-energize the control rods but would not assure scram, and the safety objective might not be satisfied.

The Staff was recomending Alternative 4, namely making the consequences of ATWS acceptable (also working on improving the reliability of the existing protection system). Mr. Minners said that this alternative was based partly on the argument that if the safety of Class B plants could be made adequate, additional requirements for Clss A plants were unwarranted, and, in fact, the provisions for Class B plants might be better able to cope with situations which were currently unrecognized. Implicitly, there appeared to be doubt among the Staff that diverse shutdown systems could or would be proposed and developed to the point where the Staff could concur that the probability of ATWS was acceptably low.

The ACRS concurred with the Staff proposal to revise the criteria for "Class A" plants, along the lines of Alternative 4, in a letter to Mr. Lee Gossick (NRC Director of Operations) dated October 17, 1975. Effectively the Staff had reverted to their original approach, that proposed prior to WASH-1270 (and the one favored by the ACRS).

In documents dated December 9, 1975 the Regulatory Staff prepared status reports on ATWS for each of the four LWR reactor vendors. In these documents the Staff took positions with regard to "acceptable fixes" for Class B plants and identified outstanding issues. The ACRS held three days of subcommittee meetings and an extensive discussion with the Staff and the reactor vendors (who disagreed with many Staff positions) at the 189th meeting, January 8-10, 1976. The minutes of this meeting give some of the flavor of the controversy which continued to exist.

> Mr. Kerr, Subcommittee Chairman, reviewed the history of the identification of the probability of unreliability of shutdown systems. If the appropriate unreliability probability is 10^{-4} to 10^{-5} per demand, then the NRC Staff concludes that there will be a shutdown failure in PWRs once every 5 to 10 years. (The NRC Staff, in trying to establish overall plant reliability of 10^{-6} per year, believes that ATWS should contribute no more than 10% to this unreliability, leading to the conclusion that the probability of an ATWS should be 10^{-7} or less. Mr. Kerr noted that there is a wide difference of opinion between the NRC Staff and the NSSS vendors regarding the following matters:
>
> - Moderator coefficient of reactivity used in calculations - vendors wish to use values covering 95% of core life; NRC Staff wants to require 99%.
> - NRC Staff has not decided on the unreliability it will accept regarding fixes for the ATWS problems; the vendors are unable to demonstrate how this unreliability will be determined.
> - NRC Staff and the vendors have not agreed on the definitions of diverse systems.
> - Vendors do not believe that diversity of shutdown systems is a universal fix while NRC Staff is inclined to assume that it is. Both sides desire ACRS backing.
>
> A. Status of NRC Staff Review
>
> W. Minners, NRC Staff, recalled that in its 1973 report on ATWS, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors" (WASH-1270), the NRC classified power reactors into three classes: Class A, Class B, and Class C. In action taken in October 1975 with ACRS agreement, the NRC Staff decided to handle Class A and Class B reactors in the same manner; Class C reactors will be handled on a case-by-case basis. The discussions at this meeting will relate to solutions for the problems of Class B plants. He noted that the current NRC Staff safety goal is to reach the probability of exceeding the WASH-1270 goal of 10^{-6} to 10^{-7} per reactor year. The anticipated transients are listed in WASH-1270.
>
> He said that agreement has been reached with PWR vendors regarding which transients will be analyzed, and noted that

the analyses are complete. Such agreement has not been reached with the BWR vendor, but an analysis for the spurious main steam line isolation valve (MSLIV) closure has been received; this transient is believed to be the most limiting.

With regard to PWRs, agreement has been reached with the PWR vendors over most of the parameters to be used in the analyses, but there is disagreement over the moderator coefficient of reactivity which will be used. The NRC Staff wants to use the moderator for 99% of the cycle, vendors 95%.

Initial conditions at the start of ATWS were discussed. He noted the loss of offsite power will be used as the initiating event only. The NRC Staff is insisting that the single failure criterion be applied; vendors want to use a single system failure only.

Agreement has not been reached over the probabilities of unreliability to be used in the analyses, especially for failure of control rod drive mechanisms.

The NRC Staff is requiring that either diverse systems be provided or that the vendors demonstrate that the NRC Staff's goals can be met without the diverse systems. Emergency stress limits must be met during the pressure transient. Fuel limits set in WASH-1270 must be met. Pressure in containment shall not exceed design pressure. In BWRs the suppression pool temperature is also limited.

NRC Staff believes that it is in a position to implement their review of individual plants.

B. Comments by Vendors

1. Babcock and Wilcox Company (B&W)

J. Penland discussed the B&W position and reviewed the results of the B&W analysis to show the adequacy of the B&W design. He presented an event probability approach for ATWS and concluded that B&W plants now meet the WASH-1270 goals. He concluded that the 95% moderator coefficient of reactivity assures conservative compliance with the safety objectives. He recommended:

- that the NRC Staff report be withdrawn for re-evaluation.
- that the safety objective of WASH-1270 be reaffirmed, and that formal reliability methods be applied.
- that ANSI standard N661 be used to formulate definitive ATWS criteria.

D. LaBelle, B&W, stated that the Babcock & Wilcox calculations were made on "worst case events" and concluded that:

- there is no undue risk to the public.
- there are beneficial PWR design features which mitigate the consequences of ATWS.
- B&W PWR designs are adequate.

2. Westinghouse Corporation (W)

K. Paulsen, W, discussed some of the transients used in the W ATWS analyses.

He concluded that:

- statistical conservatisms in the Westinghouse analysis are consistent with the requirements of WASH-1270.
- peak system pressures for ATWS transients show approximately 100 to 200 psi margins for emergency stress limits.
- cladding or fuel damage is not anticipated, therefore 10 CFR 100 dose limits will not be exceeded.
- containment peak pressure is calculated to be lower than the design pressure.
- Westinghouse calculations are conservative.

K. Paulsen commended on some of the items addressed in the NRC Staff Status Report. D. Peacock, W, stated that Westinghouse views the NRC Staff's review of ATWS to be incomplete, and that this matter should not be before the ACRS at this time. He requested a supplement to the NRC Staff evaluation. Westinghouse believes some of the NRC Staff positions are both technically unsupportable and are therefore unacceptable.

In particular, he challenged the NRC Staff's position on:

- 99% moderator coefficient of reactivity.
- NRC Staff's concept of diversity is not well defined, nor have they suggested how such diversification can be implemented.

He stated that Westinghouse is willing to work with the NRC Staff to resolve these matters, and believed that resolution might be achieved in six to seven months.

T. Novak, NRC Staff, stated his belief that the issues are clear, and that the NRC Staff position will get the ATWS problem moving to completion. The issue of the moderator coefficient is not fundamental to the solution.

3. Combustion Engineering, Inc. (CE)

P. Coppersmith, CE, used a simple block diagram to indicate how rules, regulations, and standards are factored into plant design. He said that if an ATWS solution is to be factored into plant design, there is a need to clarify the requirements before hardware can be specified. He discussed the relationship between the moderator temperature coefficient and the peak pressurizer pressure, and between the fraction percent of plant operating lifetime, and the peak pressurizer pressure.

J. Herbst, CE, stated that the ATWS analyses utilizes probabilistic methods and considers failure early in plant life.

E. Scherer, CE, said that Combustion Engineering would want to evaluate any required design changes so that special recommendations can be made to their customers. He said that CE disagrees with the NRC Staff approach.

4. General Electric Company (CE)

R. Nelson, GE, described the proposed General Electric short-term and long-term design changes. He concluded that the proposed ATWS mitigating system will

- provide a diverse shutdown system.
- impact on current systems.
- be suitable for Category "B" plants.
- will meet conventional criteria.

I. Jacobs, GE, discussed common mode failure analysis methodology, urging the use of probabilistic analysis for the solution of ATWS. He objected to the NRC Staff's arbitrary use of 10^{-4} and 10^{-3} for the reactor protection system and the diverse system failures respectively. He concluded that:

- common mode failure analysis supplies a discipline for seeking out potential failures.
- quantitative assessment of common mode failure probabilities is not absolute.
- credit must be awarded for common mode failure improvements in the reactor protection system.
- there is low common mode failure potential in control rod drives.

- electrical actuation logic is a major contributor to the common mode failure potential.

- Improvements are possible to reduce common mode failure potential in the reactor protection system to acceptable levels.

C. Statement from Public

T.W.T. Burnett, representing the Anticipated Transients Without Trip (ATWT) Working Group of the American Nuclear Society, discussed the work which has been done in developing the proposed standard, ANSI-N661. He stated his awareness that all view points may not be represented in the ATWT Working Group, but that ANS actively encourages the participation of all interested parties in the development of standards. In particular help is sought from the universities and the national laboratories. He believed that the NRC Staff evaluations of the vendor's ATWS analyses go beyond the scope of ANSI-N661, although the NRC Staff did approve of the drafts of the standard at each stage of development. He offered the opinion that the American public is due a cost-benefit analysis of the NRC Staff's position on ATWS to justify the higher cost of power that will result from the Staff's current requirements for the solution of ATWS. He proposed that the time has come to place monetary value on life, property, life expectancy, neighbor risk, etc."

Following the 189th meeting, the ACRS issued a report to NRC Chairman Anders (reproduced below) which endorsed the general approach and safety objectives adopted by the Staff, including the use of a goal of 10^{-7} per reactor year as the maximum probability from all causes of an ATWS with unacceptable consequences.*

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*The minutes of the 192nd meeting, April 8-10, 1976 record an ACRS position on acceptable risk taken in connection with preparation of a letter responding to ten questions posed by G. Murphy, Executive Director of the JCAE Staff. The position was as follows:

> The probability of an accident having serious consequences to public health and safety should be less than 10^{-6} per reactor year. A serious accident is one having consequences similar to that of the crash of a mid-sized jet airliner (approximately 150 passengers). It was generally agreed that simply exceeding the limits of part 100 would not necessarily constitute 'serious consequences'."

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 14, 1976

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Dear Mr. Anders:

At its 189th meeting, January 8-10, 1976, the Advisory Committee on Reactor Safeguards reviewed the Nuclear Regulatory Commission Staff's status reports on ATWS in water reactors and the analyses and proposals of four reactor vendors, The Babcock and Wilcox Company, Combustion Engineering, Inc., Westinghouse Electric Corporation, and General Electric Company, on this matter. Subcommittee meetings were held with representatives of the vendors, and with NRC Staff in Washington, DC, on December 11-12, 1975, and on January 7, 1976. The Committee had the benefit of the documents listed.

The Committee commented on the NRC Staff's proposal to revise the criteria for "Class A" plants, as categorized in WASH-1270 "Anticipated Transients Without Scram for Water Cooled Power Reactors," in a letter to Mr. Lee V. Gossick dated October 17, 1975. The Committee had previously commented on the regulatory position eventually published in WASH-1270, in letters to Mr. L. Manning Muntzing dated April 16, 1973, and May 10, 1972.

The ACRS endorses the general approach and safety objectives adopted by the NRC Staff including the use of a goal of 10^{-7} per reactor year as the maximum probability, from all causes, of ATWS with unacceptable consequences.

Implicit in the use of a probabilistic goal is the application of probabilistic methods in the analysis of the reactor systems. Although WASH-1400 provides assistance in this area, data for some systems are still sufficiently sparse that engineering judgment must be used, both in synthesizing the analytical models and in choosing appropriate input data. Under the circumstances there is a need for some conservatism in the choice of models and the selection of data. Even so, there are a number of approaches to the modelling that may prove to have equal validity and the ACRS suggests that both the NRC Staff and the vendors give further consideration to various alternatives. For example it may be feasible to treat the time-varying moderator temperature coefficient probabilistically.

The Committee also recommends that vendors be encouraged to continue to make design changes that decrease the probability of transients that are likely to cause difficulty and to make improvements that ameliorate the consequences. As appropriate these should be taken into account in the ATWS analysis. Continuing attention should be given to improving the reliability of the reactor shutdown systems.

During the course of these meetings comments were made which indicated that in some cases the NRC Staff needed further information from vendors in order to conclude its review of ATWS. The Committee urges that appropriate action be taken to obtain this information as soon as feasible.

The ACRS expects to complete its review of ATWS after further information has been developed and the Staff has completed its evaluation. The Committee urges that the NRC Staff and the vendors expedite efforts in this regard.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

However, the ACRS indicated a need for resolution of the arguments outstanding between vendors and Staff concerning what constituted an acceptable way of meeting the criterion.

So, ATWS was "almost resolved." However, there remained many complicating factors. Many representatives of the nuclear utilities and the reactor vendors turned to the results of draft WASH-1400 issued in August, 1974 and the final version issued in 1975, as a strong demonstration that ATWS was not a major contributor to risk from LWRs, and that hence the existing situation was satisfactory with no modifications, either in reliability of scram system or in the ability of the reactors to accept an ATWS without intolerable consequences.

General Electric concluded that implementation of the mitigating requirements defined in the Staff Status Report of December 9, 1975 would be very expensive,* and in a document dated September 30, 1976, proposed a different recommended solution to ATWS, namely incorporating a proposed Alternative Reactor Scram System as backup to the existing system so that NRC's ATWS safety objective of 10^{-7}/reactor year could be met.

In a memorandum dated November 24, 1976, ACRS consultant Epler discusses the General Electric proposal and remains unconvinced that the claimed reliability can be achieved thereby. And apparently, the Regulatory Staff also remained unconvinced.

ATWS remained a very controversial issue between the NRC and the industry, as is clearly illustrated in the exceprt on the following page from Nucleonics Week of October 7, 1976.

In the fall of 1976, a series of reports entitled "ATWS: A Reappraisal" was published by the Electric Power Research Institute (EPRI). In summary, the EPRI report re-evaluated the probability of failure to scram and estimated the risk to the public from ATWS. Using their assumptions and choice of data, the authors of the reports concluded that the probability of failure to scram was much lower than 10^{-4}/demand (by decades) and that ATWS posed insignificant risk to the health and safety of the public. The article reproduced from the EPRI Journal, March, 1977 summarizes the EPRI viewpoint.

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*In a letter to the NRC dated September 29, 1976, the Long Island Lighting Company state that "an expenditure of approximately $50 million would be required for the total installed cost of additional equipment and logic, but excludes costs of financing, outages or delays.

TWS—Impact of a Nonproblem

erald Lellouche

s to be hoped that by the time this
icle appears in print, the ATWS
ntroversy will have been resolved.
s doubtful, however, that a
blem (or as we shall show, a
nproblem) already eight years in
making will resolve itself so
ckly. ▫ An EPRI technical article

ld Lellouche is Program Manager of Statistical
Environmental Analysis in the Nuclear Power
on, EPRI.

ATWS is an initialism for anticipated transient without scram. In Nuclear Regulatory Commissionese it refers to a scenario in which an anticipated incident causes the reactor to undergo a transient. Such a transient would require the reactor protection system (RPS) to initiate a scram (rapid insertion) of the control rods to shut down the reactor, but for some reason the scram does not occur. The transient proceeds to a natural termination; potentially, the core is damaged and radiation may be released onto and beyond the plant site, resulting in property damage and personal injuries.

Several questions arise that affect this scenario. Scenarios are useful tools. They are used effectively by writers of fiction, the media, and others to guide the thinking process. Before passing from the scenario to reality, however, the question of how likely it is must be answered. Before insisting that plant design must be altered to effectively eliminate the problems in the scenario, questions concerning cost-benefit–risk reduction should be raised.

Dollar cost already in millions

Although the ATWS question has been with us with increasing impact since the late 1960s and in terms of manpower and computer time has probably exceeded a cost of \$10–\$20 million, an accepted answer to whether ATWS is real enough to require regulation has not yet been reached. In the important 1973 regulatory document WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," the AEC attempted to deal with the question of "how likely" as well as most other aspects of the ATWS. In many ways, the effort was less than successful.

WASH-1270 was particularly important in that it seemed to try to use rational methods (probabilistic/statistical) to determine whether an accident scenario was indeed worth regulating.

Unfortunately, WASH-1270 was a mixed bag. It applied elementary statistics to a situation that required a much higher degree of mathematical sophistication. It did not consider all the extant information and so ended with an incomplete data base. It concluded that the likelihood of the RPS failing to respond to a demand was less than 16 in 100,000 (1.6×10^{-4} per demand) with 95% statistical confidence (S-confidence) (*1*). It "picked" a value of 1/yr as the frequency of incidents that would lead to transients requiring scram, although it also stated that the actual frequency was more likely to be between 0.1/yr and 0.5/yr.

WASH-1270 identified a number of anticipated transient initiators that would strongly challenge the integrity of the system (if the RPS failed to act), but did not address the question whether any of them would indeed lead to consequences that would violate any of the out-of-plant radiation limits, such as 10CFR-100. It concluded that it was desirable that the probability for ATWS violating 10CFR-100 be less than about 1 in 10 million/yr (10^{-7}/yr). It also concluded that the total probability for all accidents (including ATWS) violating 10CFR-100 should be less than about 1 in 1 million/yr (10^{-6}/yr). But, it did not supply a basis for the choice of these numbers.

Since WASH-1270, these numbers (10^{-7}/yr for ATWS and 10^{-6}/yr for all accidents) have been repeatedly introduced by members of the NRC staff in NRC meetings, at the Advisory Committee on Reactor Safeguards, and in testimony at various hearings of the Atomic Safety and Licensing Board. The numbers have gradually achieved a stature that is largely undeserved because they bear no clear relationship to any underlying reality. We shall show, however, that reactors already have achieved most of this stringent requirement by considering a document that did not exist when NRC was writing WASH-1270. This newer document is WASH-1400, the reactor safety study.

The process by which the NRC staff identified serious potential transients was to require each of the vendors to perform various accident analyses. After reviewing these analyses of postulated ATWS events, the staff concluded that several anticipated transients in boiling water reactors would require prompt action to shut down the reactor in order to avoid serious plant damage and possible off-site effects. In PWRs, several anticipated transients would require rapid shutdown of the reactor to avoid pressure surges in the primary system that in some cases might exceed allowable limits.

The NRC staff found that the great majority of postulated ATWS events did not lead to serious consequences, but that design changes to improve protection against ATWS would be appropriate in anticipation of the large numbers of plants expected in the future. The point of importance here is twofold:

- No transient has been identified where the calculated off-site effects would exceed 10CFR-100.
- Most anticipated transients have no serious consequence.

Much time has passed since WASH-1270. Since 1974 the reactor vendors have been preparing increasingly sophisticated analyses of their particular systems. They have identified design changes that could lead to greatly increased RPS reliability; but NRC, while encouraging such endeavors, has failed to agree that any such changes would alter any of the values appearing in the now three-year-old WASH-1270. During this period the vendors have also identified system modifications that would tend to mitigate the consequences of an ATWS to the point where nothing significant would occur. The cost of making such changes was, however, not publicly addressed. Finally, in December 1975, the AEC regulatory staff issued a series, "Status Reports on ATWS," one report for each vendor. These reports identified a large number of significant differences between the NRC staff and each vendor. All these differences are based on the scenario method and have little to do with answering the question, How likely? Therefore, on the question of which values of lifetime varying parameters should be used, NRC requires that a value should be the worst during 99% of the cycle. A statistically valid procedure would be to repeat the analysis as a function of the variables and then average over the cycle. Thus, NRC requires that the initial conditions should be essentially at their worst (e.g., for PWRs the ATWS should be assumed to occur during a boron dilution procedure). Again, a statistically valid procedure would be to consider all the likely initial conditions, weight them with their expected time intervals, and average them. Thus, in performing the analysis, it must now be assumed that not only must the RPS fail, but other mitigating systems as well: that one relief valve does not open, and one of those that does open, does not close.

Is ATWS real?

How rational is any of this? Is ATWS real enough to warrant so much time and effort?

In October 1975 EPRI undertook the task of reappraising the entire rationale for making ATWS important enough to require regulation. The basic conclusions of the group doing the reappraisal can be stated to be that ATWS does not require regulation, with the possible exception of requiring an overpressure recirculation pump trip on BWRs. In December 1976 EPRI published the first two parts of this study: a definitive analysis of WASH-1270 (NP251) and an evaluation of societal risks due to RPS failure (NP265). Two other parts of this study will be published later.

Reality shows us that the world is less than perfect and that we cannot control, with perfect reliability, all things all the time. (A. Lincoln put it a little differently.) This implies that during the life of any power plant, events that are undesired from an operational viewpoint can be expected to occur with greater or lesser frequency. Statistically, we can be quite reasonably sure that some will occur on an average of once a year, while others may be as rare as only once in 30–40 years. The total number of such events in BWRs, for example, can be shown to depend on how c the plant is (Figure 1) and will vary (again a average) from about 23/yr the first year about 2/yr after five years of operation.

Some of these events will lead to transien that will not require the intervention of th RPS. Others would call for the RPS to sh down the system, but if no scram occurre nothing of a serious nature would result. F nally, some few events may call for a scra that if greatly delayed, would result in dan age to the core and potential radiation re lease to the off-site ecology.

What is the frequency of incidents?

The first question is, What is the frequenc of those anticipated events that would re quire scram to prevent core damage (antic pated events of consequence)? The tota expected number of events per year tha would require scram is precisely found fror the data for BWRs in Figure 1. Thus, afte about five years of operation we expect th upper limit to the frequency per year c all events to be about 2 for BWRs. Due to th difficulty of collecting the necessary data, more complete discrimination of this fre quency is not complete at this time. For th purposes of this article, the number of inci dents leading to transients of potential signif

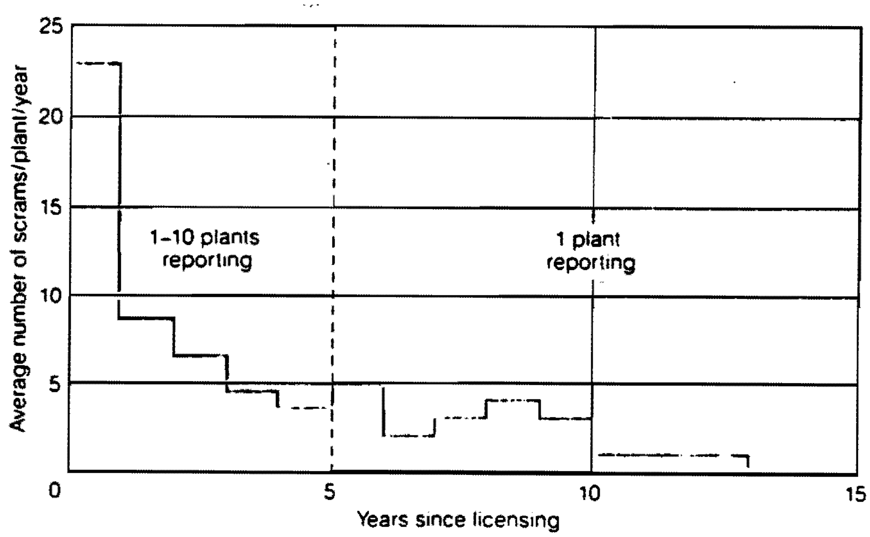


Figure 1 The learning curve with an average BWR shows sharp dependence on the length of time the plant has been in service; the number of scrams per year decreases with unit maturity.

ınce is assumed as this total number of ram initiators, a clear upper bound.

Given that in the long run one can expect e or two events a year that call for a scram, en though most of them will not cause oblems if the scram did not occur, the sec- d question of importance is, What is the obability that the RPS will fail to operate rrectly?

This question is more difficult to answer cause so few events have occurred and cause of the difficulty in correctly treating data. This latter point is important be- use we wish a realistic evaluation of the obabilistics, not one that is merely conser- ive. Two cases where the RPS was in- erative have been documented. The first curred in 1963 in the Kahl reactor in Ger- ny; the second, in 1970 in the N-reactor Hanford. The N-reactor instance is usually regarded because the N-reactor does not emble any commercial power reactor. In Kahl instance (a U.S.-designed 15-MWe /R that became operational in 1960), it s discovered on test that a scram signal uld not have been initiated if required be- se of a common-mode failure (CMF) in scram relays.

he N-reactor instance should not be in- ded in the data because of the extreme ign disparity between the N-reactor (a phite-moderated, cartridge configuration led dual purpose unit) and any commer- LWR. The German BWR instance like- e should not be included because of the cept of rectifiability. That is, any CMF is discovered is not expected to occur in (certainly not with the same frequency) e redesign, test and maintenance, and/ other quality assurance methods will be sted to eliminate that particular failure de. Thus rectification eliminates potential ure modes and produces a better-than- inal condition.

the purpose is to secure a realistic view ystem failure, one must be very careful in ning a CMF. Of interest is the class of ini- ors that will lead to a failure of all or nearly he control rods in the RPS in such a way the scram activation mechanism ap- rs to fail in a time interval that on average ss than half of the test interval. There are ditions that will affect the entire RPS, but ause of time considerations, they are not dible as CMF initiators in that they are ob- able before failure.

o this class belongs the thermal stress ini- or, which leads to collet cracking. The period for actual failure of a single collet eater than one year for this mode, and in no drive failure by collet cracking has n observed. A second class of CMF initi- s that statistically should not be included in ATWS probabilistics are those which would be discovered during startup testing or ear- lier. Thus, the initial inability of any single rod or bank to scram would be discovered during the hot zero- and low-power testing that is required of each reactor.

One is left then with a reduced class of po- tential CMF initiators that either take a num- ber of months of actual reactor operation to develop (longer than the startup period) or, because of maintenance or RPS modification subsequent to startup, are externally intro- duced. In this latter class is the only known case of inability to scram in a commercial power reactor (the Kahl instance), where the replacement for a set of contacts (2) was faulty and they stuck closed. Quality assur- ance (QA) modifications were made in LWRs so that such incidents would not go undis- covered again.

Note that we do not have to assume that such a fault will not occur to remove it from the class of credible CMF initiators, but only that it will be discovered by special testing as it occurs. Thus, the fact that 4 of about 200 contacts were discovered to be faulted sev- eral years later during startup testing of an- other reactor does not imply the failure of QA administrative methods (*3*). It actually vali- dates them (in that particular case) because the special testing of the new system did in fact bring the failures to light. Thus, rectifica- tion is a valid concept to use in eliminating certain types of known CMF initiators from consideration.

Statistical confidence concepts

Another point of interest is the use of high- level S-confidence bounds. Conceptually, the ideas of statistical confidence are most meaningful in sampling theory where one wishes to have information about a popula- tion (or lot) by sampling a relatively small por- tion. Here also the basis is that the underlying phenomena remain the same from sample to sample. If this were not true, any fixed sam- pling procedure would be invalid. It also has little meaning when one is sampling every member of the population (as one does with scrams). For this reason, the choice of any very high S-confidence level based only on previous failures that have been rectified is highly conservative.

The quantification of this discussion in Table 1 illustrates what a realistic treatment of the data implies.

The use of a median S-confidence limit of 50% is not arbitrary on our part but is consis- tent with Regulatory Guide 1.108, which de- scribes an acceptable statistical scheme for testing diesel generators. The implications of the above calculation are that on the basis of data alone, WASH-1270 was conservative by a factor of 50 in its calculation of the RPS unavailability per demand.

One need not rely on data alone to make an estimate of the statistical properties of the RPS. Obviously, before the first RPS is installed we have almost no data, and for systems with great reliability, (due to redun- dancy, for example) it may be many years or even centuries before sufficient data to yield "good" statistics are accumulated. Nor- mally, we make use of engineering judg- ment, heavy over-design, modeling studies, and so on, to give us confidence that the sys- tem will do its job. It is possible to incorporate such information into a more elaborate statis- tical treatment by means of Bayesian estima- tion. We shall return to this later.

The upperbound number 3×10^{-6} per de- mand for RPS unavailability is quite low, and if we multiply it by 1 or 2 for the upperbound on total event yearly frequencies, we still have a low number. But not yet 10^{-7}. Let us now ask a few more questions. Whatever the ATWS number may be—does that imply that we should expend time and effort to make the number smaller? What about priorities? What benefit accrues to the public if we wipe out

Table 1

RPS UPPERBOUND UNAVAILABILITY

(per demand, based on data alone)

	50% S-confidence	*95% S-confidence*
WASH-1270	3.8×10^{-5}	1.6×10^{-4}
WASH-1270 (update to 1976)	2.1×10^{-5}	8.9×10^{-5}
EPRI (1976 evaluation)	3.0×10^{-6}	1.3×10^{-5}

ATWS? How do we get a handle on these questions?

WASH-1400 contains all the information needed to answer these questions. WASH-1400 considered all the possible ways (or as many as the authors could think of) that a reactor could undergo a trauma of sufficient magnitude for extensive core damage to result with a greater or lesser release of radiation, and it incorporated meteorological effects to carry the radiation into the off-site ecosystem. Since the authors also included ATWS events in their analysis, in WASH-1400 we have a basis for establishing what portion of the total risk (expressed, for example, as a 30-day whole-body dose in rem at the exclusion boundary) is attributable to ATWS. We can also determine the probability of violating 10CFR-100.

To be able to defend the ATWS results of such an editing of WASH-1400, the reappraisal group reconstructed all the fault trees considered important. No significant changes were discovered to be needed during this reconstruction. It was found that updating the data and modifying one statistical model were all that was necessary. The results of this work show that the percentage of risk attributable to ATWS (based on WASH-1400 updated) is, for PWRs, only 0 5%; for BWRs, 5.0%.

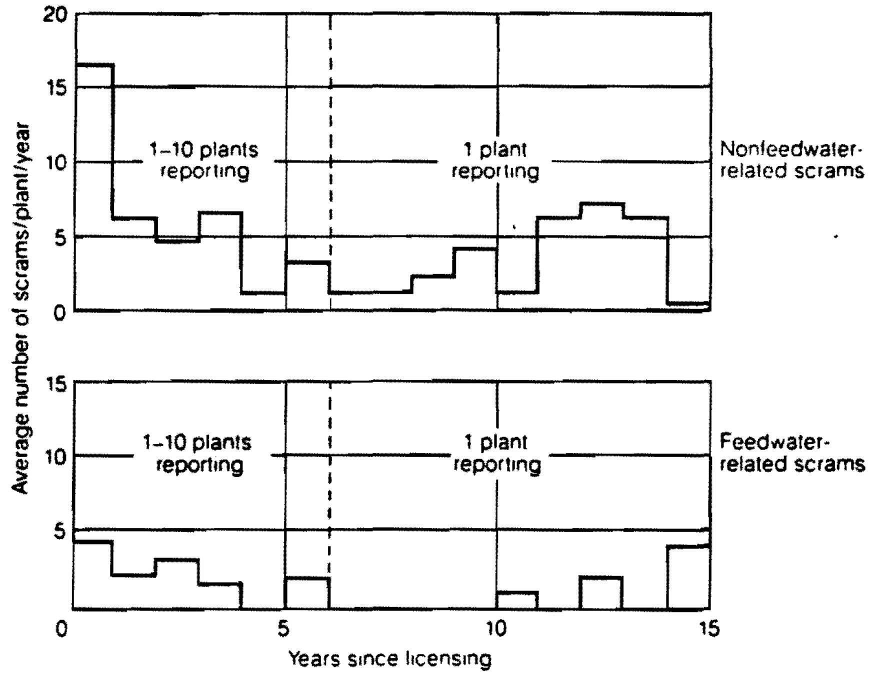
Risk studies of this type yield information on whether a particular component, subsystem, or system should perhaps be redesigned or backfitted to increase reliability. From the viewpoint of cost-benefit–risk considerations, it would seem that a situation responsible for less than 5% of the total potential risk (measured in man-rems, 30-day whole-body dose) would not be an early candidate for either redesign or backfitting unless the costs were low enough to make it an incidental expense.

One might conclude that BWRs have 10 times the ATWS risk of PWRs, but careful examination of the data input to the fault trees from areas where we were unable (because of the need to preserve a balanced estimate) (*4*) to alter data input shows sufficient conservatism to smooth out this difference.

Probabilities determined

In a second edit of the WASH-1400 data, we determined the probability of violation of 10CFR-100. For this study we were no longer interested in comparing one portion of the risk fraction with another; hence we could consider updating more of the data than in the risk comparison. For example, WASH-1400 assumes 10 transients per year with a range of 5–20. Figure 1 shows that these

Figure 2 The learning curve with an average PWR also shows a fall in scram initiators per year, but distinction must be made between those that involve a loss of feedwater and those that do not.



numbers should be closer to 2.5/yr with a range of 0–5 for BWRs. Figure 2 shows that PWRs exhibit the same sharp fall in scram initiators per year as BWRs, but here we have to discriminate between scram initiators involving loss of feedwater and no loss of feedwater. The former have a long-term expected average of about 1/yr, while the latter have an average of about 4/yr and the ranges are about 0–2 and 0–8.

Another type of update has to do with the RPS failure probability for the two types of reactors. In each case, WASH-1400 used schematic diagrams to construct a fault tree that models the RPS system. We corroborated these trees.

In quantifying the trees further, modeling of such phenomena as common-mode miscalibration of instruments and test and maintenance errors had to be made. In both cases (instrument miscalibration for BWRs and testing and maintenance for PWRs), the quantifications are strongly conservative—so much so that the RPS unavailability is completely dominated by these items (93% for PWRs and 98% for BWRs). The hardware and electronics are responsible for essentially none of the unavailability. It is not difficult to attack these aspects of the input data, but it is not terribly important to do so at the present time. The BWR modeling yields results consistent with the EPRI (rectified) case in Table 1, while the PWR yields unavailabilities that are a factor of 6 or more larger than the scram data alone would imply. In any event, if we incorporate the latest values for the expected number of transients and the modeled estimates for scram unavailability, we can use the WASH-1400 consequence model and determine the median upper-bound probabilities of exceeding 10CFR-100. For PWRs, this probability per year is 1.7×10^{-7}; for BWRs, 7×10^{-7}.

These are both for exceeding the 25-rem two-hour iodine thyroid dose If we considered the other aspects of 10CFR-100 we should in all cases find much lower values. One sees here that based on a common quantified consideration, the PWR and BWR achieve a closer comparison. Since the input to these calculations is, in our estimation, quite conservative, we would expect both these numbers to drop and perhaps still show a difference between reactor types, but at this low level of probability, further pencil-sharpening seems wasted.

Table 2

SUMMARY OF MEDIAN RPS UNAVAILABILITIES PER DEMAND AND THE EFFECT ON 10CFR-100 VIOLATION

actor	*WASH-1270*	*WASH-1400 (updated)*	*EPRI*	*EPRI Bayesian Estimate*	*EPRI Bayesian Probability of 10CFR-100 Violation (per year)*
WR	3.8×10^{-5}	2.3×10^{-6}	3×10^{-6}	3.4×10^{-7}	3.3×10^{-7}
WR	3.8×10^{-5}	5.1×10^{-5}	3×10^{-6}	9.0×10^{-6}	7.7×10^{-8}

One further aspect should be considered. e fault tree models for RPS unavailability ed in the WASH-1400 studies are based on very small fraction of the information used establish Table 1. Because of this we can fold the two results through use of the ayes equation to produce a more sophistited estimation of scram unreliability *(5)*. If e do this, we obtain the results shown in ble 2.

TWS is a nonproblem

e results of these studies indicate to the eople who did them, as they do to the writer, at except for assurances that the WASH-00 analysis applies to other reactors (the WR had a recirculation pump trip, for example, which does not appear in some of the der BWRs) ATWS is a nonproblem with a obability of occurring that is terribly small d an even smaller probability of significantly affecting the health and welfare of the iblic.

Still, one might reasonably ask, How difficult and expensive would it be to eliminate TWS apart from any other considerations? d as a corollary, ask, What benefits can e expected to accrue to the public (who in e end must pay for any such design modifitions)? For a number of years, members of e NRC staff have estimated that a complete TWS "fix" would cost only a few hundred ousand dollars, while vendors have felt at it would cost a few million dollars. This fference was due to the fact that no one had t together an actual cost estimate of backting a plant. Because NRC has required ach plant to submit proposed plant design visions, it is now possible to establish acal cost estimates. These range from $20 illion to $50 million per plant. The dollar osts, then, are very high.

If the purpose of WASH-1270 was to establish that ATWS is indeed real enough to require regulation, it arrived at an erroneous conclusion because the results presented here, insofar as they demonstrate probabilistics in the neighborhood of 10^{-7}/yr, contradict WASH-1270. Further, the technical basis for arriving at these results is much more extensive and better documented than that in WASH-1270.

There are, however, those who maintain a "hang the cost" attitude and would require elimination of ATWS, no matter what. If their view prevails, then the public will pay to eliminate a nonproblem.

Notes and references

1. S-confidence is a mathematical term meaning the probability that the value of a parameter (in this case, a failure rate) is less than some specified amount. One can write it in this case as P (rate is less than 1.6×10^{-4} per demand) = 0.95. The actual value of the rate may lie anywhere between zero and 1.6×10^{-4} per demand without altering the value of the right side of the equation.

2. Such replacements are made on a scheduled basis.

3. U.S. Atomic Energy Commission. "Instances of Relay Failure in Reactor Protection Systems." *Reactor Safety Operating Experiences.* ROE 71-16, AEC 1971.

4. Since a comparison was being made between the ATWS portion of risk and the total risk estimate in WASH-1400, it was not possible to alter values in the ATWS calculation that would induce alterations in the rest of the risk calculation (funds, manpower, and time available provide constraints).

5. The Bayesian approach is based on the concept of conditional probability (given that A is true, what is the probability of B occurring). Thus it is possible to incorporate different types of probability estimates of the same phenomena to produce a single overall estimate. The ease of such incorporation is enhanced by the independence of the original estimates.

The Regulatory Staff held meetings with EPRI on February 16 and February 22, 1977 to discuss the EPRI reports. The Staff comments (as given in a memo dated June 2, 1977) included the following:

> The Staff met with EPRI on February 16, 1977 and February 22, 1977 to discuss the EPRI studies on ATWS.
>
> At the February 16th meeting the Staff presented their estimates of the scram system unreliability and comments on EPRI evaluations (Enclosures 2 and 3). In summary, the staff noted that the probability of scram system failure due to a common mode failure is approximately 10^{-4} per reactor year based on experience. The Navy data were included to confirm the Staff judgment that protection from ATWS events was required. The Staff further noted that the probability of unacceptable consequences from ATWS events is also approximately 10^{-4} per reactor year.
>
> The Staff comments on EPRI methodology were (See Enclosure 3 for details):
>
> (a) EPRI treatment of rectification could lead to biased estimates.
>
> (b) The EPRI expression misused statistical confidence limits, probability percentiles, and Bayesian posterior probability percentiles.
>
> (c) Expression for Prob (ATWS) in the case of an assumed constant failure probability was in error by a factor of 2.
>
> (d) EPRI model for multiple control rod failures yields results which are inconsistent with EPRI data. The sources of the discrepancy were their binomial assumptions and their square root model for the probability of multiple failures.

The EPRI representatives did not agree with all the Staff comments, needless to say.

However, consultants to the ACRS also supplied comments critical of specific aspects of the statistical treatment in the EPRI reports and the assumptions concerning treatment of data. And ACRS consultant Epler, in a memorandum to the ACRS dated November 26, 1976, recalled that in a letter to the ACRS dated October 28, 1974, he had pointed out a basic fallacy in draft WASH-1400 in the determination of scram unreliability as it applied to ATWS, and noted that this error had carried over to the final version of WASH-1400 and had been missed in the EPRI-SAI study on ATWS. In effect, Epler argued that the EPRI study had devoted considerable effort to a re-evaluation of scram unreliability as it applied in the case of a LOCA, not ATWS, and that the study was not relevant to ATWS.

In March, 1977 the NRC FOUND A TASK FORCE ON ATWS in an effort to finally resolve the matter. Meetings were held with the LWR vendors. And in July, 1977 the Staff reported once again to the ACRS, reiterating their general position of December, 1975 that scram unreliability could not be shown to be acceptably low and that measures were required to mitigate the consequences of ATWS.

It is, in a sense curious, that with this relatively unchanged position by the Staff over a period of many years, little was done to implement any actual changes with regard to ATWS. For boiling water reactors, for which the question first arose, and for which at least one mitigating design change, the recirculating pump trip, was included as a partial backup to scram in all BWRs receiving construction permits after the Newbold Island - Limerick reviews in 1971, many, if not most of the operating BWRs had not incorporated this feature by 1976. On learning of this situation in early 1976, the ACRS sent a memorandum to Lee Gossick dated March 12, 1976 advising that steps be taken immediately to include this change in operating BWRs, unless it was not needed to make the consequences of ATWS intolerable. And the Regulatory Staff shortly thereafter issued letters to all operators of BWRs asking for their specific plans to incorporate this feature. However, by mid-1978 most of the operating BWRs had not implemented pump trip.

Also, the year 1977 passed without issuance of a new Regulatory Staff position on ATWS, and this generic item remained unresolved eight years after its inception.* Much had been learned about the subject. There obviously remained wide differences of opinion concerning the safety significance of the matter. And, reactors continued to be designed and receive construction permits without incorporation of mitigating features.

In April, 1978 the Regulatory Staff issued a new report, NUREG-0460, entitled "Anticipated Transients Without Scram for Light Water Reactors." Excerpts from the summary follow below:

> Based on the occurrence of transients in currently operating nuclear power plants, the Staff now concludes that transients that would result in serious consequences if accompanied by scram failure could be expected to occur in the future population of plants at a rate of five to eight per reactor-year. We also estimate that the probability of scram failure, based on nearly 700 reactor years of operating experience in foreign and domestic commercial power reactors with one observed potential scram failure, is in the range of 10^{-4} to 10^{-5} per demand.

*Although it was listed as resolved on the ACRS Generic Items list after WASH-1270.

Thus, the expected frequency of ATWS events that could result in serious consequences is approximately 2×10^{-4} per reactor-year. We recommend that a safety objective of 10^{-6} unacceptable ATWS events per reactor-year is more appropriate, and therefore, that some corrective measures to reduce the probability or consequences of ATWS are required.

Although reducing the frequency of anticipated transients might be a means of reducing the probability of ATWS events, the difficulty in accomplishing the necessarily large reduction appears to make this approach impractical. Alternatively, improvement of the reliability of scram systems, particularly with regard to potential for common mode failures, by providing a second independent, separate and diverse scram system has been considered, but no completely acceptable design has been proposed. These considerations lead us to recommend that the provision of systems to mitigate the consequences of ATWS events, should they occur, is the most promising alternative for meeting the safety objective. This approach has been the principal subject of the development, analysis and staff review presented in this report.

We have developed a set of requirements for the design and performance of systems provided to reduce the consequences or probability of ATWS events. Acceptance criteria are stated that address radiological dose limits; reactor coolant system, fuel and containment integrity; core cooling capability; and mitigating system design and performance. Requirements are given for the analysis of postulated ATWS events. The requirements would provide reasonable assurance that, considering the frequency of ATWS events, the probability of additional system failures, and the uncertainty and variation in initial conditions and parameters, the acceptance criteria are not violated.

We have also considered the value and impact of these requirements. Estimates of the impact, primarily the costs associated with implementing the requirements, range from 1 to 43 million dollars per plant, depending on the type of plant and its stage of construction or operation. The direct value consists of the cost of the averted radiological and economic consequences. Estimates of the value range from approximately 1 to 47 million dollars per plant and are generally larger than the corresponding impact for any one type of design. The averted potential for shutdown of a number of operating reactors, should an ATWS with severe offsite consequences occur, has been estimated to translate into an additional indirect value ranging from 1.5 to 23 million dollars.

> We have found that, considering the expected frequency of occurrence of transients, the reliability of current reactor scram systems necessary to meet the safety objectives has not been demonstrated and may well have not been attained. Therefore, we recommend that means of reducing the probability or consequences of ATWS events should be provided. Furthermore, we envision that the initiation of rulemaking to incorporate ATWS requirements in the Commission's regulations would fairly and clearly resolve the long standing uncertainty in the status of regulatory requirements in this area.

From the early reactions to the Staff position of April, 1978 it appeared difficult to ascertain whether ATWS was finally headed toward early resolution. The reactor vendors, the utilities and the Electric Power Research Institute vigorously opposed the Staff proposals.

In early 1979 after issuance of the report of the Risk Assessment Review Group concerning WASH-1400 (NUREG1CR-0400), the Regulatory Staff issued a revised position on ATWS, one which strongly reflected the difficulties in backfitting an operating plant or even a plant under construction. That is, for such plants, emphasis was placed on those changes in circuitry that might provide increased scram reliability, while for plants to be constructed, the emphasis was shifted to hardware changes to mitigate the consequences of an ATWS should it occur, that is to keep pressures and temperatures below acceptable limits.

As of the spring of 1979, however, the matter remained unresolved.

5. SEISMIC RISK

5.1 SYNOPSIS OF SEISMIC AND GEOLOGIC SITING CRITERIA

General Design Criterion 2 of Appendix A to Part 50 of Title 10, Code of Federal Regulations (10 CFR) requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, etc., without loss of capability to perform their safety functions. Appendix A to 10 CFR Part 100 sets forth the criteria pertaining to the effects of earthquakes and other geologic phenomena. These criteria also describe the nature of investigations and determinations required to determine site suitability and to provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public.

Appendix A to 10 CFR Part 100 requires that each applicant for a construction permit investigate all seismic and geologic factors that may affect the design and operation of the proposed nuclear power plant irrespective of whether such factors are explicitely included in the criteria. For sites located in areas having complex geology or in areas of high seismicity, the siting criteria provide for additional and/or more conservative design determinations than those specified in the regulation. Provisions are made in the regulation to also allow an applicant to depart from satisfying specific sections of the criteria if he provides supporting data whcih clearly justify such departures.

The detail and scope of the required investigations are such that the geologic, seismic, and engineering characteristics of a site and its environs are well enough understood to permit an adequate evaluation of the proposed site.

There are four major considerations required by the regulation. These are: (1) investigations for determining potential ground shaking due to earthquakes for design purposes; (2) analysis of possible surface fault movement at the site; (3) evaluation of the possibility for flooding at coastal, lake, and river sites due to earthquake disturbances; and (4) a consideration of the effects on the site of other adverse geologic conditions such as landsliding, subsidence, cavernous collapse, and liquefaction. The requirements of Appendix A are discussed below.

1. Ground Motion Due to Earthquakes

The design basis of safety features for each power plant facility must take into account the potential effects of two levels of earthquake motion. The greater motion represents a maximum earthquake potential and is designated the Safe Shutdown Earthquake (SSE). The lesser motion represents an earthquake event expected during the life of the plant and is named the Operting Basis Earthquake (OBE). Concerning the size of the region to be investigated and the type of data pertinent to the investigation, Appendix A to Part 100 indicates that it will be based on the nature of the region surrounding the site. It is further stipulated in the siting criteria that the investigations will be carried out by both reviewing pertinent literature and doing field investigations.

The design bases for the earthquakes must be determined through evaluation of the geology, and the geologic and seismic history of the site and the surrounding region. A determination is required, in this evaluation, of influences due to man's activities and local site soil conditions. To carry out these determinations the types of soils and/or rocks present at the site and in the region surrounding the site must be determined. This type of information is required in order to describe the geologic conditions at the site and the surrounding region and establish their geologic history. In addition, an evaluation of the soil characteristics of the site are required since these materials can affect the transmission of earthquake induced motions to the proposed plant's foundations as well as their stability. Earthquakes which may cause ground shaking acceleration of at least one-tenth the acceleration of gravity at the site's power plant foundations must be considered. These considerations include the application of appropriate empirical relationships between the size of earthqukes and ground motion. In addition, a comparison is made between the soil characteristics at a power plant site and at the epicenter of the controlling earthquake relative to the transmission of earthquake motion.

The largest earthquakes occurring in the site region must be assessed for their impact on design. This must be carried out through a determination of whether or not those earthquakes, within 200 miles of the site, can possibley be correlated with geologic structure. Those earthquakes which cannot be correlated with geologic structure must be associated with regions containing similar geologic structural features (tectonic provinces). An evaluation is required to determine whether faults in the site region could generate earthquakes and be of significance to the earthquake design bases.

After the above-described required investigations have been completed, a determination of the nuclear power plant's earthquake shaking design bases must be carried out. The determination should be made by an evaluation of the resultant ground motion at the site resulting from the earthquakes that are associated with geologic structures, faults, and tectonic provinces (regions of similar geologic structure).

To compensate for the limited data on the distribution, occurrence and causes of earthquakes, the Safe Shutdown Earthquake (SSE) must be derived in a conservative manner. As a minimum, the maximum acceleration of the SSE is reuired to be one-tenth the acceleration of gravity. The maximum acceleration of the Operating Basis Earthquake is required to be at least one-half the SSE.

2. Surface Fault Movement

Appendix A requires a determination to be made as to whether and to what extent a nuclear power plant need be designed to withstand differential ground displacement due to movement of a fault. To make this evaluation, the regulation requires the following:

(a) For faults greater than 1,000 feet long and within 5 miles of a site, a reasonable investigation must be made as to whether faults are capable faults. Appendix A defines a capable fault as a fault which has exhibited certain characteristics. These are fault movement within the last 35,000 years or multiple movement within the last 500,000 years, macroseismicity, or relationship to another capable fault such that movement on one could reasonably be expected to result in movement on the other. The regulation specifies a number of geologic and seismic investigations, which are essentially like those required for ground shaking, to be performed to determine if a fault is capable.

(b) For a fault identified as capable an analysis is required of fault geometry, past movement and historic earthquake activity associated with the fault, to assess the potential for causing surface displacement at or near the site. The largest magnitude earthquake related to the fault is used to determine the width of the zone which must be investigated.

(c) To determine finally whether a plant should be designed to withstand fault movement, the regulation requires consideration of the location of the plant with respect to the fault. If the plant must be designed to withstand fault movement the regulation stipulates the specific safety features which are required to be designed for such movement. Appendix A requires that the design of these features provide reasonable assurance that they will remain functional if such an event occurs.

3. Seismically Induced Floods

For nuclear power plants near bodies of water, investigations are required to determine whether seismically induced floods could affect the site. Included in the required assessment of such sites is the gathering of geologic and seismic data appropriate to establish the design basis for flooding due to local offshore earthquakes, onshore seismic events, or coastal subsidence. The data needed for this assessment are determined by procedures similar to those for determining ground motion and the potential for surface faulting discussed above.

At sites located near lakes and rivers, investigations similar to those for coastal areas are required. In addition, the consideration of possible effects of upstream dam failure and landslides is required.

4. Other Adverse Geologic Conditions

Appendix A to 10 CFR Part 100 give guidance of a general nature concerning investigations and considerations to be given to other geologic hazards which may bear on the safety of the plant, such as sudden collapse of foundation material in areas susceptable to solutioning by ground water, landslides, liquefaction, subsidence (natural or man induced), etc.

5.2 EARLY DEVELOPMENTS IN SEISMIC SAFETY

One of the first mentions of seismic matters in the minutes of the statutory ACRS occurred at its 12th meeting, December 11-13, 1958. In connection with discussion of the site for the Carolina-Virginia Tube Reactor (CVTR) 25 miles northwest of Columbia, South Carolina, mention is made of the severe Charleston, South Carolina earthquake of 1886. No seismic design basis for the proposed reactor is mentioned.

The minutes of the 19th meeting, September 10-12, 1959, report that the proposed site for the Humboldt Bay reactor is in a moderate to heavy earthquake region and that seismic safety factors are slightly more conservative than used for conventional power plants.

According to the "Survey on Reactor Safety for Nuclear Power Stations," prepared by a study group on nuclear safety for the Japan Atomic Industrial Forum, studies and investigations on "nuclear reactors and earthquakes" were underway in the U.S. in 1959; however, aseismic design was hardly studied in the construction of four nuclear power plants in the eastern part of the United States, because earthquakes in the region were considered to be small (Shibata, 1970).

At a special meeting, March 5, 1960, the ACRS prepared a hurried response to a letter from General Manager A. R. Luedecke of the AEC concerning the suitability of sites in southern California. Excerpts concerning seismology follow below.

> Seismology. Holmes & Narver personnel were present for the discussions relative to earthquake hazards. Mr. Booth reviewed briefly the earthquake features of the California area as to location, frequency, and the scale for expressing the intensity. The larger shocks have given an acceleration in the horizontal direction of about one-third of gravity, "g", (measurements are such that this might be in error by a factor of two); the vertical component has always been less than the horizontal.

At the 25th meeting, May 5-7, 1960, the ACRS considered the 40 MWt Experimental Low Process Heat Reactor for a site at Point Mugu, near Oxnard, California. The minutes record no discussion of earthquakes beyond the statement that no known faults are in the immediate area. Tidal waves (tsunamis) are described as occurring every five to ten years with heights of the order of five feet, due to distant earthquakes.

At the 40th meeting, March 29-31, 1962 the ACRS reviewed the Haynes Point site, near Long Beach, California and the Corral Beach site in Malibu. Both were relatively highly populated sites, Haynes Point being much more so. The minutes report the following presentation by Dr. George W. Housner, a consultant to the Los Angeles Department of Water and Power:

The problem of vertical shock has not been considered in structures to date, because they are, of course, designed to resist one "g". About 2,000 square miles might be affected with a 77-year frequency by such a large shock. The probability of quakes is about constant over California, except in the extreme northeast portion and the upper part of the San Joaquin Valley.

There are four seismic design codes for structures, namely: (a) An overall State Building Code, passed in 1933 after the Long Beach shocks, requires a minimum of design for 2 percent of "g" in the horizontal direction for all structures, which is a low figure that does not much change normal building practices; (b) Uniform Building Code requires 13 1/3 percent of "g" (certain structures must provide for 100 percent of "g"); (c) The San Francisco Code requires only 7 1/2 percent "g" but with special calculations required showing the effects of resonances and the like; (d) The Los Angeles Code is much like the Uniform Building Code, but with more stringent features for certain buildings. The basis for all codes is the protection of the public against damage to "life and limb", and the experience has been fairly satisfactory.

Several possible results of reactor accidents from earthquakes were outlined, and the release of radioactive material was the most important. Mr. Booth concluded that simple construction precautions should eliminate the possibility of release of radioactivity by earthquakes. However, although the frequency is less, earthquakes have been known to occur in nearly all parts of the United States, and since special seismic design features are not usually included in reactors, earthquake damage to these other reactors might lead to more serious consequences than would a shock in California.

Bechtel described the method of including seismic design factors. The geology of the area and seismic history is used to set the horizontal acceleration requirements. Sites are chosen at least a mile from a known fault. The heavy construction around the region for shielding is adequate for seismic forces. The design of any tanks for the reactor facility would be conventional; however, the requirements for containment vessels might be eased somewhat, because these are shells while most tanks are build to contain liquids. The seismic design factors for plants are about the same as for wind load.

Bechtel includes seismic design factors for all power plants build in the West, and somewhat higher values for plants in California. A common figure is 20 percent of "g" for horizontal acceleration. Most of the California coast construction is with values of 10 to 15 percent. The value for a conventional tank and a one-story building might be 10 percent and 13 percent, respectively. Bechtel's seismic horizontal design figure would be 25 percent for the containment sphere and about 20

> percent for the balance of the plant for the 360 MWe.
>
> With respect to earthquakes, the heavy concrete structure required for the Humboldt Bay plant to sink the suppression chambers, because of the high water table, are such that little additional design is required for earthquakes. However, at other sites seismic factors may have to be included. Waves caused in the suppression pool by an earthquake could be minimized by baffling to prevent water loss.

In its letter March 6, 1960 the ACRS said the following:

> In reply to the request for an advisory report from the Advisory Committee on Reactor Safeguards on the feasibility and acceptability of locating the proposed reactors in the Los Angeles area and in an area within a fifty-mile radius of San Francisco in terms of the possible hazards associated with inversion and earthquake conditions, the following advice is given.
>
> With respect to seismic considerations, we understand that it is present utility industry practice in California to locate generating stations at least one mile from known surface faults; and to design and construct these stations using local codes supplemented by special analyses and increased seismic design factors for those critical plant components necessary to maintain the station on the line. In addition, in the case of a nuclear reactor facility, special analyses and increased seismic design factors are needed for those reactor plant systems whose failure could result in a release of radioactive material. With these precautions, the Committee believes the reactor facility would be adequately protected against seismic disturbance.

At the 25th meeting, May 5-7, 1960, the ACRS considered the 40 MWt Experimental Low Process Heat Reactor for a site at Point Mugu, near Oxnard, California. The minutes record no discussion of earthquakes beyond the statement that no known faults are in the immediate area. Tidal waves (tsunamis) are described as occurring every five to ten years with heights of the order of five feet, due to distant earthquakes.

At the 40th meeting, March 29-31, 1962 the ACRS reviewed the Haynes Point site, near Long Beach, California and the Corral Beach site in Malibu. Both were relatively highly populated sites, Haynes Point being much more so. The minutes report the following presentation by Dr. George W. Housner, a consultant to the Los Angeles Department of Water and Power:

Dr. Housner [sic] went into detail about the California seismology and concluded that reactor designs could be made to withstand any earthquake shocks expected in the area. He sketched a map showing the major and minor faults; the major faults are the preferred loci for earthquakes. New faults are not expected but many old faults may be covered and their location is unknown which indicates it is better for a reactor to be a reasonable distance from a known fault. The western part of the State is moving northwest relative to the eastern part of the State at about 2 inches per year, and a main fault is the San Andreas [sic]. The active earthquake zone is near the coast and extends inland in lower California. Strain surveys have been made for thirty years; recent measurements indicate a fairly large shock is due. A formula with an intensity factor was given for the earthquake energy release in ergs. The San Francisco earthquake (intensity 8.3) involved a release of strain energy over a distance of about 200 miles. The recent Chile and Long Beach quake had intensities of 8.5 and 6.2, respectively. This latter involved a minor fault not far from Haynes Point

Dr. Housner [sic] said the high frequency components of an earthquake are damped out by distance so that a quake in the San Andreas [sic] fault would result in only the lower frequency energy reaching Haynes Point; he estimated that only equipment at Haynes Point with a period of over 1 second would be effected. A local earthquake would have additional effects. A plot of the spectrum intensity of vibrations from a quake against period peaked at a .3 or .4 seconds. Figures were given for the acceleration and the duration of the recent El Centro earthquake in the Imperial Valley.

The local geology determines the ground motions, and Corral Beach is probably better from this point of view because the Haynes site has more alluvium. Sometimes the soils consolidate and sink as at Long Beach, but Dr. Housner [sic] does not believe this would be very serious for Haynes Point.

In its report of April 4, 1962 on these sites (which was primarily concerned with population densities), the ACRS stated "The Committee believes that it is possible with present engineering technology to overcome the potential danger from serious consequences of a major earthquake."

In the spring of 1963, the ACRS completed a first review of the proposed Bodega Bay BWR, at a site less than 1/4 mile from the edge of the San Andreas fault. This landmark case is discussed in very considerable detail in the following section.

5.3 BODEGA BAY

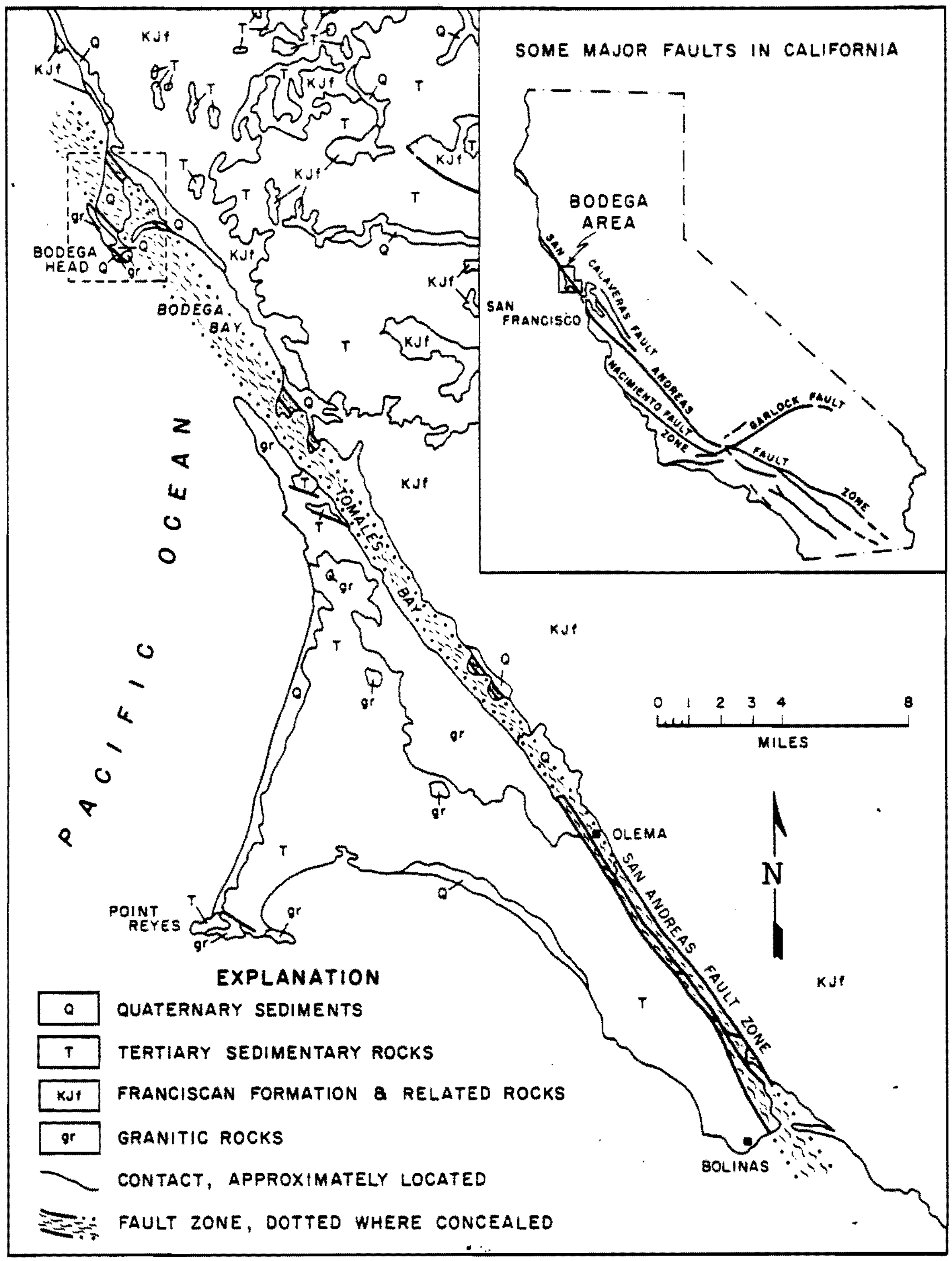
In the early spring of 1963, the ACRS completed (for the first time) its review of the application by Pacific Gas and Electric Company to construct a 1008 MWt BWR at a remote site on Bodega Bay, Sonoma County, California. The unique aspect of the review lay in the fact that the site was only about 1000 feet west of the San Andreas fault Zone.

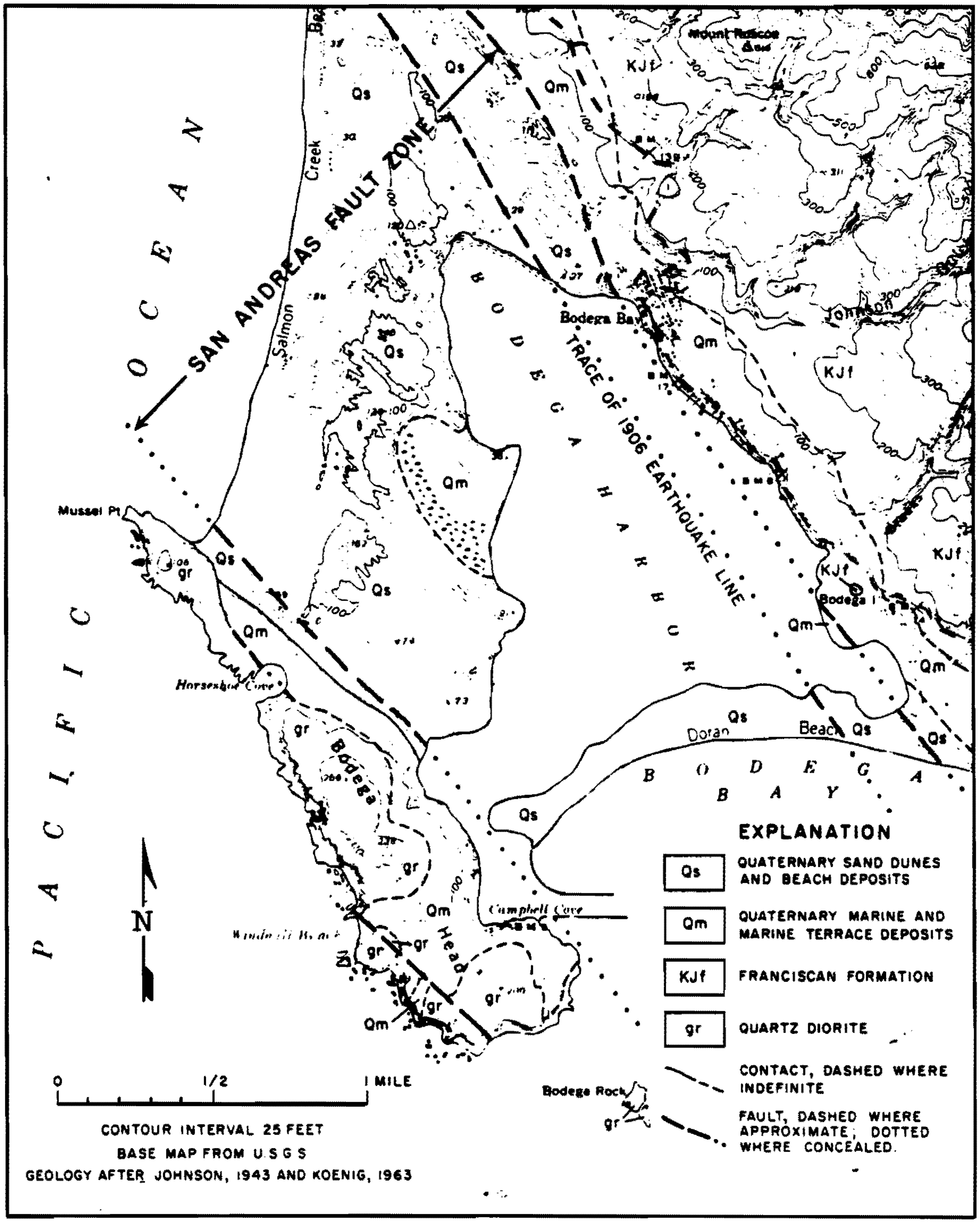
The minutes of the ACRS Subcommittee meeting held concerning Bodega Bay on March 20, 1963, report briefly on seismic matters, as one of seventeen items discussed at the meeting. It was reported that

> The plant is designed for motion in all directions. The same design criteria will be used at Bodega Bay as have been used at other (non-nuclear) plants which have satisfactorily withstood fairly severe earthquakes in this region.

In its report to the ACRS on Bodega Bay dated April 2, 1963, the Regulatory Staff approved the proposed design basis acceleration of 0.3 g subject to the qualification that no faults existed under the plant. Their total discussion of seismic matters was rather brief, as shown by the relevant excerpt from the staff report:

> The two outstanding geographical features of the site area are the San Andreas Fault zone immediately to the east, and the two rocky hills each over 200 feet high which make up Bodega Head. The reactor will be located between these hills approximately 1000 feet from the western limit of the fault zone. The 1.5 mile wide fault zone extends eastward from the site to the mainland, and has reduced the northern portion of Bodega Head to crushed rock and sand dunes due to extensive movement over the past few thousand years.
>
> The hills of Bodega Head consist of quartz-diorite rock covered by a shallow layer of sands and silts. The quartz-diorite formation is reported to be extensively fractured due to earthquake action, and evidence of old minor faults in the formation is reported.
>
> With respect to minor faults, the application states "no active faulting exists on Bodega Head and particularly under the power plant site." The applicant amplified this statement during a


SOME MAJOR FAULTS IN CALIFORNIA
BODEGA AREA
SAN
CALAVERAS FAULT
SAN FRANCISCO
ANDREAS
NACIMIENTO FAULT ZONE
GARLOCK FAULT
FAULT
ZONE
BODEGA HEAD
BODEGA BAY
TOMALES BAY
PACIFIC OCEAN
OLEMA
SAN ANDREAS FAULT ZONE
POINT REYES
BOLINAS
0 1 2 3 4 8
MILES
N
EXPLANATION
Q QUATERNARY SEDIMENTS
T TERTIARY SEDIMENTARY ROCKS
KJf FRANCISCAN FORMATION & RELATED ROCKS
gr GRANITIC ROCKS
CONTACT, APPROXIMATELY LOCATED
FAULT ZONE, DOTTED WHERE CONCEALED


PACIFIC OCEAN
SAN ANDREAS FAULT ZONE
BODEGA HARBOR
TRACE OF 1906 EARTHQUAKE LINE
BODEGA BAY
Bodega Bay
Mount Roscoe
Salmon Creek
Mussel Pt
Horseshoe Cove
Bodega Head
Campbell Cove
Doran Beach
Bodega Rock
Johnson
Qs
Qm
KJf
gr
N
0
1/2
1 MILE
CONTOUR INTERVAL 25 FEET
BASE MAP FROM U.S.G.S.
GEOLOGY AFTER JOHNSON, 1943 AND KOENIG, 1963
EXPLANATION
Qs QUATERNARY SAND DUNES AND BEACH DEPOSITS
Qm QUATERNARY MARINE AND MARINE TERRACE DEPOSITS
KJf FRANCISCAN FORMATION
gr QUARTZ DIORITE
CONTACT, DASHED WHERE INDEFINITE
FAULT, DASHED WHERE APPROXIMATE, DOTTED WHERE CONCEALED.

> meeting with the subcommittee and the staff on March 20th by stating that no faults have been located under the proposed location of the reactor power plant structures as a result of evaluations of several borings at the plant site. The applicant has described the geologic nature of the foundation materials (the fractured quartz-diorite) and has concluded that structures can be adequately anchored into this rock material and designed to withstand earthquake accelerations amounting to 0.3 of the acceleration of gravity (0.3G) unless a formation failure (faulting) occurs under the plant.
>
> The staff believes that the applicant is aware of the safety and design problems associated with earthquakes, and is proceeding with the development of an adequate design based upon the predicted earthquake loadings. It should be recognized that acceptance of this site would imply agreement that the likelihood of the occurrence of an earthquake which would cause faulting under the structures and, consequently, possible failure of the engineered safety components of the reactor facility is acceptably small. The applicant claims that slippage is more likely to occur in existing faults and that development of new faults would not be expected to occur.
>
> The staff does not know of any basis for disagreement with the applicant's assertions as to the existence or likelihood of development of faults. We believe, however, that further information should be developed on the existence of faults under the site through explorations made in the course of excavation. This information will be reviewed to determine whether or not the assumption that no such faults exist is factual; and if evidence of faults is found, the basis for approval will be re-examined.

Bodega Bay was reviewed by the full Committee at the 47th ACRS meeting, April 11-13, 1963. The minutes of the meeting indicate that while there was considerable discussion concerning seismic aspects of the proposed plant, this was not the principal focus of the ACRS review. Some portions of the meeting minutes related to seismic matters are excerpted below.

> Executive Session
>
> The plant will be designed to withstand an acceleration of as much as .3 that of gravity from an earthquake; this would be sufficient to resist the largest quake recorded in California. The containment valving arrangements needed to resist earthquake damage were discussed. Mr. Osborn said he had examined several plants following earthquakes; there was considerable damage to

the masonry and the concrete but very little to the piping.

Regulatory Staff (RS)

The representative of the Coast and Geodetic Survey said there is always some uncertainty in the distance from a fault. The center of the active San Andreas fault zone is about 1 1/2 miles away. The reactor will be within about a thousand feet of the edge of the San Andreas fault zone; the RS deems this to satisfy the 1300-feet requirement of the Site Criteria. No faults under the proposed plant site have been identified by drill holes; the search will continue during the excavation, the reactor location would have to be reassessed. Unfaulted gravels from the Pleistocene era indicate no seismic activity at the site for many thousands of years. Measuring the fault changes [sic] would be interesting scientifically, but the data would probably be of little value for this reactor design. The RS believes the reactor can be designed to withstand earthquake shocks.

Pacific Gas & Electric Co. (PG&E)

The power plant is to be on hard rock, a quartz diorite material, and no earth movement is expected. A consultant from California Institute of Technology confirmed the conclusions regarding earthquakes; these recommendations of experts are the design basis for the plant. Conventional PG&E power plants are designed to resist earthquake forces of .2 of the acceleration of gravity. A PG&E plant was subjected to an earthquake; the plant withstood the shock, but there was other local damage, e.g., tunnels gave away.

The possibility of tidal waves in the area was evaluated by the Corps of Engineers when a nearby jetty was designed to protect the harbor. The jetty height was set at 15 feet above sea level because of the maximum predicted wave height. There is no record of the jetty or the nearby sand bar being topped by a tidal wave. The base of the reactor plant will be 25 feet above sea level.

The accidents analyzed included loss of coolant accident by recirculation line rupture, fuel being dropped into the reactor, rupture of the main line outside the containment, and a rod falling from the reactor after it is critical.

The ACRS reported favorably on construction of Bodega Bay in a letter dated April 18, 1963 to AEC Chairman Seaborg. Much of the letter related to engineered safety features in a general fashion. The paragraph on seismic-related matters is reproduced below.

The requirements that are imposed on plant design because of location in an active seismic area have been considered by the applicant, and the referenced documents contain the recommendations of seismologists who have been consulted on this question. Tentative exploration indicates that the reactor and turbine buildings will not be located on an active fault line. The Committee believes that if this point is established, the design criteria for the plant are adequate from the standpoint of hazards associated with earthquakes. Careful examination of the quartz-diorite rock below should be made during building excavation, to confirm this point. Furthermore, the Committee suggests that, during design, careful attention should be given to the ability of emergency shutdown systems to operate properly during and subsequent to violent earth shocks, and to the stress effects that might be introduced because the reactor building and the turbine building are to be anchored in different geological formations. The need for earthquake-induced shutdown and isolation of the primary system can be considered at a later time.

With hindsight, both the ACRS review and that of the Regulatory Staff appear to have been relatively limited, in view of the nearness of the proposed site to the San Andreas fault. In effect, this site represented one where there was a high probability that the ability of the reactor to accomplish safe shutdown in the face of a major earthquake would be challenged during the lifetime of the plant.

Although PG&E projected schedule for approval of construction for Bodega Bay showed issuance of a construction permit by July, 1963, this schedule slipped. The foundation excavation for the Bodega Bay reactor containment proceeded, and in the process of examining the exposed surfaces, several small faults were found, one of which received the most attention and was labeled the Saft fault (named for the shaft which was dug for the reactor). Since the original review was predicated on the assumption that the reactor and turbine plant would not be located on an active fault, the entire matter was opened for re-review.

The minutes of the Executive Session of the ACRS at the 48th meeting, July 11-13, 1963 provide a status report:

> Bodega Bay Reactor
>
> Dr. Kouts commented on the geological developments. The completed U.S. Geological Survey report is not yet formally available; it is believed to support the conclusions of Pacific Gas & Electric (PG&E). Excavations show no recent faults; the latest was at least 35,000 years ago. The faults in the diorite granite rock are very ancient. Faults 10 miles to the south may be from the 1906 earthquake; there is still a question as to how this earthquake affected the Bodega Bay area. The Johnson fault, which has been mentioned by the intervenors, has not been found on the Bodega Bay head. The unconsolidated rock sediments are more firm than believed; consequently the turbine to be located on these materials should have about the same stability as the reactor located on the rigid rock.
>
> There were comments on the appropriate design measures for earthquake stresses; designing for forces of .45 of the acceleration gravity is now planned. The Regulatory Staff has been concerned over the quake effects on small components of the reactor system, e.g., switches in the controls, etc. Dr. Thompson suggested that the emergency power facilities be mounted on the same kind of base as the facilities served. It was noted that most of the damage in the San Francisco earthquake resulted from fires which went uncontrolled because of damage to the water lines.

The Regulatory Staff asked the U. S. Geological Survey to review the Shaft fault, attempt to evaluate its past history as a function of geologic time, and to provide guidance regarding its potential for future movement. They also asked the U. S. Coast and Geodetic Survey to advise them on seismological matters and on possible tsunami effects from off-shore or distant marine earthquakes. Additionally, they had Prof. Nathan M. Newmark and Dr. Robert A. Williamson of Holmes and Narver to advise them on seismic engineering design. The ACRS obtained the consulting services of a seismic engineer, Mr. Karl V. Steinbrugge.

Messrs. Julius Schlocker and M. G. Bonilla (1964) studied the matter of faulting on behalf of the Geological Survey and concluded that the Shaft fault had displaced sediments that are more than 42,000 but less than 400,000 years old; and that the last movement on the fault took place during the last 400,000 years. This report goes on to say that, because the possibility exists that the faulting occurred during the past few hundred years, it is prudent to predict that fulting is a possibility at the site during the next 50 to 200 years (which means that differential permanent displacement of the foundation might occur under the plant. In the report, Bonilla and Schlocker also discuss the past history of "sympathetic" faulting, or the creation of new faults, close to the San Andreas fault and concluded that this possibility should be considered for any site close to the San Andreas.

The report concludes that the possibility of a 2 to 3 foot offset at the site should not be ruled out.

Other interpretations of the available geological information also existed. At a meeting with the Regulatory Staff on January 30, 1964, Prof. Hugo Benioff, a consultant to PG&E, stated that auxiliary faults would slip only a matter of inches when the main San Andreas fault ruptured. Dr. Don Tocher, another PG&E consultant, estimated that differential motion on the Shaft fault would be less than an inch, given a major earthquake on the San Andreas. Benioff, Tocher and Mr. E. Marliave, another PG&E consultant, all concluded the Bodega Bay site was acceptable.

Also in the Bodega Bay file are letters from several other geologists to AEC Chairman Seaborg concerning the Shaft fault. Professor Clyde Wahrhaftig of the University of California, Berkeley, voiced the following opinion in a letter dated April 29, 1964:

> Thus the only geological evidence that we have to go on is that there have been several episodes of displacement on this fault, and that the last one occurred within the last 200,000 years. This is a totally inadequate sample on which to base an estimate of the probability of recurrence. However, keeping this in mind, the best <u>guess</u> one can make for the probability of a repetition in the next 200 years (the life of the reactor) is 200/200,000 or one in a thousand. The probability is not less than 1/50,000 and not more than 1/50.

Professor Wahrhaftig contrasted this with:

> A probability of 1 in 300 that thousands of people will be in the University of California Stadium the next time there is a strong earthquake on the Hayward Fault" [which would directly affect the stadium, and many other important structures, possibly including large dams].

Professor Wahrhaftig voiced a final opinion:

> As a citizen concerned with my own health and safety and with the welfare of the community in which I live, I can speak further. This average likelihood is greater than the risks we individually take each time we take an automobile drive or an airplane trip* (which, I, personally, consider hazardous enough). In my mind, it is high enough to justify requiring that the engineers design a reactor that cannot possibly release significant radioactivity to the atmosphere, in the event of a displacement of as much as 5 feet through the plant. (A displacement twice that which we know took place). If such design specifications cannot be met, then the site should be abandoned.

In another letter dated April 25, 1964, Professors Garniss H. Curtis and Jack F. Evernden of the University of California, Berkeley state:

> ...we believe that the total amount of displacement of any faults passing through the reactor shaft that has occurred since the Pleistocene sedimentary beds were deposited is of the order of one foot... The age of the sedimentary beds is certainly greater than 42,000 years and more likely between 200,000 and 600,000 years... The displacement in the sediments was probably the result of more than one displacement...
>
> ...If, as Schlocker and Bonilla believe and as we believe, the Shaft Fault has moved more than once during this time interval with displacements of, say, 4 inches each time, then the chances are about 1/500 for disruption of the site during the next 200 years... It seems worthwhile noting that since all controversy about the seismic hazards of the site localizes on the Shaft Fault and possible displacement along it, a simple solution would be to move the hole so that the Shaft Fault does not intersect it.

On March 27, 1964, the large intensity Alaskan earthquake occurred. In addition to very considerable damage in Alaska, particularly due to landslides, this earthquke produced an appreciable tsunami wave at Crescent City, California, providing further empirical evidence of this phenomenon. How much impact this earthquake had on the Bodega Bay review is difficult to assess. However, it seemed to have entered into the overall considerations, as shown from the ACRS summary minutes (reproduced on the following pages) of a meeting between the Regulatory Staff and the Applicant for Bodega Bay on April 14, 1964.

*Prof. Wahrhaftig did not quantify the risks he was comparing, or discuss other risks in Berkeley and throughout California due to large earthquakes, etc.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 24, 1964

MEMORANDUM

To : File (Bodega Bay)

From : R. F. Fraley, Executive Secretary
ACRS

Subject: DL&R MEETING ON APRIL 14, 1964 RE BODEGA BAY ATOMIC PARK, UNIT NO. 1

This meeting was set up by DL&R to discuss seismic design criteria, especially Amendments 6 and 7, for the Bodega Bay Plant with their consultants. ACRS members were invited to participate.

Attendees:

ACRS
D. A. Rogers
K. R. Osborn
H. J. C. Kouts (morning only)
R. F. Fraley (Staff)

DL&R Consultants
N. M. Newmark
R. A. Williamson
L. Murphy

DL&R
J. Newell
E. G. Case
R. Bryan
N. Watson
C. K. Beck
M. M. Mann
L. Murphy

Dr. Newmark expressed concern about the damping factors and maximum ground acceleration proposed by PG&E, especially at the high frequency end (below 0.2 sec period) of the design spectrum. He noted that it is difficult to predict frequency accurately in this range and since damping factors are frequency and stress dependent a conservative design curve should be used. He suggested a maximum ground acceleration value of 1.0 g vs the 0.66 g proposed for high frequency components and no attenuation of maximum component acceleration (Sa) for lightly damped components as the high frequency end of the spectrum is approached (Refer to Figure 100545 of Amendment 6).

He noted that this was the same position that he had previously suggested in meetings with PG&E. Dr. Mann inquired as to how one designs a system with interconnected components of different natural frequencies. Dr. Newmark explained that it is common practice to use the "worst component" (max. acceleration) and design the whole system to this value. Differential movement between components must, of course, be taken into account.

Mr. Murphy stated that the Coast & Geodetic Survey had reached about the same conclusion regarding maximum ground acceleration. Before the recent Alaskan disturbance a value of 0.9 g looked reasonable to C&GS, but they will probably revise this upward to 1.0 g as a result of the Alaskan quake. He noted, however, that a final conclusion is not likely for several weeks since the C&CS wants to review the Alaskan data in more detail before transmitting a recommendation to the AEC. Some difficulty may be experienced, however, in correlating the two areas since low frequency waves did most of the damage in Alaska because of the alluvium foundation material while high frequency damage would be expected at Bodega Bay because of the rock foundation.

Dr. Beck summarized the Bodega Bay study which indicates that a 2-inch differential fault movement could be accommodated with no damage to the containment and a 2-foot differential movement could be accommodated with some damage but no impairment of the containment function. Dr. Newmark seemed inclined to agree that a differential movement of one or two feet could be accommodated by the containment in a design similar to the one proposed. He did suggest, however, that some design changes might be required, for example, use of vermiculite concrete rather than sand as the pad for the containment. Apparently, this material has been used in bomb tests to protect buried structures from differential earth movements up to one foot. He did caution, however, that the probability of satisfactory component operation might be seriously reduced by this degree of movement. Mr. Williamson seemed less inclined to agree that the containment and the associated systems and components could be designed with any certainty that they would be able to withstand a differential movement of two feet. He questioned specifically the ability of isolation valves to close after lines were twisted or bent by movement of the containment through a two-foot displacement. He also noted that the decay heat removal system is not self contained and requires connection to the power source, pumps, etc. which are not integral with the containment. It was generally concluded that this area would require a careful engineering evaluation.

Mr. Osborn inquired if the Alaska quake would result in the revision of any existing building codes or criteria used on the West Coast. Dr. Newmark stated that codes specify minimum requirements and that a more elaborate dynamic analysis is usually made for unusual structures (e.g. tall buildings). A more vigorous earthquake than the nominal value is also considered. He did not anticipate, therefore, that any code requirements would be changed. J. Newell noted that existing building codes are based on prevention of significant damage to structures whereas the AEC requires no damage to structures at least as far as containment integrity is concerned.

Mr. Case requested that Dr. Newmark and Mr. Williamson submit their conclusions concerning amendments 6 and 7 as soon as possible.

* * * * * *

The ACRS met with the Regulatory Staff and Pacific Gas & Electric concerning Bodega Bay at the 53rd meeting, February 13-15, 1964. The minutes of the Executive Session indicate sentiment among some Committee members to the effect that large displacement on the Shaft Fault was very unlikely. Also, as indicated in the excerpt reproduced below, the top management on the Regulatory Staff was having difficulty resolving the wide divergence in expert opinion.

> Later, Mr. Price, Mr. Case, Mr. Lowenstein and Dr. Mann joined an Executive session of the Committee and reported their dilemma over the apparent conflict in testimony from Mr. Schlocker and Mr. Bonilla of the USGS and that of Dr. Benioff and Mr. Marliave, PG&E consultants. Further definition of the problem and more data seems needed. Mr. Price noted that Mr. Anderson and Mr. Schlocker, both of the USGS, had somewhat different opinions regarding the possibility of quakes at this site. Later Mr. Price complained about the Department of Interior trying to hold a hearing on this reactor case in the newspapers. Recently, the Secretary of the Interior requested all the reports on geology be channeled through his office.
>
> Mr. Price said that other reactors, e.g., those at Los Angeles and at Haddam, Connecticut are involved; preliminary USGS reports on these had conclusions indicating prejudice of the Department of the Interior. Some of the comments could be taken as establishing site criteria. The early information on the USGS Connecticut Yankee report is that a fault at the Connecticut River near the plant site may make Haddam as serious a seismology risk as is Bodega Bay; little was made of the fact that the fault is unlikely to move as indicated by the long stability of the New England region. Recent discussions are hoped to clarify the situation so that the AEC receives solely technical advice from the USGS. The Commission now supports the USGS and the USC&GS for geological consultation; this includes work at a number of Commission sites. The USGS has been asked by the RS for advice on geology only, but of late the Secretary's office has included seismic information.

It should be noted that at a Special Meeting the same month (February 24, 1964), the ACRS wrote a letter report to AEC Chairman Seaborg suggesting that over conservatism was being employed for seismic design in zones of relative seismic quiet in the eastern United States. We shall return to this point later.

On April 30, 1964 the Regulatory Staff submitted a report on Bodega Bay to the ACRS for consideration at its 55th meeting, May 6-8, 1964. Several excerpts from this report follow below.

Vibrational Effects

PG&E has proposed to use a ground response spectrum derived from the El Centro earthquake of 1940 for seismic design of critical structures and components. A value of maximum ground acceleration of 0.33 g would be used for the design of these structures and systems at normal working stress. PG&E has also proposed to design all structures and components important to safe shutdown of the facility so that yield stress would not be exceeded or their functioning would not be impaired due to a maximum ground motion of twice this value (0.66 g). Where appropriate, dynamic analyses of individual systems or components would be performed to assure that the design would satisfy these criteria.

Based on advice from our consultants, we believe that the method of design and the procedures proposed by PG&E would be suitable, and that the damping factors are acceptable although not particularly conservative. However, in our opinion, an earthquake occurring at or near Bodega Head with a magnitude equal to that of the 1906 earthquake could result in a maximum ground acceleration in basement rock of about 1.0 g at the site. Thus, the value of maximum ground acceleration proposed as a design criterion by PG&E is approximately 40% too low. Nevertheless, it appears that the potential effects of ground vibration could be adequately handled by well understood design measures which could be modified to accomodate any ground acceleration forces that might be deemed necessary. This is a matter of structural engineering design practice widely utilized in California, and we are satisfied that the plant and all important components could be adequately designed and build to take care of the vibrational effects of a major earthquake on the San Andreas fault.

Effects of Tsunamis

The PG&E application does not discuss the probability, magnitude or potential consequences of seismically induced tidal waves which might follow an earthquake of 1906 magnitude somewhere along the San Andreas fault. The Coast and Geodetic Survey has informed us that a serious tsunami might result from landsliding caused by an earthquake of magnitude 7.0 or greater on the Richter scale at locations within about 10 miles of the coast, or from an earthquake of this magnitude on the underwater portion of the San Andreas fault north of the site.

The Coast and Geodetic Survey has not completed its evaluation of the potential seriousness of this problem. Based on present information, it is not clear to the staff that we could rule out the possibility that the reactor facility would not be damaged or at least inundated by a seismically induced tidal wave. An investigation of the off-shore contours in the vicinity of Bodega Head would be necessary to resolve this matter. An analysis of the potential effects on the reactor facility of a tsunami would also be required. Despite the present lack of information, it seems likely that even if a tsunami were to sweep across Bodega Head design measures could be taken in order to prevent loss of containment integrity from such an occurrence. However, it is not clear at this time that adequate protection could be provided for cooling and power lines exterior to the containment system. Our calculations indicate that access to a heat sink within about 12 hours is necessary in order to prevent over-pressurization of the containment system due to release of decay heat.

Effects of Differential Ground Motion

The Company estimates that it [the Shaft fault] is over 42,000 years old and perhaps up to 400,000 years old. The geological support for this position is impressive. On the other hand, since geologists from the Geological Survey believe that the absence of traces of the Shaft fault in sediments younger than 42,000 years could also be due to its dying out upward and laterally, the fault in the sediments may have occurred more recently than 42,000 years ago, but not earlier than a few centuries ago. Regardless of its age, the existence of the Shaft fault does show that ruptures did occur in the basement rock on Bodega Head outside of the main San Andreas fault zone. The Geological Survey has stated that the Shaft fault in the sediments probably had a total displacement in a single movement of between 1 and 3 feet.

Because surface rupturing by tectonic faulting occurred outside the main San Andreas fault zone in 1906 on terrain geologically similar to that of the Bodega site, the geologists from the Geological Survey believe that the possibility should not be ruled out that there might also be a sympathetic movement of several (2-3) feet on Bodega Head in the event of a large earthquake on the San Andreas fault. They say that the possibility of such movement is low, but not low enough to be ignored in the design of this facility. The basis for this opinion is principally derived from observations of surface faulting outside the San Andreas fault zone which occurred in 1906.

The question then becomes, can a reactor facility of this power level be designed and built which would safely accommodate the differential ground movements of the magnitude that could be expected. The company has proposed a design concept which would accommodate some relative displacement. They plan to fill the annular space between the sides and bottom of the containment structure and adjacent rock with a layer of frangible material of a type yet to be selected. Theoretically, displacements of several feet might be accommodated by such a design concept. Tests have been conducted for the Defense Department in which polyurethane foam was successfully used to accommodate relative displacements of experimental structures of this magnitude. However, such an engineering safeguard has not been used in practice, nor has it been proven for structures approaching the size and complexity of the Bodega reactor facility with its complicated and sensitive components.

One of the more serious considerations involved in the use of frangible materials to protect against earthquake-induced displacements would be the vibrational effects on the structure from aftershocks which would be likely to follow the major earthquake. The deformations of these materials in the yield range are irreversible. Consequently, oscillations or displacements of the containment structure causing lateral pressures sufficiently intense to compress the material would leave an annular void. In such a situation the facility would then be vulnerable to additional damage from vibrational effects of the aftershocks. At the very least, it would appear that a large scale research and development program would be necessary in order to determine the feasibility of designing against large displacements. Since we must assume the possibility of large differential ground movement, we believe that we should not depend upon the integrity and reliability of unproven engineered safeguards to protect against such movements and the aftershocks that might ensue.

CONCLUSIONS

Although the design criteria proposed by PG&E are not in agreement with criteria suggested by our consultants to protect against vibrational effects and possibly the effects of tsunamis resulting from a large earthquake occurring on the San Andreas fault at or near Bodega Head, it appears that these problems might be adequately resolved by changes in design criteria and more complete analyses. However, despite the fact that the risk could be effectively eliminated by moving the plant to a location a couple of miles distant from a main fault zone. On this basis, we have concluded that the site proposed by PG&E is not suitable for a reactor of the general type and power level proposed.

The conclusions expressed in this report are subject to reconsideration in light of any new information which may be received at the ACRS meeting, and the recommendations of the ACRS.

Thus, on April 30, 1964 the Regulatory Staff conclusion was that the Bodega Bay site was not suitable, saying despite the fact that the risk of large differential ground movement on Bodega Bay is low, we do not believe that unproven design measures should be depended upon to solve this problem.

In passing, it is noted that, in the section on tsunamis, the staff was not yet aware of the "China Syndrome" matter, and wrote that a twelve hour delay was acceptable in providing a sink for decay heat.

The minutes of the 55th ACRS meeting in May, 1964, record large differences in opinion among experts concerning the probability of displacement along the Shaft fault in the past and in the future. The Regulatory Staff was inclined to consider it impossible to guarantee engineering safeguards for earth movements of two to three feet. PG&E consultant, Housner, stated that although no structure such as the one now planned had been built to withstand large earth movements, he believed that this design was satisfactory for significant earth movement. He went on to discuss how the design could accommodate one foot of relative displacement, and Mr. Whelchel of PG&E said it might be possible to design for three feet. PG&E noted that no steam line pipes had failed in the recent Alaskan earthquake. And Professor Housner described how much larger the vibratory motion requirements for Bodega Bay would be, compared to the San Francisco Building code.

Dr. Benioff said there is no evidence of any acceleration as high as 1.0 g earthquake, and cited how only 18% g was recorded on rock by the Japanese for a magnitude 7.7 quake.

The meeting minutes conclude as follows:

> A summary late in the meeting indicated: the chances of earthquake shear at the site are low; with proper design, displacement up to two feet would cause no building damage, a displacement of five feet would break the primary system and might violate containment; finally, the engineered safeguards, e.g., core quench sprays, must be available.
>
> Although a letter on this case was drafted, it was not dispatched because of the wishes of the Regulatory Staff.

A copy of this letter, which was never sent, is reproduced below:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

FINAL DRAFT
DAR:vb
5-9-64

SUBJECT: REPORT ON BODEGA BAY ATOMIC PARK -- UNIT NO. 1

Dear Dr. Seaborg:

At its fifty-fifth meeting on May 7-9, 1964 at Argonne National Laboratory, the Advisory Committee on Reactor Safeguards again considered the proposal of Pacific Gas & Electric Company to build and operate a 1008 MW(t) boiling water reactor on Bodega Head north of San Francisco, California. The Committee had the benefit of oral discussions with the applicant and its consultants, with the AEC Staff and its consultants, including staff of the U.S. Geological Survey; reports cited below; attendance at Regulatory Staff meetings with the applicant and its consultants; and interim discussions with the staff and its consultants including staff of the U. S. Coast Guard and Geodetic Survey.

This proposal was considered at the Committee's 47th meeting, reported in its letter of April 18, 1963 which stated

> "Tentative exploration indicates that the reactor and turbine buildings will not be located on an active fault line. The Committee believes that if this point is established, the design criteria for the plant are adequate from the standpoint of hazards associated with earthquakes. Careful examination of the quartz-diorite rock below should be made during building excavation, to confirm this point. Furthermore, the Committee suggests that, during design, careful attention should be given to the ability of emergency shutdown systems to operate properly during and subsequent to.

> violent earth shocks, and to the stress effects that might be introduced because the reactor building and the turbine building are to be anchored in different geological formations. The need for earthquake-induced shutdown and isolation of the primary system can be considered at a later time."

The exploration suggested in the above comment has been done, and the geologic features discovered have made necessary a further consideration of the design. These features include fractures in the underlying rock and sediment.

The Committee has been advised that during the life of the proposed reactor there is a high probability that it will experience at least one major earth shock. There is associated with such an earthquake a remote probability that the plant will be subjected to the effect of a shearing motion in the rock on which it is built. The Committee is of the opinion that designs of the nature proposed can be made to withstand the effects of the anticipated earthquakes such that at the worst the reactor can be shutdown and cooled without undue release of fission products.

Result of investigation of the probable magnitude of tsunamis at the site has not been received. This investigation may show the need of added protection of plant and emergency coolant system.

The Committee recognizes the presence of earthquake hazards at this site and believes that special measures will be necessary. The Committee would like to be kept informed regularly as to the developments in areas closely related to the safety considerations arising from these hazards. Among others these should include:

(1) Provisions to accommodate possible earth movements and effects of displacement along the fault. The applicant has proposed orally to design the building to withstand up to two feet displacement along the discovered fault. The Committee believes that the engineering principles are sound and, if extended to take into account the possibility of the same motion in any direction, will afford the degree of assurance required for protection of the reactor.

(2) Consideration of model testing or other experimental verification of novel design features associated with earthquake protection.

(3) Redundant provisions to assure emergency cooling water in case of damage to normal and emergency supply systems by earthquakes or tsunamies.

(4) Design and tests of critical plant components such as instrumentation and control rod operating mechanisms to withstand earthquake damage.

(5) Further clarification in specific areas of seismic design of components. Although the criteria proposed by the applicant appear to be generally acceptable, amplification of some areas is needed. Included among these are criteria regarding:

a. Design amplitudes to be used for high frequency oscillations.

b. Design basis to be used where the response of the building itself is a factor in modifying the response of equipment items within.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 13, 1964

MEMORANDUM

TO : H. L. PRICE, DIRECTOR OF REGULATION

FROM : DR. H. J. C. KOUTS, CHAIRMAN - ACRS HK

SUBJECT: BODEGA BAY ATOMIC PARK, UNIT NO. 1

At the ACRS Meeting on June 12-13, 1964, we should like to consider in detail the following questions, on which the statements that have been presented to us are still not clear:

1. We wish to be sure we understand the specific methods to be used to analyze the ability of structures to withstand earthquake oscillations. Some new features of the analysis were introduced at our 55th meeting.

2. Will the applicant modify the frequency spectrum used (El Centro, 1936) to account for the rock foundation of the Bodega reactor structure?

3. What measures are proposed to assure that the reactor can be maintained safely in a shutdown condition indefinitely if all vital lines to the reactor building are severed?

4. What is the degree of damage to the reactor building and the reactor to be expected from shear displacement along any line crossing the reactor building shaft? This analysis should not assume a size of displacement. What is desired is damage as a function of displacement. What displacement leads to fracture of the concrete structure? What displacement would rupture the containment? What displacement would lead to rupture of the primary reactor system? It is vital that these judgements be based on features of the system as it is to be built, and not be supported only in general terms. The effects of both shear and tensile strains should be considered.

5. What measures will be taken to protect against tsunamis greater in size than the breakwater at Bodega Bay would suppress?

We should also like to be able to discuss with your consultants these questions and the answers of the applicant, as was the practice in previous meetings. But to avoid having the pressure of time contribute to any uncertainty as to positions taken or their meanings, we are planning a two-day meeting to be devoted entirely to the Bodega Bay reactor.

The letter would have been favorable toward construction of the Bodega Bay plant, with provision in the design for surface displacement.

Following the 56th meeting, the ACRS dispatched the following memorandum to Harold Price, the Director of Regulation.

On May 19, 1964, the Regulatory Staff forwarded a set of questions to PG&E which included the ACRS questions. And a meeting was held by the Regulatory Staff with the ACRS Subcommittee on May 22, 1964 to review the matter. Excerpts from the minutes of this meeting follow.

> Mr. Price opened the meeting by noting that it is especially important in this case, because of the opposition of intervenors, that he have a clear understanding of the position of the various parties involved and that all understand and agree on a set of design requirements which can be defended at the hearing.
>
> Mr. Case reported orally on the position that the U.S. Geological Survey will maintain, namely that : (1) a total fault differential movement up to 3 feet in any direction should be considered in design of this plant. The possibility of movements greater than 3 feet is considered remote enough so that it can be neglected. Messrs. Case and Price reported further that the U.S. Coast and Geodetic Survey will maintain that (1) a 2.5 ft. differential movement should be considered at this site, (2) a shaking with a ground acceleration of 0.66 g and peaks up to 1 g max. in the frequency range of 0.3 to 0.7 sec. is likely (DRL therefore feels that a 1 g acceleration should be used in the design of critical components), and (3) a 25 ft. high seismic sea wave should be considered. A wave this size is apparently considered comparable in conservatism to an earthquake with an 0.66 g max. acceleration. A preliminary report to this effect is being revised for editorial changes and should be available before June 12, 1964.
>
> Mr. Osborn pointed out that the tsunami situation at San Onofre appears to be much worse than Bodega Bay and will have to be considered for San Onofre as well. Mr. Osborn questioned the conclusions of the U.S.G.S. and U.S.C.&G.S. regarding differention movement when compared to reputable PG&E consultants who maintain that only a fraction of an inch is likely. Mr. Price maintained that he would like to discuss the engineering aspects of design based on a 3 ft. movement, however, to determine if it is feasible to design a plant to this criteria with reasonable assurance of its reliability.
>
> Dr. Newmark eventually concluded that major modifications are required to withstand a 3 ft. movement with little plant damage. The major change would probably require enlarging the excavation to provide for building oscillation and differential fault movement. If crushable materials are to be used to fill the annular space around the containment, a gap of from 4-8' is considered reasonable depending on the credit to be given for symmetry of fault/plant slippage. If a crushable material is not to be used 1.5 - 3 ft. gap is considered necessary. Dr. Newmark noted

We further believe that the applicant should be made aware that shear displacements of from one to two feet are considered credible by some, and that the primary question we now face concerns the ability of the plant to withstand such displacements without undue hazard to the health and safety of the public.

The proximity of the San Andreas fault is not, in itself, an adequate reason for prohibiting the construction of the proposed Bodega Head facility.

The provisions for 3-foot radial clearance around the walls of the containment structure is satisfactory, in the writer's opinion, for any credible fault displacement beneath the structure.

The sand base beneath the containment structure probably can be designed to act as a horizontal shear limiting mechanism within reasonable limits. The effect of vertical accelerations make this type of foundation somewhat unreliable with respect to exact values of the horizontal force at incipient sliding. The horizontal displacement of the bedrock beneath the containment structure at the time of a design earthquake with faulting on the San Andreas fault complicates the problem by tending to shift the containment structure on its sand base. If the Dames and Moore study is successful, then ACRS can conclude that the sand base foundation will be acceptable.

The containment ring for the sand base is necessary, and probably should be wider than presently planned. This is neither a design nor cost factor.

Structural isolation of the plant structures as proposed by the Pacific Gas and Electric Company, is adequate when used with judgment which is experienced in earthquake engineering.

It is recommended that the low end of the spectrum be increased from 0.33 G to a higher value, possibly not less than 0.50 G for all ductile or ductile acting members and structures. It must be stressed that this is a judgment decision based on extensive experience plus some instrumental data.

It would be reasonable to request that the detailed earthquake design be left in the hands of a firm specializing in earthquake engineering.

The foregoing conclusions and recommendations have been directed towards specific problems at Bodega Head. The findings are believed to be conservative, and are consistent with the present knowledge regarding the state of the art of earthquake engineering.

On October 6, 1964, the Regulatory Staff issued a report to the ACRS concerning seismic considerations for Bodega Bay. The Report is reproduced below.

> that use of a crushable material would require consideration of its effect on building oscillations during and after the major shock. If crushable material is used its support would be changed for aftershock considerations and this point has not been discussed by PG&E. If the gap is left empty however Dr. Newmark expressed concern over the possible impingement of rocks locally on the containment and the resultant high localized stresses. Tapering of the excavation to prevent impingement was discussed as a possible solution to this second problem. Mr. Williamson noted that practically no information has been provided concerning the frangible material proposed even though this is a very important item. The use of sand as a sliding foundation pad was also questioned since it acts about like a solid under high compressive loadings.
>
> Mr. Price asked if a building can be designed with as much reliability to withstand a 3 ft. movement as to withstand a smaller movement. After some discussion it appeared that known factors could be adequately taken into account by conventional design and construction techniques, however, some unknown factors may become more significant for movements of large magnitude.
>
> Dr. Newmark also expressed concern over thelack of sophistication in the single degree of freedom, shock spectrum technique proposed for design of the plant to resist shaking. The use of a static analysis rather than a dynamic study for plant/fault slippage is also considered very elementary. He noted that more elaborate computer techniques have been developed and used for buildings to analyze 3 degree of freedom systems for different input pulses.

At a Regulatory Staff-arranged meeting on June 17, 1964, Mr. Whelchel of PG&E agreed to design the plant for a movement of three feet, including the necessary engineered safeguards. At the same meeting PG&E consultant Housner stated that he would recommend against construction at this site if he felt that a 3 ft. differential movement were a possibility. "Mr. Price pursued his reasons for this position which are apparently economics and reputation rather than concern over the ability to construct a safe plant for these extreme conditions."

On August 17, 1964, the ACRS received a written report from its consultant Karl Steinbrugge. His conclusions were as follows:

> <u>Summary, Conclusions, and Recommendations</u>

BODEGA HEAD NUCLEAR POWER PLANT

SEISMIC CONSIDERATIONS

A 325 Mwe Boiling Water Nuclear Power Plant with pressure absorption containment is proposed for construction by the Pacific Gas and Electric Company on Bodega Head sixty miles northwest of San Francisco. The plant site is located roughly 1000 feet west of the western edge of the San Andreas fault zone, a band of frequent earthquake activity running generally north and south along most of the State of California. The choice of this particular location has necessitated prolonged and intensive study of factors affecting the safety of the installation in the event of the occurrence of a severe earthquake at or near the location of the plant.

Since the field of earthquake structural design is highly specialized, it has been necessary to call upon the services of reputable expert consultants for help in analyzing the various problems involved, and to rely heavily on their advice in arriving at a decision on the technical feasibility of building the Bodega plant at the proposed location with reasonable assurance that it will safely withstand the maximum earthquake that might credibly occur during the life of the plant. The design consultants employed by the applicant, PG&E, include Dr. George W. Housner, Professor of Civil Engineering and Applied Mechanics at the California Institute of Technology, Dr. Hugo Benioff, Engineering Seismologist also of Caltech, and Mr. E. C. Marliave, Consulting Geologist. The AEC Regulatory Staff has retained the services of Dr. N. M. Newmark, Professor of Civil Engineering at the University of Illinois and Mr. Robert A. Williamson of Holmes & Narver.

OCT 6 1964

ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ON

BODEGA HEAD NUCLEAR POWER PLANT ... SEISMIC CONSIDERATIONS

Note by Director, Division of Reactor Licensing

The attached report has been prepared by the staff of the Division of Reactor Licensing for consideration by the Advisory Committee on Reactor Safeguards at its October 1964 meeting.

maintained in a safe condition in the event of the occurrence of an earthquake of the severity postulated by the USC&GS and USGS. The technical basis for their conclusion was also requested.

While continuing to disagree strongly with the credibility of such an extreme earthquake, the applicant has nevertheless proposed a design which the company and its consultants feel confident will safely ride through a 2/3g earthquake, with peak ground acceleration up to 1.0g, which is accompanied by differential shear ground displacement under the reactor containment of up to 3 feet either horizontal or vertical. If such an earthquake should in fact occur, the containment might be tipped or rotated slightly, but there would be no breach in its leak-tightness and no release of fission products, in the opinion of the applicant.

The general description of the postulated earthquake involves a pattern of ground motions similar to that recorded by the Coast and Geodetic Survey in the El Centro Earthquake of May 18, 1940, but with approximately twice the intensity, corresponding to a maximum acceleration of two-thirds gravity, a maximum velocity of 2.5 ft/sec, and a maximum ground displacement of 3 feet, and with occasional intermittent pulses of acceleration up to 1.0g. The structures are considered to be subjected to simultaneous shear displacements ranging up to 3 feet, along lines extending under the containment structure or other parts of the plant, with motions in either horizontal or vertical directions along the fault. It is also assumed that aftershocks of intensity equal to the El Centro quake might be suffered before remedial action could be taken.

There is a substantial difference between the viewpoint of the applicant and that of the U. S. Coast and Geodetic Survey and U. S. Geological Survey with respect to the maximum credible earthquake that should be taken as the design basis for the Bodega Head plant. The PG&E earthquake consultants feel strongly that the maximum ground acceleration to be expected during any credible earthquake at or near the plant site is 0.33g, and they consider it incredible that there should ever be more than a few inches of differential ground motion under the site. The USC&GS, on the other hand, has recommended that the reactor and its containment structure be designed to withstand a ground response spectrum of 2/3g, with peak accelerations up to 1.0g together with possible differential shear ground motion of up to 2-1/2 feet, while the USGS goes even further in recommending consideration of shear displacement of up to 3 feet. There is also a wide difference of opinion respecting the size of the tsunamis that may be expected to result from offshore earthquakes. Consultants to the applicant are firm in their opinion, based on all available records along the West Coast, that no tsunami will ever push water more than 15 feet above mean water level at the plant site. However, the USC&GS has recommended that protection against 50-foot tsunamis be provided in the design of the plant.

The applicant was made aware of the recommendations of the USC&GS and USGS almost a year ago, and has been asked a number of questions designed to determine whether the applicant considered it feasible to design the Bodega plant so as to provide reasonable assurance that the integrity of the reactor containment would be preserved and that the reactor would be shut down and

-73 feet to yard elevation at +25 feet. The walls of the reactor containment pit will be lined with reinforced concrete to prevent possible spalling of material into the pit. The annular space will be permitted to fill with water. The reactor containment structure will be founded on a layer of carefully selected sand of known characteristics which will permit horizontal movements up to 3 feet without impairing the function of the containment structure, although the structure might be shifted or rotated. Differential vertical motions up to 3 feet may cause the containment structure to tilt or shift, but, in the opinion of the applicant, in no case will the containment function be impaired.

The plant will be designed with no rigid structural interconnection between any major components. The reactor containment structure will be structurally independent of the turbine generator foundation, the plant control building, the radwaste facility, the stack, and the plant service buildings. Piping and wiring interconnections important to safety between the reactor containment structure, the control building and the turbine generator will have sufficient flexibility to accommodate 3 feet of relative movement. In order to prevent overstress at points of penetration for piping connecting the dry well with the turbine, the company proposes to provide adequate anchors and bracing adjacent to the containment shell and beyond the double isolation valves. These anchors will be adequate to withstand all piping loads due to differential motion in any direction up to 3 feet between the reactor containment structure and the turbine generator foundation.

There are two major problems posed by the postulated earthquake. The most unusual one is that of providing for a shear ground displacement of as much as three feet underneath the reactor building. The other is that of vibrational stresses.

Although there is a substantial design effort involved in computing the vibrational stresses, and judgment has to be exercised as to the proper vibrational spectrum and structural damping factors to use in assuring that the reactor containment structure and all the equipment inside it will safely withstand the vibrational aspects of the earthquake, the technology is well understood. The critical area here is the ability of vital structural components to withstand the stresses put on them by the simultaneous occurrence of the maximum postulated accident (rupture of reactor coolant system) and maximum postulated earthquake. Under these extreme conditions the question focuses on the maximum allowable stresses that should be used in the design computation relative to the yield stress of the various materials under consideration. While many of these details have not yet been resolved, there appears to be no reason to believe that anything of a fundamental nature will arise that cannot be successfully handled.

Building the reactor structure and its foundation in such a way that it will safely survive a shear ground movement underneath it of as much as 3 feet poses a more troublesome problem. The applicant proposes to accomplish this by a design which provides for a 3 foot unobstructed radial clearance between the outside of the reinforced concrete containment structure and the inside of a containment pit, completely around the circumference, from elevation in the difficult position of having to arrive at a technical evaluation of the public safety risks in an area where there is little or no experience background. Other reactor installations have presented seismic design problems, but not in the extreme form presented by Bodega Head under the earthquake postulated by the Coast and Geodetic Survey and the Geological Survey. One problem which the Bodega plant has in common with all other nuclear power plants subject to special seismic design considerations is the inability to conduct any sort of performance test on the finished structure that will demonstrate that the design objectives have been achieved. The uncertainty presented by this situation has been accepted as a reasonable risk at all of the other nuclear reactor installtions meeting specified seismic design criteria. There would be no difficulty in applying the same philosophy at Bodega Head were it not for the extreme earthquake postulated by the Coast and Geodetic Survey and Geological Survey.

Even so, we believe with our consultants that, with proper attention to several specific items still in the discussion stage, the earthquake design proposed by the applicant, Pacific Gas and Electric Company, is technically feasible for earthquakes up to the magnitude postulated by the USC&GS and USGS. Although the likelihood of earthquakes is high, the seismic design of the plant is commensurate with the proposed dimensions of those possible earthquakes, so that the probability that damage to the plant of sufficient magnitude to cause fission product release appears to be quite low. Even if the plant were to be severely damaged there are many safeguard systems of different types that also would have to fail before any damage to the public would result.

Although the foregoing proposal for safeguarding the Bodega Reactor and its containment structure against the postulated differential ground motion embodies concepts which are in many respects novel and for which little or no precedent exists, the Regulatory Staff Consultant, Dr. N. M. Newmark, has come to the conclusion after carefully studying the basis of the proposal that the structural integrity and leak-tightness of the containment building can be maintained under the conditions postulated. He points out, however, several items that will have to be given special attention during the design phase in order to achieve the desired objective. Perhaps the most important of these items is the main piping system leading from the reactor pressure vessel to the turbine and other equipment outside the containment building. The piping would have to be made sufficiently flexible to accommodate a relative movement of 3 feet without failure, and at the same time be damped to reduce its dynamic response to earthquake oscillations. Adequate provisions would also have to be made to have enough emergency power available locally to operate the emergency cooling system and other engineered safeguards in the event of earthquake damage to overhead power lines from outside power sources. Protection of the plant against the possible occurrence of large tsunamis has not yet been satisfactorily resolved but does not appear to offer any unsurmountable design barriers.

Both Dr. Housner, who prepared the design proposal for safeguarding the Bodega Reactor Installation against the postulated earthquake, and Dr. Newmark who has reviewed and concurred in it, have excellent professional reputations in the field of earthquake structural design. The Atomic Energy Commission is expected to consider and answer within its own top management echelon. The fact that the company has not proposed an alternate location despite the vigorous opposition which the proposed location has generated in certain sectors of the public, may mean that there is no suitable alternate. If this is the case, the question of justification still remains but might conceivably be answered in the affirmative on the basis that: (1) suitable locations for nuclear power plants in California are quite limited; (2) there is a rapidly increasing demand for electric power from "smog-free" sources in this most populous state in the union; and (3) the Bodega Head site has fairly good isolation and is otherwise satisfactory for the proposed plant.

Since the decision on whether or not to grant a construction permit involves both technical and policy considerations, both aspects of the problem will have to be given careful consideration in arriving at recommendations pertinent to the decision.

There are, nevertheless, policy considerations as well as technical considerations which must be weighed in arriving at a decision as to whether or not to grant a construction permit for the proposed reactor. The fact that the proposed site is adjacent to the San Andreas fault zone makes it almost certain that it will be subjected to one or more severe seismic disturbances during the lifetime of the plant. While there is a high probability that the plant under the proposed design can survive even a very large earthquake without damage, there is no way of being certain that this will be so. It is possible to have all seismic design criteria, computations and structural procedures checked by competent outside experts, but this will not necessarily prevent design and construction errors from creeping in unnoticed by anyone. Nor will it provide absolute assurance that all important parameters have been taken into consideration in the seismic design of the plant. These kinds of uncertainties are present in the protective systems of other reactors, but would probably exist to a higher degree in this plant.

The question then arises as to whether the public benefits to be gained from operation of the Bodega Nuclear Power Plant are high enough to justify building the reactor in close proximity to an active fault zone, which appears to involve somewhat greater problems than for other reactor locations. Stated differently, the question is, "Should a reactor be located where there is a relatively high probability of its being subjected to severe earthquake stresses, even though designs are provided which, in the opinion of experts, are adequate to counteract those stresses, if there is a possibility of placing the reactor at another location with less probability of earthquakes?" - This is a question which the Pacific Gas and Electric Company might reasonably be

Mr. Steinbrugge predicts the confinement would hold whether on sand or not. Added safety is seen if both the turbine and the reactor are located on sand; however, both located on bedrock is preferred by him. The Committee concluded that even though there is divided opinion as to whether the rock or the proposed sand foundation is better, the applicant has proposed the sand design for a large movement; and, consequently, the Committee must consider this.

Mr. Price and Dr. Doan joined the Committee briefly during this Executive session; some of the Committee professed a lack of understanding of the RS report on the reactor. It was pointed out to Mr. Price that if the applicant complies with the design precautions for the largest quake possible, even though he sees this quake as incredible, there is no reason left for RS concern over the magnitude of quakes.

Regulatory Staff (RS)

The Regulatory Staff has had an increasing number of questions to the Pacific Gas and Electric (PG&E) Bodega Bay Reactor Group.

Dr. Doan reviewed the position of the Regulatory Staff; although novel designs to resist a large earthquake of perhaps three feet movement can be realized, these are untestable. The quakes are deemed likely because of the history of the San Andreas fault. To him, the public benefits from such a reactor must be balanced against the hazards. Mr. Osborn pointed to the inability to test other reactor arrangements, e.g., the pressure suppression systems. Mr. Rogers said that other structures, e.g., bridges and dams, have been built to resist earthquakes without testing; consequently these present plans for the Bodega Reactor do not seem to him to be a very great extension in existing engineering practice. Dr. Williams recounted the change in RS position as to the amount of quake movement possible. Dr. Thompson recalled the contradictory statement about a reactor being safer close to a fault (as claimed earlier) than further away; he noted that any requirement of reactor sites away from faults, coupled with the need for cooling water, could lead to decisions against reactors in California, and rejection of the Bodega site might generate questions from utilities which would be difficult for the RS to answer. Dr. Beck also commented that if this site is excluded it will be difficult to approve other California reactors. Mr. Price reported that the large earth movement was mentioned relatively late in the review and at a time not allowing changes in the procedures. If the excavation fault had been known earlier, past actions of the Regulatory Staff would have been different; however, Mr. Price said he must act on the situation as it now stands.

There was a review of the site criteria as applied to Bodega location. According to Dr. Silverman, the site had been

The Staff report was accompanied or followed by opinions from its various advisors. The U.S. Coast and Geodetic Survey recommended that the plant be designed for a maximum ground accelaration on rock of 2/3 g and that ground accelerations as high as 1 g should be taken into account. They also recommended design for 50 feet tsunamis from nearby severe marine earthquakes and 30 feet tsunamis from distant generating areas. Consultant Newmark wrote a fairly lengthy report in which he concluded that:

> The structural integrity and leak tightness of the containment building can be maintained under the conditions postulated. However, certain precautions must be considered, especially in the design of umbilicals and of penetrations to the containment building.

The ACRS completed its re-review of the construction permit application for Bodega Bay at the 58th meeting, October 7-10, 1964. A few excerpts from the meeting minutes follow:

> Mr. Steinbrugge considers the site suitable for a reactor, from a seismic point of view. According to him, reactors can be safely build near a fault (but not on it) if the foundations are good; this is preferable to being away from a fault on a poor structure, e.g., on a mud flat.
>
> Mr. Steinbrugge acknowledged the large amount of judgment in designing for seismic forces; however, he believes that designs to resist earthquakes are feasible. He sees .33 to .5 g as the seismic forces which the design should resist.
>
> The Bodega Bay rock is a granite, which Mr. Steinbrugge considers as ideal foundation; he prefers building on the Bodega rock rather than on the proposed sand since the latter entails many design features that may not necessarily make the plant more reliable. Although he has not analyzed the design in detail, e.g., the connections, he predicts that the containment structure anchored in granite would move with the granite.
>
> Mr. Steinbrugge does not prefer the three foot annular space around the confinement, although possible from an engineering point of view; allowing the reactor structure to move with respect to the rock adds many engineering problems. If he really believed a three foot movement was possible, he would recommend another location to avoid the design complications.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

October 20, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON BODEGA BAY ATOMIC PARK - UNIT NO. 1

Dear Dr. Seaborg:

At its fifty-fifth meeting on May 7-9, 1964 at Argonne, Illinois, and at its fifty-eighth meeting on October 7-10, 1964, the Advisory Committee on Reactor Safeguards again considered the proposal of Pacific Gas & Electric Company to construct and operate a 1008 MW(t) boiling water reactor on Bodega Head north of San Francisco, California. The Committee had the benefit of oral discussion with representatives of the applicant and its consultants, with the AEC Regulatory Staff and its consultants, including staff members of the U. S. Geological Survey (USGS) and the U. S. Coast and Geodetic Survey (USC&GS) and of the reports cited below. Subcommittee meetings were held July 31, 1962 and March 20, 1963 and members of the Committee again visited the excavated site on June 3, 1964. Numerous information meetings were held with the applicant, the AEC Regulatory Staff, and with consultants.

This proposal had been considered at the Committee's forty-seventh meeting and reported on in its letter of April 18, 1963 which stated:

> "Tentative exploration indicates that the reactor and turbine buildings will not be located on an active fault line. The Committee believes that if this point is established, the design criteria for the plant are adequate from the standpoint of hazards associated with earthquakes. Careful examination of the quartz-diorite rock below should be made during building excavation, to confirm this point. Furthermore, the Committee suggests that, during design, careful attention should be given to the ability of emergency shutdown systems to operate properly during and subsequent to violent earth shocks, and to the stress effects that might be introduced because the reactor building and the turbine building are to be anchored in different geological formations. The need for earthquake-induced shutdown and isolation of the primary system can be considered at a later time."

considered a very good one originally and, to him, the important question is can the plant withstand three feet of earth movement and still be shut down safely to protect the public.

Dr. Newmark (RS consultant) commented on the reactor design to resist quake forces; although expensive, he conceives no technical problems in designing the reactor to resist large quake movements. Although major structures, e.g., bridges, have not been subject to planned testing with earth shocks, earthquakes have given much information. Bridge failures have added knowledge and there is much information on dynamic behavior of structures from the nuclear weapon tests. Dams have moved a matter of feet in quakes and earth dams have failed. All this leads Dr. Newmark to be confident about the integrity of the proposed reactor structure. However, Dr. Doan stated that none of these tests had involved ground shear motion. The sand pad is to damp the pulses from an earthquake, and, even if compacted, Dr. Newmark believes it should still shear easily; he prefers the sand rather than the reactor structure on the granite. As a parallel to the behavior of concrete structures under dynamic forces, Dr. Newmark pointed to the concrete ships, which have successfully withstood acceleration and explosive forces at sea (these ships contained about 70% as much steel as an all steel vessel of the same size, but the steel is of a cheaper variety).

The ACRS completed and issued a letter report on Bodega Bay at its 58th meeting. The letter, dated October 20, 1964, is reproduced below, followed by a public announcement released by the AEC on October 26, 1964.

of testing or other experimental verification of structural design features associated with earthquake protection; provisions to assure adequate cooling water in case of damage to normal and emergency supply systems; core behavior during earthquakes; design and tests of critical plant components such as instrumentation, isolation valves, and control rod operating mechanisms to withstand earthquake damage; additional considerations which may be needed if zirconium clad fuel is to be used.

The Committee recognizes that the applicant has accepted very conservative values for earth shear movement, earthquake magnitudes, and tsunami heights as design criteria. These criteria should not be construed as precedents for use elsewhere.

With due consideration being given to the items discussed above, the Advisory Committee on Reactor Safeguards is of the opinion that the power reactor facility as proposed may be constructed at this site with reasonable assurance that it may be operated without undue hazard to the health and safety of the public.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

References Attached.

The exploration suggested in the above comment has been completed, and the geologic features discovered have led to further structural considerations in the design. These geologic features include fractures in the underlying rock. One has been identified as the so-called "shaft fault". The character, extent, and age of the most recent activity of this fracture are controversial. Nevertheless, the applicant has considered its significance in the proposed structural protection.

Proximity of the site to the San Andreas fault system has been given careful consideration. The Committee has been advised by several consultants that, during the life of the proposed reactor, there is a high probability that the reactor site will experience at least one major earth shock. There is associated with such an earthquake a remote possibility that the plant will be subjected to the effect of a shearing motion in the rock on which it would be built. The USGS and USC&GS have proposed values for the intensity and accompanying earth motions, including shear, which could be anticipated during the worst earthquake. Determination of these values has been hampered by lack of authoritative historical records and reliable measurements. The applicant and his consultants believe that lower values are more realistic. The Committee considers that the USC&GS and USGS values are conservative.

The applicant has proposed methods for mechanical and structural design to meet the predicted seismic occurrences. The applicant also has proposed to design the building to withstand up to three feet of shear displacement along any plane at the site. The Committee believes that the engineering principles and general design proposed to incorporate them are sound. These considerations afford that degree of assurance required for protection of the reactor in the unlikely event of the predicted maximum earthquake.

The USC&GS has recommended a design height for tsunami run-up at Bodega Head. The applicant stated that the facility design and safeguard procedures will be such the plant would withstand such a tsunami safely.

The Committee is of the opinion that the applicant's design objectives may be accomplished within the scope of present engineering knowledge.

Many details of the proposed design have not yet been completed. It is understood that the applicant will continue to give careful attention to the following items during design and construction: limitations on the maximum reactivity of individual control rods; provisions to accommodate possible seismic earth movements and shear displacement; consideration reactor at the south end of Bodega Head, a peninsula separating Bodega harbor from the Pacific Ocean. The proposed reactor site is approximately 1000 feet west of the western edge of the San Andreas fault zone.

The application for a provisional construction permit was made on December 28, 1962. Nine amendments to the application have been received. In addition to information provided by the company and its consultants, the ACRS and the Staff have received data from a number of AEC consultants, including the U. S. Geological Survey, the U. S. Coast and Geodetic Survey, and Dr. Nathan M. Newmark, Professor of Civil Engineering at the University of Illinois.

Copies of the report of the ACRS and the analysis by the AEC Regulatory Staff are attached. Copies of PG&E's application and amendments to the application, and copies of the reports of the Geological Survey, the Coast and Geodetic Survey, and Dr. Newmark are available for inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., and at the Commission's office at 2111 Bancroft Way, Berkeley, California.

PUBLIC ANNOUNCEMENT

AEC RELEASES TWO REPORTS ON PROPOSED REACTOR AT BODEGA HEAD, CALIFORNIA

The Atomic Energy Commission today is making public two reports concerning the safety aspects of a nuclear power plant proposed by Pacific Gas and Electric Company at Bodega Head, approximately 50 miles north of San Francisco.

One report is from the Commission's Advisory Committee on Reactor Safeguards, a group established by law to advise the AEC on safety matters involved in reactor construction and operation. The ACRS has concluded that there is reasonable assurance that the proposed reactor can be constructed and operated at the Bodega Head site without undue hazard to the health and safety of the public.

The other report, by the Division of Reactor Licensing, has been issued by the AEC Director of Regulation, whose staff makes safety reviews of reactor licensing applications. The Regulatory Staff has concluded that "Bodega Head is not a suitable location for the proposed nuclear power plant at the present state of our knowledge."

Under AEC regulatory procedures, a decision by the Commission on PG&E's application will not be made until after the holding of a public hearing and issuance of an initial decision by a three-member atomic safety and licensing board.

Pacific Gas and Electric Company has applied to the Commission for a permit to construct a 325,000 electrical kilowatt boiling water emergency equipment designed to assure the safety of the reactor.

The fact that a novel method is proposed for safeguarding the Bodega Head reactor against differential ground movement of its foundation rock is not in itself a cause for concern. The nuclear power industry is replete with new methods of coping with a large variety of problems never previously encountered until the arrival of the atomic age. What is of concern is the lack of any experimental or experience proof-test of the proposed novel method that could form an acceptable basis for the required safety evaluation.

The pressure-suppression concept of reactor containment was completely novel when it was first proposed for use at the Humboldt Bay Nuclear Power Plant. In many respects it was much simpler than the pedestal concept for safeguarding against shear movement of foundation rock proposed for the Bodega Head reactor. But it was not approved for use as reactor containment until after the successful conclusion of a long series of engineering tests at the PG&E's Moss Landing power generation plant which proved beyond reasonable doubt that the concept could be utilized safely for boiling water reactors. Similar proof-test standards have consistently been applied to other new and previously untried features incorporated from time to time in nuclear power plants.

The fact that meaningful proof tests are difficult to achieve in the case of seismic safeguards does not, in our opinion, constitute a valid reason for accepting these safeguards in critical areas on the basis of theoretical reasons along. Nor does it justify relying on opinions as to the feasibility of the proposed seismic safeguards unless these are are supported by somewhere near the same kind of experimental evidence required of all other safeguards. We do not see such evidence in support of the pedestal concept for safeguarding the Bodega Reactor against differential ground motion.

Whether the public benefits to be gained from operation of the Bodega Nuclear Power Plant are high enough to justify acceptance of the added uncertainties involved in the seismic design of the plant to withstand several feet of shear ground movement is, of course, a matter of judgment.

The regulatory requirement that there be "reasonable assurance" that any licensed nuclear reactor can be built and operated without undue risk to the health and safety of the public recognizes that there is no way of eliminating all of the uncertainties; experienced judgment is therefore required. The standard of "reasonable assurance" is more difficult to meet when it becomes necessary to take into consideration external forces having the potential of invalidating some of the safeguards build into the reactor installation. The difficulty is enlarged when there are uncertainites in the design measures intended to counteract the external forces. Somewhere along the line enough uncertainties will create

Some excerpts from the Summary Analyses of the Regulatory Staff follow:

> The foregoing proposal for safeguarding the Bodega Reactor and its containment structure against the postulated shear differential ground motion embodies concepts which are in many respects novel and for which little or no precedent exists. The Regulatory Staff Consultants, Dr. N. M. Newmark and Mr. Robert Williamson have come to the conclusion after carefully studying the basis of the proposal. that the structural integrity and leak-tightness of the containment building can be maintained under the earthquake conditions postualted.
>
> They point out, however, that certain precautions must be considered, especially in the design of umbilicals and of penetrations to the containment building.
>
> It is difficult to evaluate the public safety risks involved in a new type of construction for which there is little or no experience background. Other reactor installations have presented seismic design problems, but not in the extreme form presented by Bodega Head under the earthquake postulated by the Coast and Geodetic Survey and the Geological Survey. One problem which the Bodega plant has in common with all other nuclear power plants subject to special seismic design considerations is the inability to conduct any sort of performance test on the finished structure that will demonstrate that the design objectives have been achieved. The uncertainty presented by this situation has been accepted as a reasonable risk in the seismic design at other locations where the only problem involved is the ability to withstand vibrational stresses, since this aspect of earthquake design is well understood and has a considerable amount of experience background. There is difficulty in applying the same philosophy at Bodega Head, however, because of the necessity of considering the additional problem of designing the reactor structure to safely withstand differential ground motion as well as high vibrational stresses, and because there is no realistic way of evaluating the proposed solution to the problem.
>
> The fact that the proposed site is adjacent to the San Andreas fault zone makes it almost certain that it will be subjected to one or more severe seismic disturbances during the lifetime of the plant. While there is a high probability that the plant under the proposed design could survive the vibrations from even a very large earthquake without damage, it must be recognized that if such an earthquake should also involve several feet of shear ground movement as well as ground accelerations as high as 2/3g to 1.0g there is presently no sound experimental or experience basis for predicting the extent of damage that might be incurred by the reactor containment structure and

safety of the public. We have carefully considered the views of the ACRS. We have the highest respect for those views and we do not lightly reach an opposite conclusion. This is a kind of case, however, on which reasonable men may differ. In our view, the proposal to rely on unproven and perhaps unprovable design measure to cope with forces as great as would be produced by several feet of shear ground movement under a large reactor building in a severe earthquake raises a substantial safety question.

In all respects except one the proposed design of the Bodega Nuclear Power Plant provides reasonable assurance that the plant can be built and operated without undue risk to the health and safety of the public. However, the single exception is quite important if one accepts the credibility of an earthquake of sufficient magnitude to cause a major displacement of foundation rock underneath the plant. Although there is a wide difference of expert opinion on the credibility of such an earthquake, prudent judgment favors accepting the conservative recommendations of the USC&GS and the USGS. On this basis and for reasons given above, it is our conclusion that Bodega Head is not a suitable location for the proposed nuclear power plant at the present stage of our knowledge.

a situation in which the "assurance" can no longer be said to be "reasonable." We believe that this situation would exist if the proposed pedestal concept of seismic design were to be approved without more convincing evidence of its validity than is afforded by presently available information.

Conclusions

The containment and all of the emergency equipment for shutting down the Bodega reactor and maintaining it indefinitely in a safe condition in the absence of seismic disturbances are designed on the basis of well-established engineering principles. They can also be tested to ascertain that the design objectives have been achieved. Consequently, there is a high degree of assurance that the reactor can be built and operated without undue risk to the health and safety of the public in the absence of seismic disturbances.

The seismic design of the reactor structure to withstand purely vibrational effects is also based on well-established engineering principles which in some cases at least have been verified in the presence of earthquakes. Thus, while it is not possible to carry out any measurements on the finished structure to assure that the seismic design objectives have been accomplished, there is sufficient experience background to justify a conclusion that the specified seismic vibrational criteria can be achieved and that the plant can therefore be safeguarded against any credible earthquake that does not rupture the foundation rock.

We believe there is room for reasonable doubt, however, that a comparable situation exists with respect to that particular aspect of the proposed seismic design of the Bodega reactor structure intended to assure that the containment and reactor shutdown functions will remain intact in the event of a shear displacement of its foundation bedrock as great as three feet in any direction. While the proposed engineering principles appear reasonable, experimental verification and experience background on the proposed novel construction method are lacking. If approved, this would, to the best of our knowledge, be the first attempt on record to design a building structure and its associated vital equipment to withstand the effects of substantial movement in its foundation simultaneously with the vibration accompanying a severe earthquake. Because of the magnitude of the possible consequences of a major rupture in the reactor containment accompanied by a failure of emergency equipment, we do not believe that a large nuclear power reactor should be the subject of a pioneering construction effort based on unverified engineering principles, however sound they may appear to be.

The Advisory Committee on Reactor Safeguards has reached the conclusion that the reactor can be constructed and operated at the proposed location without undue risk to the health and

5.4 THE MALIBU NUCLEAR PLANT

The Malibu (or Corral Canyon) reactor site and reactor had received preliminary consideration by the ACRS and Regulatory Staff as part of the review of potential reactor sites and reactor concepts which was conducted for the Los Angeles Department of Water and Power (LADWP) in 1962. By mid-1964, when a construction permit review was reaching its culmination, additional seismic questions had arisen, partly from matters directly related to the site, such as landslides, and partly from the increased consideration of seismic matters in California arising from the Bodega Bay review and the Alaskan earthquake of 1964.

The ACRS Subcommittee meeting of June 18, 1964, on Malibu considered several seismic design questions, but no very difficult obstacles seemed to arise. At the 56th meeting, July 9-11, 1964, there was considerable discussion of seismic matters, particularly the potential height of tsunami waves at the site. The ACRS concluded it could write a letter favorable to construction of the Malibu reactor, subject to certain reservations, as shown in the ACRS report which is reproduced on the following pages.

The Regulatory Staff had brought the U.S. Geological Survey and the U. S. Coast and Geodetic Survey into the case as advisors, and the Staff report to the ACRS which was received on July 1, 1964 concluded that a seismic design acceleration of 0.3 was acceptable and that the probability of potential hazard to the public from differential ground movement due to an earthquake at the site was low enough to be disregarded.

In the months following July, 1964 there was a very considerable discussion between the applicant and the Regulatory Staff and its consultants concerning the actual seismic engineering criteria, stress limits, and analytical methods to be used. This was probably the first reactor to receive such detailed evaluation of seismic engineering considerations, and out of this review evolved much of the approach which was generally adopted for upcoming reactors. Among other matters, the capability of the containment to withstand concurrent loads from a postulated LOCA and SSE was required and examined in detail.

By December, 1964 the Regulatory Staff had satisfied itself with the seismic engineering approach which had been developed, and in that month the USGS issued a report which accepted the proposed seismic design bases and concluded that "the probability of permanent ground displacement by faulting in the Corral Canyon site in the next 50 years is negligible," although faulted deposits, probably less than 100,000 years old, had been exposed in a recently opened test trench at the site.

At the 60th meeting, December 10-12, 1964, and the 61st meeting, January 14-16, 1965, the ACRS reviewed the Malibu reactor, and in a letter dated January 25, 1965 concluded that the seismic engineering approach was adequate and that, subject to previous reservations, the reactor could be constructed with reasonable assurance that it could be

Thus, the final positions of the ACRS and the Regulatory Staff disagreed with regard to the acceptability of the Bodega Bay site. Such a disagreement had not occurred previously and it created a considerable stir when it occurred.

Although the Regulatory Staff position of April 30, 1964 had been to reject the Bodega Bay site, the Regulatory Staff position in their report to the ACRS for the October, 1964 ACRS meeting had been less definite. The latter report acknowledged that reputable seismic engineers stated that a design for surface offset of 2 feet could be made; it also acknowledged there were policy considerations involved. However, no final conclusion was drawn in the report; and in their discussions with the ACRS at the October meeting, the Regulatory Staff did not state that their final position was to reject the site.

Undoubtedly, the ACRS thought that its recommendation was going to become that of the Regulatory Staff. The ACRS appeared to be rather skeptical that displacements as large as 2-3 feet were an appropriate design basis; and when Housner, Newmark and Steinbrugge all said that a design could be accomplished, the Committee accepted this as adequate.

PG&E withdrew its application in the face of the Regulatory Staff decision. Looking back with roughly 15 years of hindsight, it appears likely that the proposed design bases for vibratory motion might not have been acceptable, after the experience obtained from the 1971 San Fernando earthquake. As more strong motion accelerograms were obtained from locations near the source of major earthquakes, and as increased knowledge of earthquake generation developed, the specification of accelerations much larger than 2/3g for sites so close to a major fault has occurred. The Bodega Bay reactor, as proposed in 1964, might have difficulty in satisfying today's increased vibratory motion requirements. Additionally, since the probability of a large earthquake at the site is close to unity over the reactor lifetime, this could have posed very difficult problems for an already constructed reactor.

The difference in final opinion between the ACRS and the Regulatory Staff came as a surprise to the ACRS, and there was considerable discussion concerning the procedural and technical aspects of the matter. It was agreed that, in the future, steps would be taken so that the final positions of each group were known to both groups prior to issuance of final reports.

The total containment feature of the building is to be achieved by providing two complete steel liners separated by a layer of porous concrete. The space between the liners will be maintained at a sub-atmospheric pressure by continuously pumping back air to the containment volume. An air recirculating and cooling system is required to remove any heat that is generated within the containment volume. Power and water to assure operation of these systems under all conditions must be provided.

Detailed design of the reactor core has not been established yet, but the general features will be similar to those of other nuclear plants proposed for construction by the same nuclear contractor, and expected to be tested in operation prior to completion of the Malibu plant. Nuclear reactivity coefficients are expected to be negative in this reactor. The probability and effects of control rod ejection require further evaluation. The applicant has suggested several possible means of limiting the consequences of such an accident, and the Committee believes that this question can be resolved satisfactorily during the design stage.

Although stainless steel cladding is planned for the first core, it is anticipated that zirconium alloys may be used in future cores. Complete information on the effect of a possible zirconium-water reaction on the course of accidents is not available. Hence, further review will be needed prior to use of zirconium alloy clad cores.

The Committee was informed that the geology of the site was suitable for the proposed construction. It was reported that no active geological faults are present at the site. Grading of the canyon slopes is proposed to ensure that potential landslide motion does not present a hazard to the plant. It is proposed that critical structures be designed for a suitable response spectrum associated with an earthquake which has a maximum acceleration of 0.3 g. occurring when the containment is under the pressure associated with an accident. The resulting stresses will not exceed 80% of the minimum yield value. Components within the building will be designed to withstand 0.3 g. acceleration acting simultaneously in horizontal and vertical planes.

The ability of the plant to withstand the effects of a tsunami following a major earthquake has been discussed with the applicant. There has not been agreement among consultants about the height of water to be expected should a tsunami occur in this area. The Committee is not prepared to resolve the conflicting opinions, and suggests that intensive efforts be made to establish rational and consistent parameters for this phenomenon. The applicant has stated that the containment structure will not be impaired by inundation to a height of fifty feet above mean sea level. The

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 15, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON CITY OF LOS ANGELES - MALIBU NUCLEAR PLANT - UNIT NO. 1

Dear Dr. Seaborg:

At its fifty-sixth meeting at Brookhaven National Laboratory on July 9-11, 1964, the Advisory Committee on Reactor Safeguards reviewed the proposal of the City of Los Angeles to construct and operate a 1473 MW(t) pressurized water reactor, Malibu Nuclear Plant - Unit No. 1, at Corral Canyon, twenty-nine miles west of Los Angeles. The Committee had the benefit of discussions with representatives of the Department of Water & Power of the City of Los Angeles, Westinghouse Electric Corporation, Stone & Webster Engineering Corporation, the AEC Staff, their consultants, and of a Subcommittee meeting on June 18, 1964. The Committee also had the benefit of the documents listed below.

The proximity of large population centers and the probable growth of population in the vicinity of the proposed reactor site require dependence on engineered safeguards to limit the consequences in the unlikely event of a major credible accident. For this reason, safeguard provisions more extensive than those normally employed in nuclear power reactor plants must be provided in lieu of the distance factor to protect the public.

The applicant has proposed as engineered safeguards a novel containment structure intended to prevent any leakage to the environment, and additional features consisting of:

1. A reinforced concrete containment structure.

2. A containment volume spray system, and

3. An emergency borated-water injection system.

operated without undue risk to the health and safety of the public.

Construction of the Malibu reactor was contested at the hearing of the Atomic Safety and Licensing Board (ASLB) where the adequacy of the seismic design was one of the major points of contention. The intervenors in the case had the benefit of several well qualified consultants in the field, and a considerable technical discussion ensued.

The unfolding developments in the next several months are summarized in an interesting way by several documents which are listed and reproduced below.

1. Memorandum from R. F. Fraley to ACRS members dated March 18, 1965.

2. Letter from Thomas B. Nolan (Director, (USGS) to Harold Price dated March 19, 1965.

3. Memorandum from M. C. Gaske (an ACRS Staff Engineer) dated July 26, 1965.

4. Memorandum from R. F. Fraley to ACRS members dated September 28, 1965.

On March 28, 1967, the Atomic Energy Commission issued a final decision (USAEC, 1967) in which they upheld the decision of the ASLB. The landmark decision, in which an intervenor against a nuclear plant won his case, is reproduced on the following pages. Of some general interest is that, by not accepting, as having established an acceptable risk, the data that movement had not occurred in the last 10,000 years and possibly for 180,000 years, the AEC was establishing, in a crude way, a benchmark on "How Safe is Safe Enough?"

integrity of emergency in-house power supplies should also be assured by location at a suitable height and by using water-proof techniques for the vital power system. The emergency power system should be sized to allow simultaneous operation of the containment building spray system and the recirculation and cooling system. Ability to remove shutdown core heat under conditions of total loss of normal electrical supply should be assured. If these provisions are made, the Committee believes that the plant will be adequately protected.

The applicant has proposed to deny entrance to the containment while the reactor is operating. This mode of operation does not permit frequent surveillance of equipment and prompt detection of incipient defects. Operating experience at other power plants has demonstrated the value of accessibility for inspection. The Committee suggests that the applicant reconsider this question and explore design modifications which will allow entrance without violating the containment integrity.

As the Committee has commented in its earlier letters, the hold-up of routine gaseous and liquid releases may be necessary during unfavorable conditions. In this connection, it will be necessary to conduct additional pre-operational meteorological and oceanographic survey programs.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be suitably dealt with during construction, and that the proposed Malibu Nuclear Plant can be constructed with reasonable assurance that it can be operated at the site without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

References Attached.

5-66

IN REPLY REFER TO:

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

212

MAR 19 1965

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
4915 St. Elmo Avenue
Bethesda, Maryland

Dear Mr. Price:

The enclosed report entitled "Geologic Investigations, December 1964 to March 1965 at the Proposed Nuclear Power Plant Site, Corral Canyon Los Angeles County, California," by C. M. Wentworth and R. F. Yerkes is the result of the investigations we have carried on at the Corral Canyon site in the last few months. The primary purpose of these investigations was to determine the age of the youngest faulting that has occurred in this area. From study of the soils and the faulted surficial deposits, it has been determined that faulting occurred more than 10,000 years ago. In addition, we have estimated the probability of permanent surface ground displacement in the Corral Canyon area from an earthquake of magnitude 7 to 7.5, occurring in the vicinity of the site as postulated recently by Kamb and Benioff. We have concluded that permanent surface ground displacement would occur but that the amount of displacement could not be predicted from the geologic evidence.

We do not object to making this report a part of the public record.

Sincerely yours,

Thomas B. Nolan

Director

Enclosure

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

March 18, 1965

MEMORANDUM

To : ACRS Members

From : R. F. Fraley, Executive Secretary ACRS

Subject: MALIBU NUCLEAR PLANT, HEARING TESTIMONY

The hearing testimony of the LADWP consultants has been developed for the Malibu hearing which begins on March 23, 1965.

The testimony of Hugo Benioff has raised a serious question regarding the magnitude of an earthquake which could occur on faults in the vicinity of the site. He predicts a magnitude not greater than 7.25 in contrast with the USC & GS conclusions that at 5.5 magnitude quake is the largest that could occur on the Malibu fault at the site. The LADWP consultants conclude that the plant design is adequate to withstand this situation (e.g. 0.3g maximum acceleration and no ground surface displacement). I understand, however, that the USC&GS does not agree with these conclusions if one accepts a 7.25 quake at the site. They apparently feel that an acceleration as high as 0.6g could occur and surface displacement is a possibility for this size disturbance.

center near Corral Canyon will occur during the next 50 years is very low. ...the probability of permanent displacement of the ground surface by faulting at Corral Canyon during the next 50 years is very low...".

It is not clear whether the above report is favorable or not to the granting of a construction permit for the Malibu reactor. No ACRS action regarding the above appears warranted.

Attachments:
1. USGS letter dated 7/14/65
2. USGS Report dated July 1965

cc: Remainder of ACRS, w/o encs.
D. Duffey, w/o encs.

UNITED STATES GOVERNMENT

Memorandum

TO : L. Silverman, City of L.A. (Malibu) Subcte Chmn DATE: July 26, 1965
W. D. Manly

FROM : M. C. Gaske, ACRS Staff

SUBJECT: CATEGORY B REPORTS - CITY OF L. A. (MALIBU) - USGS REPORT, DATED JULY 1965, AND TRANSMITTAL LETTER, DATED JULY 14, 1965.

At the request of the Commission, the U. S. Geological Survey made a geologic study of the mouth of Corral Canyon and the adjoining area near the proposed Malibu reactor site. Results of this study are contained in the USGS report, "Structure, Quaternary History, and General Geology of the Corral Canyon Area, Los Angeles County, California," dated July 1965. This report was forwarded by the USGS to the Director of Regulation by letter dated July 14, 1965, and copies were then transmitted by the Regulatory Staff to the Atomic Safety and Licensing Board by letter of the same date. The report is accompanied by detailed maps and sections of the proposed Malibu reactor site. It contains data obtained from exploratory trenches and laboratory analyses subsequent to recess of the public hearing on April 1, 1965 as well as previously reported information regarding the site.

The recess of the public hearing on April 1 was to permit excavation at the site to obtain further information regarding the geology and seismology of the site area. The public hearing was originally scheduled to reconvene on June 21, 1965 but was postponed until July 19, 1965 to permit the USGS to complete the report on the results of their study.

Faults of several magnitudes have been found to be present in bedrock at the proposed reactor site. All demonstrable fault movement at the site is reported to be more than 10,000 years of age. Eighty feet of a 270 foot long exploratory trench exposed mudstone and sandstone that is within a landslide. A fault crosses one of the excavated trenches at a point approximately 35 feet from the center of the proposed reactor location.

The USGS report states that "Because surface faulting has commonly accompanied earthquakes of Richter Magnitude 6.0 or greater in Nevada and California, and because estimates of the largest earthquake ever to be expected along the Santa Monica fault system range as high as M.7¼, the probability of future surface faulting at Corral Canyon must be based in part on the location of any future large-magnitude shocks in the Santa Monica fault system. ...the recurrence interval for large-magnitude faults in this system...is greater than 200 years...and may exceed... 10,000 years.... ...the probability that a large-magnitude shock with within the site but the displacements have not been detected because of generally poor exposures. On the basis of this record the probability of ground displacement at Corral Canyon in the next 50 years is very low."

The Committee in its letter of July 15, 1964 interpreted this report and discussions at the meeting as follows:

"The Committee was informed that the geology of the site was suitable for the proposed construction. It was reported that no active geological faults are present at the site."

The USC&GS report considered by the Committee at the 60th meeting (January 1965) included the following summary:

"The Corral Canyon site is bisected by the east-trending Malibu Coast fault, which is part of a wide, east-trending zone of north-dipping faults, asymmetric folds and shears, the Malibu Coast Zone." "Evidence indicates that the zone is active on a regional scale."

"Based on available geological evidence the probability of permanent displacement on the ground surface by faulting in the Corral Canyon site during the next 50 years is negligible. Seismic shocks can be expected at the Corral Canyon site; more than 54 seismic events of magnitude 4 to 6.3 have been recorded within 62 miles of the site in the past 112 years."

The Committee, in its letter of January 25, 1965, "reiterates its belief that the proposed Malibu Nuclear Plant can be constructed with reasonable assurance that it can be operated at the site without undue risk to the health and safety of the public."

Exploratory trenches were dug at the site during April 1965, at the suggestion of the ASLB and a subsequent USC&GS report was issued in July 1965. Three copies were provided to the ACRS as a Category B Report. (This report was summarized in a memo by M. C. Caske which was distributed to all ACRS Members on July 26, 1965.) The following comments were included in this report:

"Faults of several magnitudes are present in bedrock of the Corral Canyon site. The Malibu Coast fault, about 800 feet north of the reactor location is of regional significance and large magnitude of displacement; where well exposed, its trace is marked by a zone of brecciated and sheared rock as much as 75 feet wide. Faults of lesser magnitude such as fault A near the north boundary of the plant, separate different formational units and are characterized by zones of sheared and brecciated rock up to several feet wide. Such faults can be traced for only hundreds to thousands of feet; they probably have displacements of hundreds of feet. Intraformational faults, such as fault F, exposed in Corral Creek and trench 3 (the reactor-location trench), are characterized by local truncation of structure and are commonly marked by thin, but recognizable zones

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 28, 1965

MEMORANDUM

To : ACRS Members

From : R. F. Fraley, Executive Secretary
ACRS

Subject: MALIBU NUCLEAR PLANT, UNIT NO. 1 - QUESTIONS RELATING TO SEISMIC DESIGN RAISED BY THE ATOMIC SAFETY AND LICENSING BOARD

The Atomic Safety and Licensing Board has recessed the Malibu Hearing until October 19th with an outline of some of the matters that have occupied the Board's concern. This outline, summarized in the attached list, was submitted to the participants "for consideration, comment or submission of evidence, if they deem it necessary, in the October session." E. G. Case has indicated that he would like to discuss item 4 with the Committee during the October ACRS Meeting to determine if the absence of an active fault at the Malibu site was an ACRS requirement, as interpreted by the Hearing Board, or merely a statement of the information presented to the Committee at the 56th ACRS Meeting (July 1964).

It should be noted that the draft USC&GS report considered at the 56th meeting included the following statement:

"All of the known surface ground displacement on the Malibu Coast fault zone is prehistoric - that is, more than 200 years old. If the band of deformed rocks just south of the Malibu Coast fault trail is considered to be part of this fault zone, the most recent ground displacement occurred sometime between about 200 and 400,000 years ago.

"The likelihood of ground displacement at the site due to earthquakes depends on the frequency and severity of earthquakes along the Malibu Coast and related Faults.

"The Malibu Coast fault is considered to be part of an active system that includes the Newport-Inglewood zone. Only 3 to 5 magnitude shocks have been associated with the Malibu Coast fault; none of these has resulted in known displacement at the ground surface in historic time. However, in prehistoric time faulting at eight known localities along the general trend of the Malibu Coast fault has displaced rocks no older than 400,000 years. It can be inferred from these data that similar faulting may have occurred

Summary of Matters Identified by the Malibu ASLB

1. Since the containment building is not specifically designed to withstand ground displacement, has it been established and on what basis: (a) what is the ground displacement that it can withstand; (b) what would be the amount of fission products released if a displacement greater than that identified in (a) occurred; and (c) what would have to be provided in the design to give such resistance?

   The Board suggests that experience with relative ground movement be used as the basis in the Southern California area for the selection of useful values in these parameters.

2. Is it acceptable to grant approval on the basis that the structural requirements are "within the range of accepted practice and established knowledge" even though the detailed design has not been presented?

3. What is the meaning of the phrase "without undue risk to the health and safety of the public" as understood by all participants, especially the word "undue"? References to appropriate authority are requested.

4. The Board interprets the ACRS Report of July 15, 1964 as carrying "the admonition that this reactor should not be located over an active fault." The Board requests standards or suggestive standards to measure an active fault.

September 27, 1965

of sheared rock or breccia. Such features can be traced tens to hundreds of feet; their displacements are probably on the order of tens of feet. Finally, innumerable shears, locally continuous or concentrated in narrow bands, pervade the mudstone of the Corral Canyon site. Minor displacement has occurred on these features, as indicated by disrupted sandstone beds and slickensides. Aggregate displacement across several feet of such sheared rock may amount to several feet. All demonstrable fault movement in the Corral Canyon site is pre-Recent (more than about 10,000 years) in age."

"Comparisons of degree and time of deformation in different parts of the fault system indicate that future faulting is at least as likely to occur in the Malibu Coast zone as in any other part. The available seismic record is not sufficient to establish the recurrence interval for large-magnitude faults in the system; this interval is greater than the approximately 200 years of historic time and it may exceed the approximately 10,000 years of Recent time. As this recurrence interval is large compared to 50 years, the probability that a large-magnitude shock with center near Corral Canyon will occur during the next 50 years is very low. This very low probability, coupled with the lack of evidence for surface faulting in the Malibu Coast zone during Recent time, indicates that the probability of permanent displacement of the ground surface by faulting at Corral Canyon during the next 50 years is very low (this same very low probability was described in the U. S. Geological Survey report of 1964 as negligible, which was used there in the sense of very low). This assessment implies no judgment of public risk; it is not intended as a judgment of the consequences of surface faulting in any particular utilization of the Corral Canyon site."

It should be noted that fault F is about 35 feet northeast of the center of the reactor building which means that it passes under the reactor containment structure.

E. Case will be prepared to discuss this information in more detail.

Attachment:
Summary of Matters Identified by the
Malibu ASLB dtd. 9/27/65

THE POSSIBILITY OF A CONGRESSIONAL INVESTIGATION was raised by Sen. Murphy in a letter to the AEC, wherein he questioned AEC's site selection criteria and the behavior of AEC Regulatory Staff in connection with reactor proposed by the Los Angeles Department of Water & Power at Malibu, Calif.

Sen. Murphy stated that going any further with the project would undermine the public's confidence in the nuclear power industry.

He recommended to the Commission that AEC "Staff should be candid with the applicant", that its site selection program be revised, and that the AEC take a "very close look" at its staff procedures.

The Senator questioned whether AEC Staff had made a "hasty" and "ill conceived" judgment in favor of construction of the Malibu reactor and now desired to "save face" by taking exception (See our V.12, N.36) to the decision of the Atomic Safety & Licensing Board which recommended plant construction with the condition that design criteria provide for permanent ground displacement (See our V.12, N.29).

Responding, Dr. Glenn T. Seaborg, AEC Chairman, told Sen. Murphy in a Dec. 6 letter:

"Your letter of November 21, 1966, raises substantive issues concerning the application of the City of Los Angeles for a construction permit for a nuclear reactor near Malibu, California. The initial decision dated July 14, 1966, by the Atomic Safety and Licensing Board appointed to hear this case is now under review by the Commission. You will understand that for this reason it would be inappropriate for me to discuss the merits of the matter at this time.

"In your letter you object to the right of the Commission's regulatory staff to file exceptions to the initial decision of the licensing board. The Commission's rules constitute its regulatory staff as a party to proceedings such as the instant one in order that the staff may participate fully therein. In this respect, our rules follow those of other regulatory agencies for adjudications under the Administrative Procedure Act. The Commission's regulatory staff has the same right to appeal a preliminary decision of a licensing board as any other party.

"Your letter also suggests that the Commission's 'site selection program' should be revised to prevent consideration of a site which you consider unsuitable from a safety standpoint and which also lies outside the service area of the applicant. Under the licensing procedures established by the Atomic Energy Act, the Commission is obliged to consider the issuance of a construction permit for a proposed facility at the site selected by the applicant."

On Nov. 21, the Senator had written Dr. Seaborg:

"About two years ago some of my constituents called to my attention problems which were arising because of a proposal to build a nuclear power generator at Corral Canyon in the Malibu community. These problems arose because:

On July 14, 1966, the ASLB issued a ruling that it would be necessary for the plant to be designed to withstand differential ground displacement if it were to be constructed but without specifying a quantitative figure; the ASLB thereby sided with the intervenors and against LADWP and the Regulatory Staff. The Regulatory Staff appealed the decision of the ASLB to the Atomic Energy Commissioners themselves. An interesting exchange of correspondence occurred between the United States Senator George Murphy of California and AEC Chairman Seaborg. The exchange is reproduced below in an excerpt from the Atomic Energy Clearing House dated December 12, 1966.

"My view of the Staff's attitude and activities might be more charitable were it not for the fact that during the course of the hearings it was proved that the Staff had caused the wording of a key conclusion of the United States Geological Survey to be changed against the wishes of the authors of the report. This change would make it appear that the U.S.G.S. had concluded that the risk of permanent ground displacement at the site was 'negligible', i.e., could be disregarded. The involved and unconvincing explanations produced by the Staff to justify what seems to me to be highly improper conduct boiled down to the point that the Staff seemed to wish to avoid comparisons with the report of the U.S.G.S. in the Bodega Head case, where the U.S.G.S. had noted, as here, that the risk of permanent ground displacement was 'low' but not 'negligible.'

"In the Bodega Head matter, as I am sure you recall, P. G. & E. proposed a design of the facility that specifically took ground displacement into consideration. The P. G. & E. design, however, was rejected by the Regulatory Staff on the grounds that the design was untested.

"I realize that factually each case must be considered separately, but it also seems clear to me that there must be a high degree of consistency in the safety standards applied to each case. Surely, it would be strange, to say the least, for the Staff to reach one judgment on Bodega Head, and the opposite judgment on Corral Canyon, when the U. S. G. S. had found in each case that the risk of permanent ground displacement is 'low' but cannot be neglected. An additional element of the problem of consistent application of consistent criteria is this:

"How can a Bodega Head facility specifically designed to resist the effects of permanent ground displacement be rejected on the grounds that it is untested, and a Corral Canyon facility, even if similarly redesigned, not meet the same fate on exactly the same grounds?

"You remarked recently in a press conference in California that the AEC only seems to encounter these problems in California. I believe your reference perhaps ran more to the determined opposition which has been manifested to both Bodega Head and Corral Canyon in Malibu, but your remark was also true in another sense: I don't know of anywhere else in the country where applicants have tried to locate nuclear reactors in close proximity to faults in a state which has an unparalleled history of tectonic activity.

"Bearing all this in mind, it becomes easier to understand why the Staff seems so anxious to absolve the Corral Canyon site of any geological taints, why it is willing to force crucial change in wording in a key conclusion in a report by an independent government agency, why it blandly accepts all the evidence which supports its position, and blithely rejects all the evidence which disagrees with that position, and why it takes exception to a finding of the ASLB which the applicant itself is willing to accept.

"(1) It was considered that the land might be subject to earthquakes because of an existing fault;

"(2) The site is close to a well developed community which appears to be expanding into one of the most attractive beach communities in the area, and,

"(3) In the event there ever was an unfortunate experience connected with this construction, the prevailing winds would carry whatever radioactive material was released directly to the adjacent city of Los Angeles.

"One of the constituents involved has been known to me for many years as one of the top geologists on the West Coast. Because of this information, I have watched with great interest subsequent developments and hearings.

"In my opinion, the important point for decision was whether or not the risk of permanent ground displacement exists at this site which would preclude the construction and operation of a plant specifically not designed to withstand the effects of permanent ground displacement and whether or not this condition would create undue hazard to the public health, safety and welfare. It would seem that the intervenors have prevailed before the ASLB on this central point of controversy,

"It has now come to my attention that the Regulatory Staff has filed exceptions to the initial decision of the ASLB, asking, in effect that the ASLB be reversed on the very point to which the vast bulk of testimony in the 41 days of hearings was devoted, to which the ASLB obviously attaches great importance, and to which the Board members plainly gave the most careful consideration.

"It seems to me proper to inquire just what point the ASLB is competent to decide if not this one. Is it the position of the Staff that if the ASLB (after more than a month of hearings, visits to the site, the opportunity to cross examine all the expert witnesses involved, and an intimate first hand knowledge of the entire case) disagrees with the Staff, the ASLB has to be wrong?

"Or do we have here in fact a situation in which the Staff, having made a hasty and possibly ill considered judgment in the first place, based on incomplete and erroneous information, now finds itself backed into a corner, its infallibility successfully challenged, its 'facts' controverted and its conclusions found wanting, is trying to save face? May I remind you that the applicant in this matter did not appeal the initial decision and that following that decision Mayor Samuel W. Yorty of the City of Los Angeles announced that the DWP was going to attempt to redesign the proposed plant to meet the objections of the ASLB. Why, if the applicant itself was willing to accept an unfavorable decision of the ASLB, did the Staff find it necessary to object to the decision?

"2. Your site selection program could easily be revised, as was suggested during the course of the ASLB hearings by a number of prominent geologists, to the end that thorough assessment of the geological factors involved, and other important considerations of acceptability of this type of construction in the areas, all precede the initial application, and not follow it.

"3. The AEC itself might take a very close look at its Staff procedures in regard to these applications. To my mind, the Corral Canyon case raises serious questions concerning your criteria and procedures for licensing nuclear facilities as well as questions regarding the competency and objectivity of the Regulatory Staff in making judgments to the public health and safety.

"These questions are so serious that I have been urged by some of the interested parties to consider whether a Congressional investigation into this whole field may not be necessary. I would be reluctant to ask for such a hearing when such simple, corrective measures seem to be so available."

On March 28, 1967, the Atomic Energy Commission issued a final decision (USAEC, 1967) in which they upheld the decision of the ASLB. The landmark decision, in which an intervenor against a nuclear plant won his case, is reproduced below. Of some general interest is that, by not accepting, as having established an acceptable risk, the data that movement had not occurred in the last 10,000 years and possibly for 180,000 years, the AEC was establishing, in a crude way, a benchmark on "How Safe is Safe Enough?"

"The fact is, Dr. Seaborg, that as far as California is concerned at least, there seems to be something seriously wrong with your site selection procedure.

"It seems to me to be quite obvious that the geological risks involved in building this plant at Corral Canyon should not have been forced to emerge only through a protracted and costly adversary proceeding after the DWP has selected a site which, in addition to the fact that it is geologically and seismically unsound, also lies in territory not under its jurisdiction and for which the DWP has been twice refused a zoning variance by the government body having jurisdiction, i.e., the Los Angeles County Board of Supervisors. The Corral Canyon site lies in the heart of one of California's finest residential and recreational communities and the DWP's choice of Malibu for a major industrial facility flies in the face of the Los Angeles County master plan for the development of this beautiful beach community for residential and recreational purposes.

"A much sounder approach, it seems to me, is the one taken by the State of California, which in contemplation of future development of nuclear facilities, began with an independent evaluation of a number of possible coastal sites. I am informed that of all those surveyed by the State, Malibu was regarded as the worst. If you have not already been informed of the details of this evaluation, I am sure that the California State Department of Water Resources would be glad to supply it to you.

"I am convinced that to go any further with the Malibu project will only lead to further bitterness, and controversy and cost that will undermine public confidence in the entire nuclear power industry and damage the development of nuclear power in California which is so necessary to our development. I am afraid that even if the AEC ultimately issues a license for the construction of this facility, protracted litigation will inevitably follow in the state courts on questions involving the legal authority of the DWP to acquire the land and use it for a nuclear power plant, including among other things, the question of whether or not the DWP has the right to run roughshod over the zoning laws of the County of Los Angeles. It seems that there is a much more practical way to achieve our goal.

"In summary, to my mind the controversy over the Corral Canyon site has gone on long enough. Extensive public hearings have been held, with the result that the DWP and the Staff have lost on the central question before the ASLB -- whether the hazard of permanent ground displacement can be neglected in the design of the facility. The matter should not be dragged out any longer.

"What should be done now, in my opinion, is the following:

"1. The Staff should be candid with the applicant, and point out to the DWP that once the risk of permanent ground displacement has been established, it will be most difficult -- indeed, impossible with any consistency -- for the Staff to approve any untested design in view of the possibility of public danger and the precedent of the earlier Staff decision on Bodega Head. The DWP should be assured that reapplication for another and sounder site will not entail loss of federal subsidies and the DWP should be encouraged to reapply on this basis.

for design against permanent ground displacement but, like the staff, takes exception to the issuance of a provisional construction permit with the design condition. Marblehead's position has been endorsed by The Malibu Citizens for Conservation, Inc., and Lester T. (Bob) Hope, also intervenors in this matter. The applicant filed no exception to the initial decision but did submit a brief in support of the staff's exception to the need for design against permanent ground displacement and, should this staff exception be denied, in support of the present issuance of a provisional construction permit with the design condition, as ordered by the board.

Application for a construction permit and a license pursuant to Section 104(b) of the Atomic Energy Act of 1954, as amended, and Part 50 of our regulations was filed by the Department of Water and Power on November 22, 1963. The application and eight amendments, the last of which was filed on January 8, 1965, were reviewed by the regulatory staff and the Advisory Committee on Reactor Safeguards, both of which concluded that there is reasonable assurance the proposed facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public.

A "Notice of Hearing" was published in the Federal Register on February 9, 1965. This was followed by a prehearing conference on February 26, 1965, and 41 days of hearings held in intermittent sessions thereafter, commencing on March 23, 1965. A posthearing conference on May 19, 1966, concluded this phase of the proceedings. Marblehead Land Company, The Malibu Citizens for Conservation, Inc., and Lester T. (Bob) Hope, intervened in the proceedings and opposed construction of the proposed facility. The County of Los Angeles also intervened in opposition to the application. The State of California, while it intervened and introduced two geological witnesses, did not formally take a position. In addition to the intervenors, there were 62 limited appearances, 11 of these favoring and the balance opposing the application.

The proposed 310-acre site is located in Los Angeles County, but outside the territorial limits of the City of Los Angeles. It is situated at the mouth of Corral Canyon along a stretch of east-west Pacific Ocean coastline, approximately 10 miles west of Santa Monica and 30 miles northwest of the center of Los Angeles. The south side of the site abuts U.S. Highway Alternate 101 which separates the site from Corral Beach, the proposed nuclear containment structure to be located approximately 550 feet from Corral Beach. The Santa Monica Mountains lie immediately north of the site and rise to about 2,500 feet, separating the site from the San Fernando Valley. The area surrounding the site is used primarily for residential and recreational purposes.

The facility would utilize a closed cycle pressurized light water reactor, designed to operate at 1,473 megawatts thermal and to produce 490 megawatts electrical. The reactor would be similar in substantial respects to the Yankee, Connecticut Yankee, Saxton, and San Onofre reactors. A significant safeguard feature of the facility is the proposed housing of the primary system in a massive containment structure, designed so that there would be essentially no leakage to the environment. The containment would consist of an exterior shell of about four feet of reinforced concrete, with two steel liners

DOCKET NO. 50-214

IN THE MATTER OF DEPARTMENT OF WATER AND POWER OF THE CITY OF LOS ANGELES (MALIBU NUCLEAR PLANT UNIT NO. 1)

Issued March 27, 1967

COMMISSIONERS:
GLENN T. SEABORG, *Chairman.*
JAMES T. RAMEY.
GERALD F. TAPE.
SAMUEL M. NABRIT.
WILFRID E. JOHNSON.

DECISION

This matter comes before the Commission for final decision upon exceptions filed by the regulatory staff and by three intervenors to an initial decision of an atomic safety and licensing board dated July 14, 1966. In its initial decision, the board ordered that a provisional construction permit be issued to the Department of Water and Power of the City of Los Angeles (hereinafter the applicant) to build a pressurized water reactor to be located at Corral Canyon, Malibu, California. The board, while finding that the proposed facility in all other respects met the safety requirements of 10 CFR § 50.35(a), imposed the condition that the design criteria be modified and supplemented to include adequate provision for permanent ground displacement (ground rupture) from earthquake activity. The initial decision directs that the modified and supplemented design criteria be made available for such review procedures as the Commission may provide.

The regulatory staff has filed exceptions to the initial decision on two grounds. The staff contends, first, that the board erred in its holding that the probability of permanent ground displacement at the Corral Canyon site is sufficiently high that the proposed design must be modified and supplemented to include criteria for such displacement before there is reasonable assurance the facility can be constructed and operated at that site without undue risk to public health and safety. The staff contends, further, that if the Commission decides the facility must be designed to withstand permanent ground displacement, the criteria for such design accommodation are "principal architectural and engineering criteria" within the meaning of § 50.35 (a)(1) of our regulations and the initial decision should be set aside and the proceeding remanded to the atomic safety and licensing board to determine: (1) the amount of permanent ground displacement which the facility must be designed to withstand; and (2) the adequacy of the design criteria which may be proposed by the applicant to accommodate the permanent ground displacement.

Marblehead Land Company, an intervenor in this proceeding, supports the board's findings and conclusions with respect to the need upon the request of the board. There was considerable dispute during the course of the hearings on the question of whether or not fault F is in fact a fault and then on how large fault F might be. The initial decision states that fault F extends eastward from the containment site about 2,000 feet and that it may extend westward under a landslide as far as Solstice Canyon (an additional 2,000 feet), in which event it may be truncated by another fault. Each of these faults, as well as the Malibu Coast fault, is associated with a band of sheared or fractured rock of varying width.

According to the interpretation of the United States Geological Survey (USGS), formation of faults A and F could have been sympathetic or secondary to the movement on the Malibu Coast fault. In any event, it is the view of the USGS that faults A and F cannot be discussed in isolation from the Malibu Coast fault.

Much time at the hearings was spent in dispute over the origin of the deformation of the rock underlying the proposed facility. The initial decision describes the rock materials as having been folded, faulted, and sheared, and concludes that a tectonic origin is supported by the weight of the evidence. In this connection, the USGS has stated: "The intensely deformed bedrock in the plantsite might be a locus for future displacement along the general trend of the Malibu Coast zone."

The Malibu Coast zone lies within an arrangement known as the Santa Monica fault system. The area of the system wherein this zone is located is described by the initial decision as being tectonically active at depth. While there are no actual measurements of the current amount or rate of local strain accumulation in the Malibu Coast zone, the initial decision recites that the general structural and tectonic environment that produced the strong faulting, folding and shearing in the zone probably still exists.

It was established to the board's satisfaction that "the youngest known displacements at the proposed reactor site most probably occurred more than 10,000 years ago, and possibly as much as 180,000 years ago." The board adds: "There is no known evidence for movement of the rock in Recent time (younger than 10,000 years) on the faults in the Malibu Coast zone."

As the initial decision indicates, the central attention of the board in the proceedings below was directed to the assessment of geological and seismological data and opinion bearing on the need for design against ground displacement. The extensive presentations of the parties in this regard—which we have summarized in part—reflect numerous areas of disagreement with respect to both regional and local geological and seismological considerations. In the exceptions and briefs filed with us, however, these areas of dispute have been greatly reduced and the positions of the parties brought into sharper focus. Neither the applicant nor the intervenors take exception to the geological and seismological findings of the board, and the staff, in framing its first exception, has accepted all of the board's findings in this respect and disputed only the board's conclusion therefrom under 10 CFR § 50.35(a). Moreover, as the staff presents this exception, it has narrowed the safety issue to the problem of assessing the significance, with respect to the probability of future ground displacement, of the accepted fact that there has been no surface faulting at or in separated by two and one-half feet of pervious, "popcorn", concrete. The pervious concrete zone would be maintained at a slightly negative pressure with respect to the atmosphere by a pump-back system.

The facility design contains several other important safeguard systems, including: (1) a safety injection system to inject borated water into the core; (2) a seal water injection system; (3) an internal air recirculation and cooling system to decrease the pressure within the containment structure; and (4) a containment spray system to decrease pressure.

The proposed facility is specifically designed to withstand the effects of ground shaking resulting from earthquake activity. It has not been specifically designed to withstand ground displacement (ground rupture) resulting from earthquake activity.

While the lengthy proceedings before the board involved consideration of a number of safety questions raised with respect to the Malibu facility, the board's initial decision and the parties' exceptions reflect that all of these questions save that relating to the probability of ground displacement from earthquake activity were satisfactorily dealt with to the extent required at the construction permit stage by 10 CFR § 50.35. The board specifically found that the applicant's design criteria for accommodation of the shaking caused by earthquake activity met the requirements of § 50.35. Our own review of the record leads us to agree that the only present outstanding safety question is the one relating to design for ground displacement which the parties have submitted to us on this appeal. It is to that question we now turn.

To facilitate understanding of the basis for our decision today, a brief discussion of the geology and seismicity of the site is appropriate at the outset. Since the parties have not excepted to any of the board's geological and seismological findings, our summary is drawn substantially from the contents of the initial decision.

As is well known, California and other sections of the western United States have a number of areas in which there is earthquake activity. The Corral Canyon site is located in the Malibu Coast zone. This is an east-west trending zone of tectonic deformations, about a mile wide, which runs along the California coastline and includes a number of roughly parallel faults of varying size. On the northern edge of the Malibu Coast zone lies the Malibu Coast fault. This fault bisects the site area about 800 feet north of the proposed reactor containment building. The initial decision recites that the Malibu Coast fault separates two quite different kinds of basement rock, which suggests that it is one of California's major crustal boundary faults.

Among the other faults in the Malibu Coast zone, two warrant specific mention: fault A, which crosses Corral Canyon about 150 to 200 feet north of the proposed containment location and extends eastward therefrom about 8,000 feet, with a westerly extension which is not well defined; and fault F which lies directly beneath the proposed reactor.[1] Fault F had not been discovered at the time of the ACRS and initial staff considerations. It was uncovered during the hearings when a trench was dug across the proposed reactor location

[1] The designations of these features correspond to those used by the board in its initial decision. For purposes of more graphic description, we are appending to our decision the maps of the site which were included in the initial decision.

It might be noted, parenthetically, that this same evidence appears to be the basis for the opinion in the USGS report that the probability of surface displacement through faulting at the site during the next 50 years is very low.

We see no need here for an extended semantic discussion of whether this evidence should be characterized as "negative" or "positive". In our view, evidence indicating the absence of an occurrence of this type plainly has some probative value. This evidence, however, cannot mark the end of our inquiry, since we must proceed to ascertain its significance in the light of geological and seismological knowledge available to us in the record.

Unfortunately, the expert witnesses at the hearing could not, in the light of the evidence, speak with much assurance regarding future surface faulting during the life of the proposed facility. The initial decision notes that all of the seismologists in the proceeding implied or stated that more data were needed before worthwhile predictions could be made and the board cites similar uncertainties with respect to the present state of the art of geology. Thus, while the board accepts that the probability of surface faulting during the lifetime of the facility may be "low", uncertainties in the data available lead it to find ultimately "that reasonable certainty or assurance does not exist regarding all pertinent phases of earthquake activity."

The foregoing are necessarily considerations to be taken into account in making the judgment required by Section 50.35(a). These considerations, moreover, must be viewed in the light of the following matters of record: that the general structural and tectonic environment which produced the strong faulting, folding and shearing in the Malibu Coast zone probably still exists; that future fault movement, if any, is likely to occur along existing breaks; that fault F passes directly through the ground upon which the proposed reactor is to stand; and that, aside from fault F, the intensely deformed bedrock in the plant site might be a locus for future displacement along the trend of the Malibu Coast zone.

In the proceedings before the board, the applicant attempted to provide a basis for the inference which it draws from the lack of surface faulting for 10,000 or more years, in its interpretation of the "recurrence interval theory". It assumes that the earthquake phenomenon can be described by a model in which there takes place a series of cycles involving a slow accumulation of strain followed by a sudden release of energy and that these cycles will have a roughly constant recurrence interval. The applicant reasons that the average probability of an earthquake occurring in a year will be the inverse of the average recurrence interval and that the longer the time since the last displacement the more likely it is that the recurrence interval is large. However,as the testimony of the United States Geological Survey indicates, it is not known whether the recurrence interval concept has validity. We note, further, the response to this argument by the intervenors that, even assuming the validity of the theory, the regular repetition of events which it envisages implies instead that the probability of earthquake events increases progressively as the time since the last event passes and we approach the next event.

the vicinity of the site for at least 10,000 years and possibly as long as 180,000 years.

In reaching our decision in this matter, we are called upon to apply the standard enunciated in § 50.35(a)(4) of our regulations, i.e., whether there is "reasonable assurance" that:

> "(ii) taking into consideration the site criteria contained in Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public".

The only elements of our regulations bearing specifically on this design-siting problem are to be found in 10 CFR § 100.10(c)(1) and (d):

> "(c) * * * (1) The design for the facility should conform to accepted building codes or standards for areas having equivalent earthquake histories. No facility should be located closer than one-fourth mile from the surface location of a known active earthquake fault."
>
> * * * * * * *
>
> "(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards."

While the second sentence in subsection (c)(1), above, was the subject of much discussion during the hearings, the initial decision states the board's view—with which we are in accord—that "[n]o standard for decision can be derived from this section of the rule" as presently formulated. Thus, the Commission at this time must apply the more general standard of "reasonable assurance" of no "undue risk to the health and safety of the public", as broadly expressed in § 50.35(a).

Because of the importance and novelty of the questions raised by this appeal and the complexity of the material to be considered in reaching our decision, we have devoted considerable time to the evaluation of the voluminous record (over 13,000 pages of testimony and exhibits) which these proceedings have generated. Our evaluation has, of course, taken appropriate account of the findings and conclusions of the board. A board's assessment of record evidence and the conclusions it draws therefrom are entitled to respect on our part. In the present case this respect is necessarily enhanced in consequence of the board's lengthy and direct involvement with the data and expert analyses (much of it conflicting) relating to the critical geological and seismological questions. Our assessment of the record evidence and opinion, within the foregoing framework, leads us to the same conclusion as that of the board in that we are not afforded the assurance required under Section 50.35(a) for a determination that the proposed facility need not be designed to withstand ground displacement from earthquake activity.

As indicated, in support of its position that the probability of ground displacement is so low the facility need not be designed to withstand such displacement, the staff places principal reliance on the record evidence indicating absence of surface faulting at the site for at least 10,000 years and perhaps as long as 180,000 years. This evidence, the staff urges, is the best guide for estimating what might reasonably be expected to occur during the lifetime of the facility.[2]

[2] The maximum period for which a license may be issued under our regulations is 40 years. 10 CFR § 50.51 In the proceedings below, the parties assumed a physical lifetime for the facility of 50 years.

for establishing new Part 100 criteria. The record here—massive though it is—is hardly of the scope required for this task. Nor do we here have access to the variety of views from within our own agency, from industry, from the scientific community and from the public, which are essential to our reaching an informed judgment with respect to the necessary criteria. These are matters which lend themselves to our rule making function.

Our regulatory staff has been devoting efforts toward the development of revised criteria for Commission consideration within the rulemaking framework. We have not reached any conclusion as to the approach new criteria should take, but our experience with this proceeding indicates that such criteria might well cover the need for design against ground displacement from earthquake activity and might also address themselves to the amounts of displacement for which provision must be made.

In a related connection, accelerating research on earthquake prediction and the nature of surface faulting may offer a better opportunity in the future to plan new facilities on a more calculable basis with respect to seismic risk. While hopes in this regard must be tempered by an appreciation of the task, the frontiers of knowledge concerning these matters have been undergoing expansion and efforts are presently underway on several fronts further to increase that knowledge. The Commission will follow developments in this area with great interest.

Since we uphold the determination of the atomic safety and licensing board that the design criteria for the proposed facility must be modified and supplemented to include adequate provision for ground displacement from earthquake activity, we must now consider the propriety of the board's issuance of a provisional construction permit before a determination has been made on the matter of design adequacy in this respect. This course is disputed here by both the staff and the intervenors as not meeting the requirements of 10 CFR §50.35.

Section 50.35(a)(1) of our regulations requires that before a provisional construction permit may be issued, the Commission must find that:

> "* * * the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design * * *."

In view of our earlier determination today as to the need for design against ground displacement to safeguard the integrity of the facility and in view of the relationship of such design accommodation to the basic structure of the facility, we think it clear that design criteria in this respect must be considered to be among "the principal architectural and engineering criteria for the design". Such criteria, therefore, must be appropriately described by the applicant and reviewed in accordance with our procedures before a provisional construction permit may be issued.

It is plain from the record, and the initial decision itself demonstrates, that the requisite criteria have not been submitted for consideration. Moreover, to be meaningful, any such consideration of design adequacy must be preceded by a determination of the amount

A different approach to this matter, indicated in a 1965 article by Allen, St. Amand, Richter and Nordquist,[3] involves correlations between earthquake magnitude and frequency of occurrence. Further study of this method may indeed prove profitable, but as used in the record it would not appear to offer support for the applicant's position that the probability of ground displacement at the proposed site can be disregarded.

As the foregoing discussion would indicate, we are not persuaded by the record that because there has been no surface faulting at the site for 10,000 or more years the probability of ground displacement during the lifetime of the facility can be disregarded in its design. The geology and seismicity of the proposed site bespeak some risk in this regard. The time factor relied upon to demonstrate a sufficiently low risk is not, in our view of the record, adequate for this purpose and does not, we conclude, satisfy the standard of reasonable assurances which underlies Section 50.35(a).

Our determination in the above respect has necessarily taken into account the proposed placement of the facility within Corral Canyon. Particular concern arises from the fact that fault F cuts across the ground upon which the intended reactor would stand. Should the applicant propose a different location for the facility at Corral Canyon, it may well be that the relevant geology and seismicity would still require design for ground displacement from earthquake activity. However, the location of the facility in relationship to faults in the area could have a bearing on the amount of displacement for which design accommodation must be made. These are matters which we cannot pass upon here in the abstract but rather are ones to be considered by the board should a relocation of the facility on the site be put forward by the applicant.

A further comment is in order before we leave this aspect of the case. Both the staff and the applicant have expressed concern that the board, in reaching its determination, has converted our standard of "reasonable assurance" of no "undue risk" into one of assurance of absolute safety. We do not take the board's decision to mean this and, if there be a residue of doubt on this point, we wish to make clear that no such implications should be drawn from our action today. As we have stated in the past, both our statute and implementing regulations show that such an absolute guarantee was never contemplated, and that "the concept of reasonable assurances of safety must be sensibly, though severely applied". (*In the Matter of Power Reactor Development Company*, 1 AEC 65, 73; see also, 1 AEC 128, 147.)

Turning now to a matter of which earlier mention was made, we note that the applicant in its brief to us expresses concern as to the adequacy of the present Part 100 criteria to deal with the problem of reactor siting in seismically active areas and urges further Commission guidance. While these criteria may, in the light of knowledge existing at the time of their issuance, have marked the outer limits for possible guidance in this respect, we agree that Part 100 calls for reexamination and elaboration in the light of present-day needs and knowledge. This ajudicatory proceeding is not, of course, the appropriate forum

[3] *Relationship Between Seismicity and Geolcgic Structure in the Southern California Region*, Bulletin of the Seismological Society of America, Vol. 55, pp. 753–797 (August, 1965).

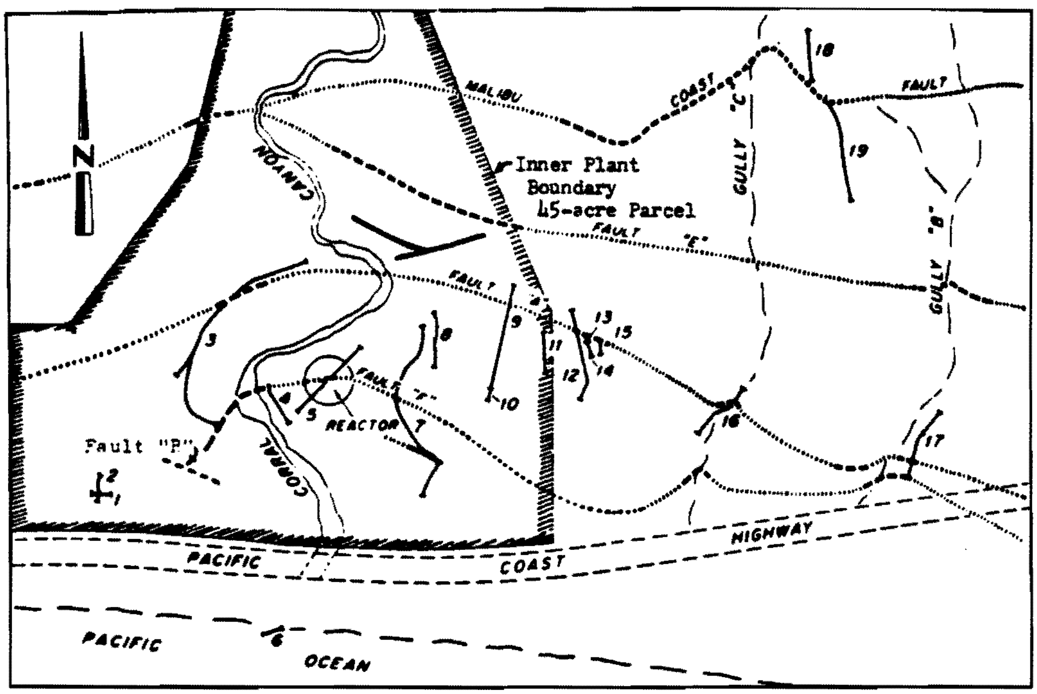
3. The matter be remanded to the atomic safety and licensing board where, upon submission by the applicant of appropriate amendment to its application and after review by the regulatory staff and the Advisory Committee on Reactor Safeguards, there be further proceedings pursuant to Parts 2 and 50 of our regulations to determine:

(a) The amount of differential ground displacement which the facility must be designed to withstand; and

(b) The adequacy of the design criteria which may be proposed by the applicant to permit the facility to accommodate the amount of differential ground displacement determined in accordance with (a), above.

By the Commission,
W. B. McCool, *Secretary.*

Appendix "A"



Map of a Portion of the Proposed Site Showing Faults Mentioned in the Text and Trenches Dug by the Department, Together With the Approximate Location of a Portion of the Inner Plant Site and of the Reactor Containment Building

This map is a reproduction of Figure 1d from the Department's rebuttal testimony of October 19, 1965, as supplemented by Staff Exhibit 14, which latter portrays the inferred extension and connections of Fault F to the east. The map also shows Fault B, which is believed by the Department's geologists to truncate Fault F, and if it does so, may interrupt the latter's extension westward to Solstice Canyon. A landslide between the reactor site and Solstice Canyon prevents certain identification of Fault F westward.

Fault X is not shown on the map because its exposure was too short and its extension difficult to infer. A number of smaller faults with displacements of a few feet are recognized throughout the plant site but are not projected on this map. A solid line apparently to designate a fault, which, however, was not identiof ground displacement the proposed facility need be designed to withstand—and the record is clearly not sufficient for this purpose. Accordingly, further proceedings to deal with these matters will be necessary.

While our holding on this aspect of the initial decision fixes a different procedural course than that deemed appropriate by the board, it does not signify any lessening of Commission reliance on atomic safety and licensing boards for effecting appropriate resolution of safety issues in proceedings of this type; nor does it mark a curtailment of the procedural latitude which we have left to boards for this purpose. (See, e.g., *Matter of Jersey Central Power & Light Company*, 3 AEC 28, May 6, 1965.) That reliance, as indicated in the discussion attendant our earlier holding, and that latitude, continue. The scope accorded to boards, however, must be exercised within the limits defined by our regulations. The descriptive requirement of §50.35 (a)(1)—specifying a cardinal element for consideration at the construction permit stage—is such a limit and, accordingly, must be observed here.

Because of the further proceedings which our holding today makes necessary, some comment is here warranted with respect to determining the character and amount of ground displacement the proposed facility must be designed to withstand and the adequacy of design criteria which may be proposed in that regard. The initial decision and the presentations to us by the parties have used the phrase "permanent ground displacement" to characterize the rupture of the ground which may result from earthquake activity. We believe, however, that the matters yet to be considered with respect to design accommodation are better described in terms of "differential ground displacement", which more aptly denotes that the relative movement of the two sides of a fault (horizontal, vertical, or both) is of significance. Accordingly, in our Order we have used the term "differential ground displacement" to describe the matters to be taken up in the proceedings to follow. We would expect, of course, that any factors which properly bear on the matter of design accommodation receive appropriate attention.

It is anticipated that at some time in the future, determinations as to the amount of differential ground displacement a facility must be designed to withstand will be aided by more specific Commission criteria of the type to which we made reference in connection with possible revision of Part 100. The proceedings to date on the instant application have not dealt with this determination, so we cannot assess whether the present lack of such criteria will give rise to difficulties in the further proceedings which are required. This being the case, and because we do not wish to impose unnecessary delay in a matter which has already consumed much time, we leave it to the parties to go forward under our regulations as they presently exist. In this connection, it may be that information generated in the course of developing revised criteria will be of assistance to the parties and board.

It is therefore ORDERED that:

1. Exception "A" filed by the regulatory staff be denied;
2. The order of the atomic safety and licensing board authorizing issuance of a provisional construction permit be set aside; and

5.5 SEISMIC SAFETY - LATE 1963 to 1977

In the second half of 1963, the San Onofre Unit 1 reactor on the Camp Pendelton Marine Corps Base near San Clemente, California was reviewed for a construction permit. At the 48th ACRS meeting, July 11-13, 1963, representatives of Southern California Edison described their planned approach to seismic design in terms of an acceleration of about 0.2g as shown in the excerpt from the meeting minutes.

> <u>July 11-13, 1963 - 48th ACRS Meeting</u>
>
> The earthquake characteristics of the area were discussed in detail. There was a map showing the location of faults near the plant. The Newport-Inglewood fault is to the north. Another fault goes through San Diego to the south and there is an Elsinore fault inland to the northeast; one eastern seismologist attempted to connect these two known coastal faults, which would bring the projected fault to 10 to 12 miles off shore from the site. However, there seems to be little evidence supporting this extension. There are no measurable strains or surface ruptures in the nearest fault.
>
> There were comments on the consultation from seismologists engaged; one is a non-Californian. Hindsight is the only basis for earthquake predictions. The San Diego Gas & Electric Company has built conventional power plants south of the Camp Pendleton site using .2 of the acceleration of gravity (g) for an earthquake safety factor, and SCE has built a number of plants north of the site using the same factor. At the present time, the plan is to design the nuclear plant for forces of about 0.2 the acceleration of gravity. There were comments on the earthquake damage to a steam plant operating in the 1933 Long Beach earthquake; this experience formed the basis for much of the later design precautions. The El Centro earthquake intensity was .33 g in one direction; there were seismographs at San Diego, Santa Ana, and three at Long Beach to record this intensity. The El Centro quake area was over water soaked sediments which is believed to have aggravated the intensities felt. It is not believed that forces of .33 g would exceed the yield strength of the nuclear plant.
>
> For an earthquake analysis, or so-called ground response spectrum, the experts assign intensities expected in different rock types, which are mainly rigid materials in the area; this is then related to the center of the earthquake to predict the forces on the plant. Such studies indicate forces corresponding to .07 g are expected in the underlying sandstone material with a value of .15 g in the overlying earth where structures such as switchyards might be anchored; this is the basis for .2 g figure. There was further information on the method of earthquake analysis as related to typical quakes, kinds of earth, and the geographic location which determine the natural frequencies expected.
>
> Southern California Edison does not feel that a nuclear plant need be designed with any more attention toward earthquakes than a conventional steam plant with the exception of the critical sections, e.g., safety injection systems. Mr. Gould said the design would consider the .2 g factor as a minimum with a

fied by any alphabetical designation, is located above Fault A, just easterly of Corral Canyon and within the plant site.

The trenches dug by the Department are shown by numbered designations. The No. 5 trench was dug after the April 1965 recess of the hearings.

The certainty of location of faults is roughly indicated by the continuity of the lines representing them.

APPENDIX "B"

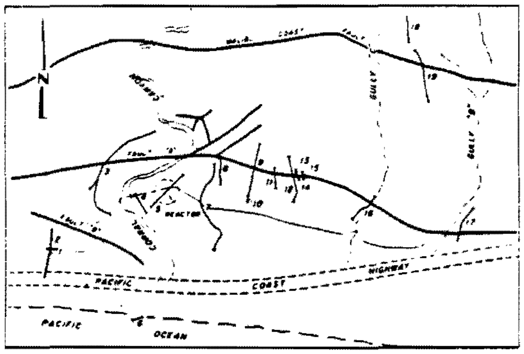


Fig. 1a. Smith-Hoffman
(For the Department)

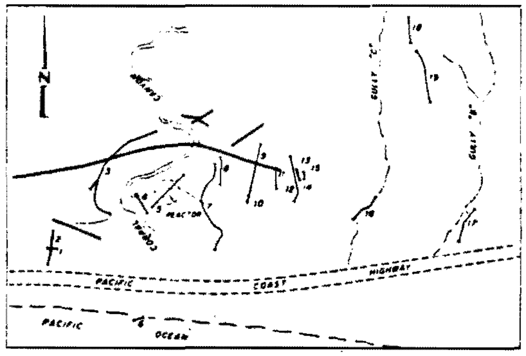


Fig. 1b. Jahns
(For the Department)

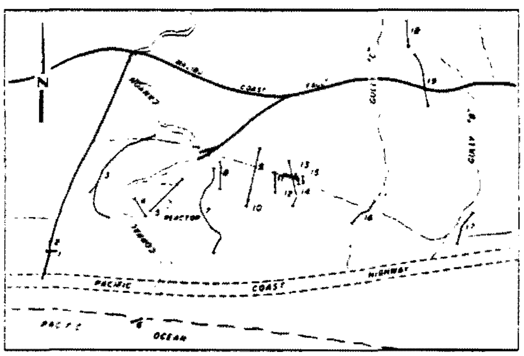


Fig. 1c. Cleveland-Troxel
(For the State of California)

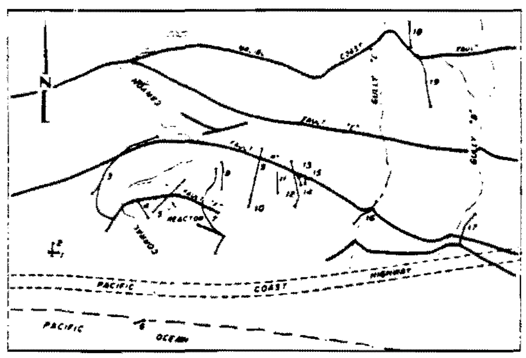


Fig. 1d. Yerkes-Wentworth
(Of the U. S. Geological Survey for the AEC Regulatory Staff)

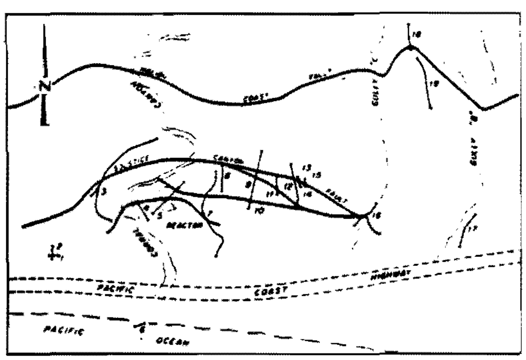


Fig. 1e. Kamb
(For Marblehead Land Company, Intervenor)

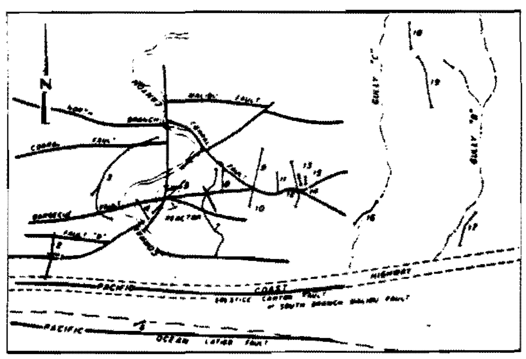


Fig. 1f. Bailey
(For Marblehead Land Company, Intervenor)

EXPLANATION

2 *Exploratory trench (Smith-Hoffman)*

Fault, minor faults in Trench 5 not shown

Contact between mudstone and mudstone with sandstone (Figs. 1a & 1b)

Contact between siliceous shale & mudstone with sandstone (Figs. 1c)

SCALE: 1" = 800'

0 400 800 1200 1600'

Figure 1. **Index map of trenches and pertinent geologic features.**

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

February 25, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: SEISMIC DESIGN OF NUCLEAR POWER PLANTS

Dear Dr. Seaborg:

The question of appropriate seismic design of nuclear power plants has recently occupied the attention of the Advisory Committee on Reactor Safeguards. Most of this study has been called for because of proposals to construct nuclear power plants in seismically active zones. There have, however, also been questions relative to plants to be built in regions where seismic activity is low or where the seismic history is based on uncertain documentation of earth movement that took place long ago.

The Committee believes that, especially in cases of the latter kind, uncertainties in the geology or the seismicity may cause a trend toward choosing design methods or design parameters with more conservatism than is necessary. Adequate seismic design guides that provide protection comparable with that provided by other safety features of nuclear plants are needed.

Before seismic design guides or parameters are adopted for zones of relative seismic quiet, the Committee suggests further consideration of the question of earthquake probability and magnitude for these zones, and of the question of adequate engineering for small earth movements.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

However, by the 49th meeting, September 5-6, 1963, the seismic design approach was more specific and more conservative. Southern California Edison stated that the plant would be designed for a safe shutdown earthquake of 0.5g. During this same time period, the proposed seismic design basis for Bodega Bay was also being increased from 0.3g to about double that figure for safe shutdown.

In late 1963 - early 1964, the question of seismic design basis for an eastern reactor (Connecticut Yankee) introduced some relatively modest controversy. At the ACRS Subcommittee meeting held December 13, 1963, Mr. L. E. Minnick, speaking for the applicant, pointed out that earthquakes are generally not considered in New England. There had been minor earthquakes in Connecticut in the last 150 years, and the site was stated to lie on the boundary between seismic zones 1 and 2. He, therefore, suggested that a 0.1g design basis for safe shutdown would be appropriate.

The Regulatory Staff obtained the advice of the U. S. Coast and Geodetic Survey (USC&GS) who recommended that a larger peak acceleration be used for the SSE, namely 0.17g. This contrasted to the recommendation in a letter sent by Father John J. Lynch, a consultant to the applicant, who stated that 0.03g was a suitable safety factor for the design.

Mr. Roger Coe agreed to the design basis recommended by the Regulatory Staff on behalf of the applicant, but complained about the extensive and costly engineering necessary to study seismic safety for the staff's recommended value. The Committee did not disagree with the Staff in its report of February 19, 1964 on Connecticut Yankee. However, at least some of the ACRS members appeared to question the need for as high an acceleration as proposed by the Staff. And, at the Special Meeting held on February 24, 1964, the Committee wrote a letter suggesting this point of view. The letter is reproduced on the following page.

The ACRS also requested advice from an independent consultant concerning a suitable seismic design basis for the Haddam Neck (Connecticut Yankee) site.

The consultant, Dr. Perry Byerly, in a letter dated August 12, 1964, stated "the design factor of 0.17g is reasonable and just. I would not accept a lower factor."

Thus, Dr. Byerly supported the recommendation of the USC&GS, and differed widely from Father Lynch. Connecticut Yankee had introduced a wide variation in expert opinion on seismic events for the eastern U. S.; and it brought the need for increased attention to seismic design for eastern sites into focus.

In May, 1964, Dr. R. A. Williamson prepared a report for the Regulatory Staff in response to ACRS interest in two questions:

1) To what extent do seismic considerations influence reactor costs?

Draft #1
RFF:vb
10/27/64

MEETING OF
SEISMIC DESIGN CRITERIA SUBCOMMITTEE
Washington, D. C.
September 28, 1964

This meeting was called to determine what course of action should be followed in establishing seismic design criteria for nuclear power plants.

Attendees:

ACRS

C. R. Williams, Subcommittee Chairman
W. D. Manly
D. A. Rogers
R. C. Stratton
K. V. Steinbrugge, Consultant
P. Byerly, Consultant
W. G. Van Dorn, Consultant
R. F. Fraley, Staff

Division of Reactor Licensing

C. K. Beck
E. G. Case
S. Levine
J. F. Newell
L. Kornblith
R. R. Maccary
R. H. Bryan
D. Knuth
R. A. Williamson, Consultant
(Holmes & Narver)

General Counsel

G. F. Hadlock

USGS

A. Clebsch

USC&GS

B. D. Zetler
A. Miller
R. J. Brazee

Executive Session - Dr. Williams opened the meeting by expressing concern over the policy of DRL to accept standards recommended by other government agencies without any substantive technical backup. He also expressed concern over the fact that many reactors in the past have been approved with little review of seismic effects and that points which get major emphasis by DRL appear to be determined by the intervenors.

2) How much seismic resistance is available in facilities not designed to withstand seismic forces?

Dr. Williamson estimated that seismic engineering might add 20 to 50% to the engineering costs but only 2 to 4% of the total construction costs. He also estimated that the effects of seismic design on actual construction costs would be very small, except for a reactor like Bodega Bay, where there might be an increase of 5 to 15%. He advised that containment vessels would have a considerable inherent degree of seismic resistance, but that equipment items and related systems tend to be vulnerable and are not normally supported for seismic forces. Dr. Williamson concluded,

> It is strongly recommended that no projected power reactor facilities for location in areas currently regarded as seismically inactive be designed without complying with certain minimum standards for earthquake resistance.

On September 28, 1964, the ACRS held a meeting of a newly formed Seismic Design Criteria Subcommittee, the complete minutes of which are reproduced on the following pages.

In retrospect, this was a very important meeting. Not only was there a general concensus on the need for seismic design of reactors and the establishment of seismic criteria or guidelines; it was a beginning of the development of criteria for all natural phenomena such as floods, wind-waves (seiches), etc. The minutes record some rather different, possible, philosophic approaches among the participants with regard to the establishment of appropriate seismic design bases.

The Regulatory Staff initiated work with its consultants to develop more specific seismic engineering criteria in 1965. And in a letter to Chairman Seaborg dated March 31, 1965, the ACRS recommended the formation of a specially qualified panel to recommend a long range research program on seismic phenomena. In that letter the ACRS also recommended that all power reactors be equipped with strong motion monitoring instrumentation.

In a letter dated May 22, 1967, Edson Case, then Director of Reactor Standards in the AEC Regulatory Staff, forwarded to the ACRS a draft document entitled "Seismic and Geologic Siting Criteria for Nuclear Power Plants", and requested ACRS review and comment, initially via an appropriate subcommittee. To assist it in accomplishing this task, as well as to enable an independent review of sites having difficult seismic questions (such as the proposed Bolsa Island site in California), the ACRS obtained the services of several well qualified consultants in seismology and geology.

The draft criteria (dated May 11, 1967) represented a considerable effort by the Regulatory Staff and their consultants, and provided a very good starting basis for the development of criteria. The criteria included a minimum design basis earthquake (or floor) of 0.1g. The seismic engineering criteria were not included in the draft and were stated to be under development.

3. **Provisions must be adequate to maintain containment.**

Mr. Case agreed that 2 and 3 are redundant but indicated that he considered this a necessary and reasonable conservatism.

Criteria making use of appropriate intensity zones was discussed. Mr. Williamson cautioned that the establishment of appropriate zones will be difficult since neither Richter zone maps or Uniform Building Code (UBC) maps are necessarily applicable. The UBC maps are based primarily on the worst recorded event but are decreased with time if no repetitions occur. Mr. Case noted that a zone concept appears adequate for general guidance but that a "finer" structure or method of evaluation is needed to settle on a specific site. Dr. Byerly again recommended use of the maximum recorded event at a specific site as the basis for design. Dr. Beck noted that this presents a problem at locations where there is no recorded history. Since records concentrate on loss of life and money, bad shocks in remote areas often are unrecorded. In addition, there have occasionally been "bad" shocks in areas of previously recorded low intensity.

Mr. Brazee explained the USC&GS rationale behind their recommended figures for a specific site. The maximum quake of record on the appropriate fault is translated to the location of the fault closest to the site. The intensity of shaking at the site from this quake at this location is then predicted. A vibration spectrum assuming rock as the foundation material is used. Mr. Brazee noted that engineering judgment is used in selecting the appropriate fault and the maximum quake of record (e.g., unconfirmed reports are discounted). The assumption that future quakes can be predicted by the maximum recorded is in itself an assumption since they might in fact be larger. Mr. Steinbrugge and Dr. Byerly stated that too much emphasis is being placed on which faults are active and which are not since this is difficult to predict. Dr. Van Dorn noted that new faults frequently develop from most major quakes. Often they occur as far as 10 to 100 miles from causative faults. He suggested that the 1/4 mile criteria is really meaningless. This introduced the possiblity that design for differential fault movement as well as ground shaking might be required.

It appeared generally agreed that the definition of a "fault" must be improved, however, since many small cracks which could be called "faults" are of small consequence. Dr. Beck noted that the USGS is doing a study of fault systems in the United States.

Col. Stratton proposed that the time aspect should be considered in establishing criteria since the life of a plant is small compared to the interval between major disturbances.

A brief period was devoted to a discussion directed toward the development of a better appreciation of the competence of the 40 seismologists, geologists, seismic engineers, etc. who have been involved recently in several reactor cases.

The need for design standards was then discussed. Mr. Steinbrugge recommended that a set of seismic design standards be developed for reactors as soon as possible. This would provide guidance badly needed by applicants and would establish a basis for consistent and comprehensive evaluation of plant design. Mr. Steinbrugge expressed concern over the fact that many of the studies and designs today consider only major systems and components. "Smaller" systems are constructed by conventional practice (e.g., piping runs laid out by pipefitters rather than designers) with no appreciation of their importance to plant safety.

Dr. Byerly noted that the AEC must decide if it is willing to gamble (probability approach) or take into account the worst disturbance which could occur. He recommended use of the worst recorded event in an area as the basis for design. He noted that Richters Seismic maps are based on the geologic foundation and therefore predict what might happen in an area rather than what has happened.

Mr. Steinbrugge noted that many deficiencies in plant earthquake resistance result from a lack of coordination between the various professions involved (e.g., structural vs systems vs electrical engineers). The necessary coordination to provide adequate design would add little to plant costs. It was generally agreed, however, that cost is not an important factor to be considered where safety is involved. Mr. Rogers noted that limits for safety should be adequate but not "impossible" however.

Dr. Williams inquired if there is a minimum level of activity where seismic considerations can be ignored. It was generally agreed that there is no level at which a design review for seismic adequacy can be neglected (e.g., minimum design criteria should be developed for all reactors). This review may, of course, indicate that no special provisions are required in a specific plant at a specific site.

Meeting with Regulatory Staff - E. Case reported on the internal guides used by DRL in the evaluation of seismic design. These are basically that:

1. A plant is not required to operate normally during and after a disturbance.

2. Provisions must be adequate for shutting down the reactor and removing decay heat.

2. All plants west of the Rockies should have a strong motion seismograph installed.

3. Damage studies and investigations should be supported by the AEC.

W. G. Van Dorn: A 50 foot tsunami is arbitrarily large but a smaller wave cannot be defended as a general, worst case wave. Smaller waves can be accepted on a case basis.

P. Byerly: Seismic parameters should be based on the worst recorded event.

W. D. Manly: The possible generation of new faults by seismic disturbances is a question which appears to need resolution.

The following general conclusions were reached:

Nominal guidelines would be very helpful and would reduce delay, uncertainty and paperwork involved in review of reactor proposals. A detailed review would be required to adjust the nominal values for each specific site. Development of these guidelines should begin as soon as possible.

* * *

There was a brief discussion of the situation for plants already built and those under construction. E. Case reported that design parameters for plants reviewed recently (e.g., San Onofre, Connecticut-Yankee) are considered adequate. The design of plants approved some time ago (e.g., Humboldt Bay, Yankee) will require review. Dr. Bryan indicated that a set of criteria would be very helpful in a review of plants already built.

The question of tsunamis was then discussed. It was suggested that DRL should also consider, in the development of criteria, other natural phenomena such as forest fires, wind waves, land slides, floods, etc.

Mr. Zetler, in response to a question from Dr. Williams, stated that tsunamis are a problem on the west coast but not on the east coast because of the larger width of the continental shelf on the east coast. Storm driven waves on the east and Gulf coast of 14 to 15 feet must be considered, however. Storm waves on the west coast would be on the order of 2 feet. He reviewed the record of tsunamis on the west coast and noted that, in general, they are on the order of 10 to 15 feet. There is a report of a 50 foot wave near Los Angeles in 1812 but this figure has not been confirmed. In 1946, however, a lighthouse with a base of 50 feet above sea level was washed away by a tsunami. Mr. Zetler noted that most tsunamis recorded on the east coast have been from distant earthquakes and that those resulting from nearby earth movement can be much higher (50 feet). Dr. Van Dorn stated that tsunami damage is often very local (e.g., at Hilo, Hawaii a distance of 60 feet along the beach made the difference between no damage and complete destruction). Prediction of the exact point at which a high point will occur is difficult. Dr. Van Dorn cautioned that a local fault movement will generate a wave only if a vertical displacement occurs. He has suggested additional study of the San Andreas fault to determine if it can, in fact, act as a tsunami generator, since movement along this fault is primarily horizontal. He also suggested that at some sites, (e.g., wide, flat, level plains) the predicted height can, even now, be reduced considerably (e.g., to a height of 10 feet) below the nominal 50 foot value.

Executive Session:

Various participants made specific comments which are noted below:

D. A. Rogers: Maximum quake requirements should be used and specified from the start. It's the delays and stepwise development of criteria which cost the applicant money.

K. V. Steinbrugge:

1. The start of guidelines is mandatory. Applicants would be much "happier" with specified guidelines, even if severe at first.

One important potential difference of opinion between the ACRS and the Regulatory Staff was discussed by the Committee at its 94th meeting, February 8-10, 1968. It was reported in the Executive Session that all the consultants to the Seismic Subcommittee had expressed concern over whether the approach being taken to earthquake acceleration (SSE) east of the Rocky Mountains was sufficiently conservative. It was reported that Mr. Case of the Regulatory Staff had agreed to examine the possibility of defining a higher, base-level (or minimum) acceleration east of the Rockies.

In discussions with the Regulatory Staff, Mr. Case was asked if he had a basis for comparing the cost of construction employing a minimum value of 0.2g compared to a minimum of 0.1g. He replied he did not know how to make a comparison (although Mr. Williamson had responded to this question in a memorandum to Mr. Case some years earlier and estimated that only small differences in cost would ordinarily occur, and later estimates tended to confirm Williamson's early report).

The matter of what "floor" on seismic design should be used remained an open question for a long time; it was eventually resolved by accepting the original Staff position of 0.1g, without benefit of a comprehensive study of the matter.

It was in November, 1971, after very many major redrafts, that the Atomic Energy Commission fianlly issued a Notice of Proposed Rule-Making to amend the Commission's regulations, 10 CFR Part 100, to add a new appendix, Appendix A "Seismic and Geologic Siting Criteria for Nuclear Power Plants." The proposed criteria reflected the practice which had been followed in actual construction permit reviews and were reasonably specific in their definition of a "capable" fault. They also gave guidance as to the general extent of the geologic and seismic investigation required. However, they did not provide a quantitative criterion for establishment of the design basis (or safe shutdown) earthquake, and as time went on, the wording used proved to provide great flexibility, i.e., required the exercise of judgment wherein seismic experts could and would continue to differ greatly.

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effect the Bolsa Island project. In summary, Mr. Price reported that he was still awaiting DRD&T comments on the draft seismic criteria dated May 6, 1968. Mr. Price said that he was trying to arrange a meeting with Mr. Shaw to attempt to resolve the DRD&T comments. Mr. Price further stated that he was prepared to submit the draft criteria to the Commission with or without DRD&T input. It was noted that DRD&T may have a conflict of interest between the seismic design criteria and tentative RD&T comments were forwarded to ACRS members on June 10, 1968.

Seismic Design Criteria - Mr. Price reported that DRS had finally been able to meet with DRD&T and that the January 8, 1969 draft of the "Seismic and Geologic Siting and Design Criteria" reflected the resolution of the DRD&T comments. The Committee was advised that the Regulatory Staff would like to schedule a

One of the most controversial aspects of the first draft related to a curve, based on a report under preparation by M.G. Bonilla, of the U. S. Geological Survey, which related surface displacement to magnitude of earthquake on a surface fault. Also controversial was the choice of the minimum distance from the center line of a fault beyond which a reactor might be located. Another item which received much scrutiny and comment was the definition of a "capable fault," meaning a fault having surface expression and which was deemed capable of exhibiting permanent relative displacement on the two sides of the fault as the result of an earthquake.

A very large number of meetings were held between the ACRS and the Regulatory Staff, and many revised drafts were prepared. In the latter half of the 1960's, the Regulatory Staff seemed to have the requirement that it obtain comment and preferably concurrence from the AEC Division of Reactor Development and Technology on such criteria, although the latter represented the AEC "promotional side."* And, during the period when the Bolsa Island project was active, the seismic criteria were held in a state of abeyance by Mr. Price while their potential impact on Bolsa Island (a project important to the AEC) was assessed.

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*See following excerpt from the 98th meeting, June 5-8, 1968, and the 105th meeting, January 1969.

98th Meeting, June 5-8, 1968 - Seismic Design Criteria

Dr. Okrent inquired into the status of the seismic design criteria. Mr. Price reported that a draft of the proposed criteria had been sent to DRD&T. A copy of the draft had been reviewed with Dr. Lieberman, and then it had been sent to Mr. Shaw about six weeks ago. Mr. Price said that he did not want any more unmanageable comments. He stated that a couple of days prior to this meeting he received a draft memo and comments, however the date established for that meeting was the day prior to this meeting, however it did not materialize. Mr. Price said that he expected such a meeting to be held within the next week or so. He said his present mood was to submit a draft seismic design criteria to the Commissioners with or without Mr. Shaw's input, however he said he was trying to reach agreement with DRD&T so as to approach the Commission without conflict.

Dr. Okrent noted that even if Mr. Shaw didn't have a conflict of interest, that DRD&T should not control the regulation of criteria. Dr. Okrent said that he would like a copy of the draft that Mr. Shaw is looking at now. He suggested that if there is a serious disagreement between the Division of Regulation and RDT that the disagreement should be taken to the Commission. He pointed out that it was a Committée privilege to recommend criteria. Mr. Price said that he would like to know the areas of disagreement between Regulation and RD&T, and he would be pleased to provide Dr. Okrent with a copy of the present draft. He pointed out that there seemed to be some technical difficulties between Regulation and DRD&T, and he is not sure how this criteria would

Project: Virgil Summer

Status: C.P. review, geology-seismicity, letter requested

Chronology: 9/8/72 - Site visit, Subcommittee meeting
9/14/72 - ACRS review
10/6/72 - Subcommittee meeting, geology-seismicity

Background:

Immediately prior to the September ACRS meeting, a question was brought up regarding the adequacy of the 0.15g DBE for the Summer site. The question was based on the proximity of this site to Charleston (125 miles) and on a recent paper by Bollinger, VPI, from which some had concluded that he felt that "structure" extended from Charleston to the northwest and through the area which includes the Summer site. The inference here is that perhaps one might have to assume that the 1886 Charleston event (Int. XI) could occur at the site. A special Subcommittee meeting was held on October 6, 1972, with ACRS consultants present, to review this aspect of the project.

Results of October 6 Meeting:

The Regulatory Staff is satisfied with the 0.15g. They feel that this is consistent with values being applied to other sites, and that to require moving the Charleston event violates their seismic siting criteria. The Staff's consultants from USGS and NOAA, and the applicant feel that the event was a local one and that there is no justification for moving it away from Charleston.

The ACRS consultants are not so optimistic, however, and are unanimous in their feeling that a research or investigation program should be initiated to help explain the cause of the event, and therefore give more confidence in the conclusion that what happened in 1886 cannot occur elsewhere. The problem is that 2 years might pass before any real conclusion can be drawn, and what does one do about nuclear plant licensing in the interim? (The plants in this general area include Farley, Hatch, Oconee, Robinson, McGuire, Brunswick, Harris, etc., and could include those all the way to N. Jersey). Some possibilities were explored by the Subcommittee, and included: stop licensing plants on the east coast for 2 years, increase the DBE for Summer while the exploration program is in progress, tell the Summer applicant that they are proceeding at their own risk, assume the Charleston earthquake can occur anywhere.

Meanwhile, there is no obvious source of funding for an investigation program though there was a suggestion that the AEC might fund one. A

In 1972, seismic design basis again became a matter of some controversy in connection with review of the Virgil Summer plant. An ACRS Staff engineer prepared the following status report prior to the Special ACRS meeting, October 26-28,1972, which is reproduced on the following pages together with an excerpt from the meeting minutes (which summarizes ACRS review action on the matter) and excerpts from the ACRS report on Summer.

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conference with industry representatives to review these criteria as soon as possible and possibly before the ACRS review was complete. Members of the Committee expressed a concern about requesting industry comments on the proposed criteria before the ACRS has made its comments.

Subsequent to this meeting a Subcommittee meeting was scheduled to review the changes made as a result of the DRD&T comments.

Seismic Engineering Design Criteria - Mr. Case reported that a document was being developed that would set forth the engineering that would be required for the design of a plant that would withstand the accelerations and displacement specified in the site criteria. He hopes to have comments of the most recent draft from his consultants in the next several months.

ꓤGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 (C.P.)

Executive Session

Dean Palladino reviewed the history and design of the Summer plant. It is a three loop, 2775 MWt Westinghouse reactor. The project was initially considered at the September, 1972 ACRS meeting at which most of the outstanding issues were discussed. The primary issue for consideration at this meeting was the zero frequency ground acceleration for the safe shutdown earthquake (SSE). The SSE was put in question based in part on a paper by Dr. G. A. Bollinger that suggested the earthquakes in the Charleston area might lie on a NW trending line that could extend near the Summer site. The ACRS called in its seismic consultants and the Subcommittee met with them and the applicant on October 6, 1972. The general feeling as a result of that meeting was that it would be well to have the same margin of conservatism in seismic design throughout the U.S. and that further information was needed to decide if the Charleston earthquake of 1886 should be moved closer to the site. There is nothing particularly alarming about the Summer site but the consultants felt there was little evidence to prove that the Charleston earthquake could not occur at places other than at Charleston, including locations nearer the site.

One problem is that the Regulatory Staff is convinced that the Charleston quakes are a local phenomena and the seismic design value of 0.15g is adequate. The Committee's consultants believed that additional work needs to be done to explain the Charleston events and suggested values for the site ranging from 0.2 to 0.3g with a tendency toward 0.2g.

The applicant was asked to evaluate the changes in plant design that would result from adoption of 0.2g.

The Committee's consultants, Page, Philbrick, and Wilson, were present and Dr. Allen had sent in his written comments. Dr. Philbrick thought the Charleston earthquake could be moved anyplace on the eastern face of the Appalachian mountains and as far north as New Jersey. Dr. Wilson estimated the ground acceleration at Charleston at 0.4-0.5g. If it had occurred in California it would have been on a fault about 100 km long.

$\$2\text{-}3 \times 10^6$ program seemed to be the scope visualized by one of the ACRS consultants.

For the October Special Meeting:

The applicant has been asked to be present to discuss the site seismicity. He was told that the Subcommittee was leaning toward some increase in g value, and that he should be ready to discuss how this affects the plant. The economic breakover point was of particular interest. Other areas in which there might be questions were identified. They included: ECCS, in-core monitoring, pipe whip, Class I dam design. The applicant is also to bring certain information requested by Dr. Isbin.

* * * * *

The Staff consultants studied the earth's magnetic field and gravity field in the vicinity of the site. They also looked at geothermal manifestations (hot springs) to try to establish some correlation with the geology and seismology of the region. The basement rock in the Charleston area is overlain with several thousand feet of coastal sediments. Seismic profiles show what may be a fault south of Charleston and there is a sharp magnetic anomaly in that region. Seismic refraction studies suggest a ridge in the basement rock (called Yamacraw Ridge) in the vicinity of Charleston. The northern extension of this ridge appears to end about Charleston while the southern extension is not clearly defined.

Mr. Devine claimed that, even without the 1886 earthquake, other evidence would indicate the Charleston area as one of high seismic activity.

In response to a question by Dr. Okrent, Mr. Devine said that, in the presence of a fault 60,000,000 years old with magnetic anomalies and some seismic activity, one would have to consider a quake as severe as the 1886 Charleston event. It was also noted that there is no place in the U.S. where the absence of earthquakes can be guaranteed.

No one had a good explanation of a geologic structure that would be essentially a point source for earthquakes.

In response to Dean Palladino's question, Mr. DeYoung (L) stated that the applicant had studied the full range of break sizes in his ECCS analysis.

Mr. DeYoung also stated that it was the Staff's intent to require fixed incore neutron detectors in cores where the flux peaking factors are required to be below 2.5. The applicant claims the peaking factor in Summer is 2.4 so, although no fixed incores are being required, the Staff will require added surveillance. The applicant is retaining some flexibility to install fixed incore monitors if that becomes necessary.

Mr. DeYoung said that he expected the applicant to come in with a completely new core design by the time of the operating license review. He recommended a meeting of the Westinghouse Subcommittee to discuss the new core design. (A Westinghouse Subcommittee meeting has been scheduled for November 18, 1972).

Since there was no surface rupture, the deep fault would be about 40 km long and would produce a shock with a higher frequency than those observed in California.

Dr. Siess told the Committee that the Staff's NOAA consultants would show a map plotting the frequency of occurrence of earthquakes of greater than a given magnitude in the eastern U.S. It will show a bullseye around the Charleston area. Dr. Wilson tended to question the value of such a plot and said that in California, where the frequency of earthquakes is much higher, if one made a similar plot for a 50 year record one would get similar bullseyes but if one took data from a 200 year record the bullseyes would be smeared out somewhat.

The Committee and its consultants discussed the value and delay of a more detailed seismic and geological study by the applicant. It was decided that this applicant could do little more on his own that would be useful on a reasonable time scale. Dr. Wilson pointed out that there is a finite possibility that the Charleston quake is tied to a NW trending geologic structure and not until the applicant can come in with an extremely strong case can the possibility be ruled out. The probability of the Charleston quake moving to the NE is greater than of its movement to the NW. There is also a possibility that it won't move at all. The Regulatory Staff and its consultants believe that it won't move.

Dr. Siess noted that the seismic design basis proposed by this applicant provides the same degree of conservatism as exists at other plants in this region.

MEETING WITH THE REGULATORY STAFF

Mr. Devine (NOAA) presented the Staff's case for considering the Charleston earthquakes as local phenomena. He showed a plot of eastern U.S. earthquakes for which there is high confidence in their occurrence. He pointed out some seismic patterns that include a NE trend with specific concentration areas and some lineations. He had prepared a contour map using seismic events with an Intensity of V and above. The contours were based on a geometrical progression (2, 4, 8, 16, 32, 64) of frequency of occurrence per 10,000 sq km. He noted that there has been a strong earthquake near the center of each of the 64 contours and not outside of the closed 64 contour. Mr. Coulter (USGS) confirmed that there had not been a major earthquake on a totally new structure. Some have occurred on previously unrecognized structures however.

The applicant reviewed the status of research and development being done in support of this application (See Section 1.8 of the PSAR). The applicant is monitoring Westinghouse's tests in support of the ECCS analysis. The applicant doesn't plan to perform any tests himself and doesn't believe any R&D beyond the Westinghouse effort is required.

The applicant has been exploring with Westinghouse a system for detecting vibrations or loose parts in the primary system. Westinghouse claims there is not yet a system to do this. The applicant is pursuing the matter with other manufacturers and is keeping up with the technology. If such a system is developed in the next five years the applicant will look at it to see if he would want to install it at Summer.

The applicant described his three leak detection systems; each of which will be able to detect leaks as small as 1 gpm.

Caucus with ACRS Consultants

Dr. Page stated that he was aware of the Savannah River drilling data and he thought that it ruled out the possibility of a great fault running NW-SE. He doubted the existance of a NW trending fault.

Dr. Wilson did not feel that the relatively shallow geology (several thousand feet) was significant for the deep earthquakes that occur in this area.

All three consultants present thought that a ground acceleration of 0.2g would provide about the same level of conservatism at this site as was finally obtained for San Onofre Units 2 and 3. Neither Page nor Wilson were particularly distressed at the prospect of the Committee accepting a design value of 0.15g but Dr. Wilson was afraid that it would become a precedent for future plants. Dr. Philbrick, however, thought that 0.2g was at the low end of acceptable values.

Everyone agreed that more information about the Charleston earthquakes would be very desirable.

MEETING WITH SOUTH CAROLINA ELECTRIC AND GAS COMPANY

Mr. Fischer presented the applicants analysis of the basis for the proposed seismic design values. He pointed out that the site is located in a different geologic province (Piedmont) than is Charleston. The applicant took the largest earthquake that had occurred in the Piedmont province (Intensity VII) and transferred it to the site. He also considered the largest Charleston quake (at Charleston) and attenuated it to obtain accelerations at the site. On this basis he had proposed a design value of 0.12g but the Regulatory Staff had asked for more and they finally agreed on a value of 0.15g.

Because of the questions raised by Dr. Bollinger's paper, the applicant wrote a letter inquiring into his studies. Dr. Bollinger replied and a copy of his letter is available in the ACRS files. Dr. Bollinger said that the nature of the fault system in the South Carolina-Georgia Seismic Zone is unclear. Mr. Fischer noted that Dr. Bollinger's paper reports a NW trending zone of seismic activity; he does not conclude, however, that there is a NW trending tectonic fault.

In addition, the AEC's Savannah River Laboratory made a detailed geologic study of a region around the Savannah River site in connection with a radioactive waste storage proposal. The area covered lies across the proposed NW trend line. The study included many bore holes that confirmed the presence of buried triassic basins that had been implied from aeromagnetic data. No evidence of faulting was observed from the data taken from bore holes drilled into bedrock on two mile centers. The lineations in the triassic basins are oriented in the NE-SW direction.

Mr. Fischer pointed out that, geologically speaking, things change when you get to Charleston. There is a lump in the basement rock, called the Yamacraw Ridge. It is believed to be a fault that was active between 14 and 20 million years ago. Here one sees geologic structure that is different from anyplace else and may be the source of the over 400 earthquakes felt in the Charleston area since 1698. The basement rock is brittle and will not deform in a ductile manner.

The applicant stated that he was maintaining the flexibility to install four strands of fixed incore monitors consisting of seven detectors per strand. More strands could be installed if some of the moving incore monitors were replaced with fixed units.

Additional Remarks by D. Okrent

Significant uncertainties exist concerning the probable cause of the major 1886 Charleston earthquake, and the ACRS has received conflicting opinions regarding the probability that a major related earthquake might occur closer to the applicant's site. Several geologic and seismic experts recommended that use of a higher Safe Shutdown Earthquake bedrock acceleration of 0.2 g would be prudent unless the applicant can confirm by field studies his theory that the Yamacraw ridge is a fault responsible for the 1886 Charleston earthquake and that there are no structures which might lead to extension of the Charleston earthquake activity toward the site, or confirm the existence of some other source of the Charleston earthquake which permits the same conclusion with regard to the Summer site. I agree with these recommendations.

References

1. South Carolina Electric and Gas Company letter dated June 30, 1971; Application for Licenses; Preliminary Safety Analysis Report (PSAR), Volumes I through VI

2. Amendments 1, 3-14, 16 and 17 to PSAR

3. Safety Evaluation by Directorate of Licensing, dated August 29, 1972

References - ACRS OFFICE COPIES ONLY

1. DL Preliminary Report dated August 25, 1971

2. Fish and Wildlife Service letter dated June 22, 1972

3. J. A. Blume Report, dated September 29, 1972

Caucus (ACRS Members Only)

The Committee agreed to accept the applicant's and Staff's proposed seismic design value of 0.15g on the basis of the work that was presented and the stated intent for the ACRS to write a letter to the Commissioners urging that additional seismic research be performed in the Charleston area. Dr. Okrent could not agree with this position and indicated that he would append additional remarks to the Committee's letter on Summer.

The Committee agreed that a letter favorable to the construction of the Summer plant could be written and that there will probably be additional remarks attached.

The consultants were advised of the Committee's decision.

Conclusion

The Chairman informed Mr. Virgil C. Summer that the Committee believes it can write a letter favorable to construction of the Summer station with the seismic design values proposed by the applicant. There probably will be additional remarks appended to the letter that will relate to the seismic design values.

This ACRS request was acknowledged in a response by Lester Rogers of the Staff to ACRS Chairman Mangelsdorf, dated July 3, 1973. A study as requested was promised; however, the results of any such study by the Staff did not surface in the next several years.

Seismic engineering criteria had been developed and were being applied on construction permit reviews during this period. However, rather than being incorporated into the AEC Rules and Regulations via a rule-making procedure, they were made public and implemented via a simpler procedure, namely, Regulatory Guides.

Thus, the need for seismic criteria was recognized relatively early in the Regulatory process, and with time such criteria were developed. For reasons which are obscure, however, only a very modest, seismic safety research program was funded by the AEC during this time period. In a report dated January 17, 1968, the ACRS noted the formative stage of the AEC seismic safety research program and recommended work in several areas, including soil-liquefaction and soil-structure interaction. It also recommended development of a detailed program of earthquake engineering research although there apparently did not exist strong support for such a program within the AEC.

In a letter to Mr. Price dated May 17, 1971, the ACRS noted that its geological and seismological consultants had expressed concern that seismic conditions in the eastern U.S. were poorly understood and had recommended that emphasis be placed on the early development of information that would aid the AEC Regulatory groups in the determination of conservative parameters for the SSE. The latter identified the Charleston earthquake as requiring accelerated study, and made some other specific recommendations.

The thoughts in this memorandum were repeated in an ACRS report to AEC Chairman Dixie Lee Ray, dated May 16, 1973, which discussed the entire eastern U.S., and called out areas warranting special attention, including South Carolina, the St. Lawrence Valley, southeastern Missouri, western Ohio and the Cape Ann region of Massachusetts. During the review of the Greenwood Energy Center at the 172nd meeting, August 8-10, 1974, Dr. Carl Stepp of the Staff gave an estimate of the SSE return frequency which led to an ACRS memorandum to Mr. Muntzing, the Director of Regulation, as is reproduced on the following page.

Appendix A to 10 CFR Part 100 was adopted by the AEC in 1973. In giving its assent to publication of the seismic criteria as a proposed rule, the ACRS again raised the question of using a higher minimum design basis acceleration in its memo to Mr. Muntzing which is reproduced below.

June 11, 1973

L. Manning Muntzing
Director of Regulation

PROPOSED SEISMIC AND GEOLOGIC SITING CRITERIA, APPENDIX A TO 10 CFR PART 100

The Committee believes it acceptable to publish the draft Seismic and Geologic Siting Criteria, dated June 7, 1973. However, the Committee notes that the minimum acceptable accleeration currently in use (0.1 g SSE) frequently leads to considerable debate among seismic experts concerning an acceptable seismic design basis, even in regions of low seismicity. The Committee recommends that a study be initiated to determine the practicality and cost of raising the minimum design basis acceleration to 0.15 or 0.2 g for future reactors, except where foundation conditions and geological and seismic information are recognized by a clear consensus of expert opinion to warrant a lesser value. The Committee expects that an increased minimum design basis acceleration would assist in standardization, would provide increased margin to cover uncertainty, and would facilitate seismic aspects of regulatory review for many sites, without involving an important increase in cost, if factored into the original design.

/s/

H. G. Mangelsdorf
Chairman

cc: L. Rogers, RS
R. Minogue, RS
P. Bender, SECY

The Regulatory Staff response to this memorandum was twofold. First, they argued that the use of Appendix A did not introduce any requirement for a quantitative criterion on return frequency for the SSE. Secondly, they argued (qualitatively) that there were many large safety factors inherent in the seismic engineering design so that, overall, serious reactor accidents due to earthquakes should have a very low probability per year of occurrence.

More details are to be found in the letter from Muntzing to Stratton which is reproduced on the following pages.

AUG 1 2 1974

L. Manning Muntzing
Director of Regulation

SEISMIC DESIGN BASES

During the ACRS review of the Greenwood Energy Center on August 9, 1974, a member of the Regulatory Staff identified 10^{-5} as the probability per year of exceeding the proposed safe shutdown earthquake (SSE) of 0.12 g at the foundation. The Staff member stated further that this design basis is equivalent to that estimated for most of the recently reviewed construction permits east of the Rockies. This probability is clearly difficult to assess in view of the limited historical data and understanding of earthquake sources. However, 10^{-5} is a factor of 100 larger than the value of 10^{-7} identified by the Director of Regulation and the Regulatory Staff for possible sources of uncontrolled accidents in connection with considerations of anticipated transients without scram and of long-term safety objectives to be sought in the context of large numbers of reactors. There may be an additional factor of safety available in other aspects of seismic design, but it is not currently quantified. The Committee, therefore, requests the Regulatory Staff to address this apparent inconsistency and provide the Committee with those justifications and recommendations which are required to insure compatibility of this aspect of safety design with the overall objective.

Original Signed by
W. R. Stratton

W. R. Stratton
Chairman

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g-values selected for given sites. The estimated probability of 10^{-5} per year for the Greenwood site noted in your letter was one such staff judgment. Other experts might have different judgments about this probability for Greenwood and other sites. We believe that while such judgments are interesting, they do not yet rest on a sufficiently developed technological base to be used in making licensing decisions. The difficulty of making probability estimates results from the short sample of available earthquake data, the incompleteness of the data resulting in an inhomogeneous sample, the lack of knowledge about earthquake source regions, and the lack of information regarding maximum earthquakes in most regions.

We wish to emphasize that the selection of the SSE and the geological and seismological bases for the design of all nuclear plants are determined by the criteria and procedures prescribed by the Commission's rules as set forth in 10 CFR 100, and not on the basis of the above or any other quantified probability estimates not contained in those rules.

It is worth noting that conservatisms in the establishment of the safe shutdown earthquake acceleration spectrum and in the design of structures to withstand the safe shutdown earthquakes provide substantial margins of safety. The results of an earthquake damage study that has been completed for the February 1971 San Fernando earthquake (Ref. 5) show that many seismically designed conventional buildings experienced little damage, despite the fact that the majority of these buildings were designed in accordance with much less stringent seismic design requirements (UBC lateral force design requirements) than those applicable to seismic design of Category I structures in nuclear plants. For example, the following significant additional conservatisms are inherent in the seismic design bases required for all Category I structures, systems, and components:

a. An elastic dynamic analysis is performed using, as input, wide-band response spectra with conservative amplification factors. The use of wide-band spectra is conservative since the response spectra of recorded earthquake motions do not normally exhibit such wide-band characteristics. The amplification factors used are, in general, greater by one standard deviation than the mean amplification factors of actually recorded strong motion earthquake spectra.

b. Damping values, that are used in seismic analysis, are considerably smaller than those indicated by currently available test results. This tends to increase seismic responses significantly and thus provide a large margin of safety.

ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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ADVISORY COMM. ON REACTOR SAFEGUARDS
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Dr. W. R. Stratton, Chairman
Advisory Committee on Reactor Safeguards
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Dr. Stratton:

Your letter dated August 12, 1974, requested that the Regulatory staff address what the ACRS termed an apparent inconsistency between the staff's long-term safety objectives and comments made by the staff during the ACRS review of the Greenwood Energy Center regarding the probability of exceeding the safe shutdown earthquake.

As you noted, in discussions of anticipated transients without scram (ATWS) and elsewhere, the staff has identified as a desirable safety objective for a large population of reactors that the probability of an accident with consequences that would significantly exceed the Part 100 guidelines from one accident source should be 10^{-7} per reactor-year or less. For postulated accidents where we have been able to quantify or bound the probabilities (for example, in the ATWS case, and in considering aircraft crashes), we have used probability assessments in making licensing decisions (i.e., in the selection of design criteria).

In the case of seismic design, however, we are faced with very limited historical data on earthquake frequencies as a function of intensity. Rather than set design criteria on the basis of probability analyses, the Commission has chosen to develop the methodology given in 10 CFR Part 100, Appendix A, for selection of safe shutdown earthquakes (SSE) for proposed sites. We recognize that attempts have been and are being made (Refs. 1, 2, 3, 4) to define site seismicity quantitatively by probabilistic approaches. Although these approaches are of interest, much work remains to be done to achieve sufficient confidence in the methods to permit their use as a basis for specifying seismic design criteria.

Nevertheless, in response to persistent questioning in various forums (including ASLB hearings and ACRS meetings), knowledgeable members of the Regulatory staff have from time to time expressed their personal judgments as to the probability of equaling or exceeding the SSE

References

1. Otto W. Nuttli, "Magnitude-Recurrence Relation for Central Mississippi Valley Earthquakes", Bulletin of the Seismological Society of America, Vol. 64, No. 4, August, 1974

2. Earthquake Notes, Eastern Section, Seismological Society of America, Vol. XLIV, Nos. 3-4, July-December, 1973

3. Hans A. Merz and C. Allin Cornell, "Seismic Risk Analysis Based on Quadratic Magnitude-Frequency Law", Bulletin of the Seismological Society of America, Vol. 63, No. 6, 1973

4. C. Allin Cornell and Hans A. Merz, "Seismic Risk Analysis of Boston", ASCE National Structural Engineering Meeting, April, 1974

5. R. V. Whitman, et al, "Damage Statistics for High-Rise Buildings in the Vicinity of the San Fernando Earthquake", Department of Civil Engineering, MIT, Structural Publication No. 363, April 1973

c. Use of state-of-the-art soil structure interaction analysis procedures accounts for the deformability and compliance of plant foundation media and provides conservative system seismic response. In the soil structure interaction analysis, variations of soil properties are considered in order to assure conservative responses.

d. Seismic responses are determined by accounting for three conservatively defined input components (i.e., the maximum acceleration amplitude is assumed for both horizontal components and two-thirds of the maximum horizontal acceleration amplitude is assumed for the vertical component, at near rigid frequencies).

e. Either the time-history analysis method or the response spectrum method of analysis is used for seismic analysis of Category I structures, systems and components, accounting for all significant modal responses. This is significantly more conservative than the essentially single-mode response and one-directional input consideration usually employed in the design of conventional structures.

f. The allowable stresses and strains used in design of all safety-related structures, systems and components are selected such that the structures, systems and components are capable of withstanding, without failure, the design loadings with margin beyond the specified loads.

On the basis of such considerations, individual staff members and consultants have on occasion expressed their judgments as to the probable margins inherent in Category I structures, systems, and components. However, as your letter notes, the margins of safety inherent in use of our seismic design requirements are not currently quantified, nor do we believe that they can be, at this time, with sufficient rigor to be used in making specific licensing decisions.

Sincerely,

L. Manning Muntzing
Director of Regulation

The Seabrook Station site is near what is generally recognized as the Cape Ann-to-Ottawa Trend. Mechanisms for earthquake generation in the New England area are not well understood, and expert opinion differs concerning the potential for and probability of relatively large earthquakes at or near the site.

The Regulatory Staff have ultimately based their judgment as to an acceptable safe shutdown earthquake on the application of 10 CFR Part 100, Appendix A, rather than a probabilistic estimate of earthquake size versus recurrence interval. It is of interest to note that Appendix A provides only general guidance; furthermore, it specifically refers to the possible choice of a safe shutdown earthquake larger than that found in the historical record for a tectonic structure or province.

During the ACRS review the Regulatory Staff did state that the seismicity of the tectonic region applicable to the Seabrook site could be interpreted to be about an order of magnitude larger than other tectonic provinces having a similar maximum historical seismic event. Furthermore, a member of the Regulatory Staff stated that his estimate of the probability per year of occurrence of an earthquake of intensity MM VIII at the Seabrook site is about 10^{-4}, and the Staff did not rule out the possibility of a larger earthquake occurring within the region under consideration. They stated that conservatisms in analysis, stress limits, and other factors decrease the overall probability of a seismically induced accident exceeding 10 CFR Part 100 would be acceptably low. However, earthquakes are almost unique in their ability to fail each and every structure, system, component, or instrument important or vital to safety, and, in my opinion, the Staff evaluation of additional margin available from stress limits, methods of analysis, etc., did not consider all such systems, e.g., D.C. power or emergency A.C. power.

It is clear that the capability of a reactor to achieve safe shutdown, assuming its SSE occurs, cannot be fully demonstrated by test. Those limited, detailed and independent audits of seismic design of actual plants that have been published indicate that some inadequacies in design and construction exist. Equally or more important, it appears to be unlikely that the plant could survive safely, with a high degree of assurance, a larger earthquake having one or two orders of magnitude lower probability than the proposed SSE.

Given this background, and recognizing the substantial surrounding year-round population density and the very high nearby population during the summer months at Seabrook, I am left uneasy and believe it would be prudent to augment the proposed SSE

Since Appendix A did not provide any real guidance on how to allow for the very limited empirical data which was available (i.e., the short history of earthquake records) the matter was left to the judgment of the individual reviewer when small return frequencies were sought, and differences of opinion continued to exist.

As more return frequency estimates were provided by the Staff or the applicants for construction permit reviews following Greenwood, the estimates tended to become larger, and values larger than 10^{-4} per year for the probability of exceeding the SSE were introduced into the record.

Some seismic engineers began to provide their estimates of a rough quantification of the risk from earthquakes. For example, Professor Whitman of MIT, speaking for the Atomic Industrial Forum Ad Hoc Committee on Seismic Design Bases at a meeting held September 19, 1974, discussed conservatism in seismic design and listed the chain of steps and associated probabilities that combined to give the overall probability of an accident, as follows:

Structure (probability of failure due to an overstress)	$P_{F/O}$ =	$10^{-1} - 10^{-2}$
Analysis (probability of overstress occurring	$P_{O/S}$ =	$10^{-4} - 10^{-5}$
Response (response/intensity)	?	
Seismic risk (intensity for earthquake and probability of earthquake)	P =	$10^{-4} - 10^{-6}$
Overall risk		$10^{-7} - 10^{-8}$/yr

In 1974, the review of the Seabrook reactors again provided a focal point for the question of seismic design adequacy. Additional comments by member Okrent to the ACRS letter of December 10, 1974 are on the following pages.

considerable body of information which suggests that its estimate of seismic risk, and that of Whitman, may be much too low.

At a meeting of the ACRS Seismic Subcommittee held March 22-23, 1976 on Seismic Activity in the Eastern United States, a large body of new information appeared, as did a continuing wide disparity of expert opinion on the adequacy of 200 years of history to predict future low probability seismic events. At a next meeting of the Seismic Subcommittee on February 8-9, 1977, which dealt primarily with matters related to the soil-structure interaction part of seismic engineering, many differences of opinion concerning the adequacy of previously accepted practice were expressed.

Papers were published which disagreed with the methods and the results concerning seismic risk which had been given in WASH-1400 (NRC, 1975) including differences of a few orders of magnitude. (Hsieh, 1977).

Seismic risk estimates performed for Diablo Canyon 1 and 2 (Brunot, W. K., 1977) and for the proposed Clinch River Breeder Reactor (CRBRP-1, 1977) both disagreed with the conclusion in WASH-1400.

The controversy over what constitutes an adequate design basis appeared directly within the framework of the regulatory process. In connection with the operating license review for North Anna Units 1 and 2, the ACRS noted that its consultants favored a higher minimum seismic design basis (0.2 g) for future reactors in the eastern U.S., and recommended that the Regulatory Staff assure itself that there was considerable margin in the existing seismic design (SSE = 0.12 g on rock) with regard to equipment needed for safe shutdown of the reactor. The ACRS also made a similar recommendation in connection with an operating license review of the Davis-Besse Nuclear Station, as shown below:

> <u>Excerpt from letter of January 14, 1977 on Davis-Besse</u>
>
> The structures and components of Davis-Besse, Unit 1, were designed for a Safe Shutdown Earthquake (SSE) acceleration of 0.15g at the foundation level. Because of changes in the regulatory approach to selection of seismic design bases, the Committee believes that an acceleration of 0.20g would be more appropriate for the SSE acceleration at a site such as this in the Central Stable Region. The Applicant presented the results of preliminary calculations concerning the safety margins of the plant for an SSE acceleration of 0.20g. The Committee recommends that the NRC Staff review this aspect of the design in detail and assure itself that significant margins exist in all systems required to accomplish safe shutdown of the reactor and continued shutdown heat removal, in the event of an SSE at this higher level. The Committee believes that such an evaluation need not delay the start of operation of Davis-Besse, Unit 1. The Committee wishes to be kept informed.

acceleration of 0.25 g.

I also wish to reiterate my conclusion previously stated in connection with the review of Grand Gulf Units 1 and 2, namely that it would be prudent to provide some additional margin in the seismic design bases for most future nuclear plants sited east of the Rockies.

The Reactor Safety Study, WASH-1400 in 1975 provided support to those who estimated a low risk of a serious reactor accident from seismic causes. The estimate given there was that seismically induced core melt has a frequency of 5×10^{-7} per reactor year for a reactor on an average foundation, and less than 10^{-7} per reactor year for a reactor on a rock foundation.

In a preliminary evaluation of WASH-1400 issued November 15, 1975 by a Review Group from the NRC Regulatory Staff, the conclusions on seismic risk in WASH-1400 were generally endorsed, as follows:

Seismic Considerations

Comments on the WASH-1400 draft report identified large earthquakes as a potential mechanism for causing multiple system failures and questioned the probabilities given for seismic failure modes. The Study has revised its treatment of this issue to include explicit recognition of the observational data regarding the distribution of earthquakes with respect to intensity. Also examined were the safety margins in a facility when subjected to earthquakes larger than used for design purposes. The calculations refelct the views of acknowledged authorities and current information.

The Study provides a significant perspective on the relationship between basic ground motions, the energy absorption capacity of a structure, and the probability of exceeding design limits. The Study conclusions relating to the probability of failure may be open to question in view of the differing seismic margins in the design of electrical, mechanical, and structural components and the potential for common mode failures. While a more detailed application of the Study's methodology would be needed to definitively establish the meltdown probabilities, it does not appear that seismic events are strong contributors to risk.

However, since issuance of the Reactor Safety Study, there has been a

And in connection with ACRS review construction permit application for the Cherokee-Perkins reactors in 1977, there were additional remarks which questioned the adequacy of the proposed SSE and the correctness of the treatment of seismic risk in WASH-1400. Excerpts from the ACRS letter on Cherokee-Perkins follow:

> ACRS letter on Cherokee-Perkins, April 14, 1977
>
> For the safe shutdown earthquake for both the Perkins and Cherokee Stations, an acceleration of 0.15g will be applied at the foundation level of rock-supported structures. For structures not supported on rock, the design ground motion will be applied at the level of continuous rock and propagated upward to the foundation level. The operating basis earthquake acceleration will be 0.08g, similarly applied.
>
> Additional remarks by ACRS member D. Okrent
>
> I believe that the philosophy and criteria of Appendix A of 10 CFR 100, and their application by the NRC Staff in setting SSE values, should be re-evaluated as part of an early overall reassessment of the current approach to seismic safety design. I believe that the estimates of the contribution of earthquakes to overall nuclear reactor safety risk, as given in the Reactor Safety Study (WASH-1400) are not without fault, and that seismic contribution to risk is underestimated in that study.
>
> I find the Applicant's estimate of the return frequency of the SSE at the Cherokee and Perkins sites of greater than 10^{-4} per year to be unsatisfactorily large, particularly in view of his arbitrary cutoff at MM VII of the earthquakes permitted to contribute to this probabilistic assessment. For Cherokee/Perkins, I find the proposed SSE of 0.15g marginally acceptable and would prefer that a value of 0.2g be employed at the foundation level on rock.

In 1977, the Nuclear Regulatory Commission initiated a major new research program in seismic safety including the possible application of probabilistic techniques.

IV. BROWNS FERRY

A. Executive Session

The Committee considered the format for discussion with the applicant, which Dr. Monson had prepared.

During the discussion, Dr. McKee indicated his concern over leaving any items of significance to the operating license review, in view of the Committee's present difficulties with the Connecticut Yankee project.

Dr. O'Kelly thought that the approach which called for proving out the diesel generator startup mode prior to operating license was a good one.

Dr. Okrent, however, was in sympathy with Dr. McKee. He anticipated overwhelming pressure at the time of operating license review when the Committee might be faced with as many as 20 Browns Ferry type reactors. Dr. Okrent was skeptical that all of the reservations could be satisfactorily answered and felt that the Committee should consider carefully the wisdom of proceeding in this way.

Dr. Gifford did not think that there was any alternative.

Dr. Okrent observed that answers to such questions as the mode of fuel failure are in fact necessary to an adequate ACRS review. He pointed out that the Committee would be told by the Connecticut Yankee group that over a wide range of accidents 40 - 50 - 75% of the fuel fails. Their picture is of the cladding ballooning out and finally rupturing and of fairly undesirable results for a large number of accidents. In addition, the meltdown/meltthrough Task Force has concluded that the course of a loss-of-coolant-accident is uncertain after many fuel failures.

Mr. Palladino commented that he had realized in the course of the Connecticut Yankee Subcommittee meeting that all of the larger power reactors sustain considerable fuel damage after loss of coolant and there is no information about what follows.

Dr. Bush felt that definitive answers to the questions should be postponed and that if they were not forthcoming the applicant would simply have wasted 200 million dollars. Dr. Hanauer thought that that was not a realistic appraisal of what the situation would actually be.

B. Regulatory Staff

Dr. Morris reported that the Committee's reservations had been

6.1 GENERIC ITEMS

Although matters of generic importance either to a class of LWRs or to all LWRs had been previously identified, for example the recommendation for improved primary system integrity in the Indian Point 2 and Dresden 3 letters, the ACRS review of a construction permit application for Browns Ferry Units 1 and 2 early in 1967 can probably be considered to be the genesis of the generic items list. This was a very difficult and controversial ACRS review. While the Browns Ferry site was quite remote, these BWRs represented a large increment in power level over those BWRs previously approved for construction and were a very large factor in power above any operational BWRs. They came after the pressure vessel letter in late 1965 and the "China Syndrome" matter of 1966, wherein major safety issues had been raised and had been either resolved in only a preliminary (conceptual way) or where only limited steps had been taken by the industry to improve the situation. Furthermore, a higher power density was being proposed for Browns Ferry, bringing fuel performance limits and potential safety concerns related tofeul melting more into focus. And finally, Browns Ferry was expected to be the prototype of similar reactors soon to be proposed for much more populated sites.

By the time of the 83rd meeting, March 9-11, 1967, the ACRS had prepared a draft position which included a very considerable number of specific matters on which more information was required. The applicant had been advised of the specific nature of each, and had been requested to be prepared to discuss these in detail. The Committee members had mixed opinions about the proper course to follow. The summary of the Executive Session, reproduced on the following pages, illustrates some of the thinking.

Also included is the brief session with the Regulatory Staff.

> nature or extent as to interfere with heat removal sufficiently to cause clad melting would not occur. The Committee believes that additional evidence, both analytical and experimental, is needed and should be obtained to demonstrate that this model is adequately conservative for the power density and fuel burnup proposed.*

The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly, by inlet coolant orifice blockage or by other means, to be remote. However, the resulting effects in terms of fission product release, local high pressure production, and possible initiation of failure in adjacent fuel elements are not well known. Information should be developed to show that such an incident will not lead to unacceptable conditions.*

A linear heat generation rate of 28 KW/ft is used by the applicant as a fuel element damage limit. Experimental verification of this criterion is incomplete, and the applicant plans to conduct additional tests. The Committee recommends that such tests include heat generation rates in excess of those calculated for the worst anticipated transient and fuel burnups comparable to the maximum expected in the reactor.*

The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system and of inspection during service life. Because of the higher power level and advanced thermal conditions in the Browns Ferry Units, these matters assume even greater importance. The Committee recommends that the applicant implement those improvements in primary system quality which are practical with current technology.*

* The Committee believes that these matters are of significance for all large water-cooled power reactors, and warrant careful attention.

> transmitted to the applicant. TVA has requested that, if a letter is written, items which are not specific to the Browns Ferry system but are rather industry-wide problems, should be so identified. Such things as the fuel damage limit, mode of cladding failure, effects of flow blockage, etc., would be included.
>
> The Regulatory Staff feels that a construction permit can be issued. The reservations on fuel element failure mode, adequacy of the core spray system and results of meltdown of a subassembly will require that more information be developed between now and the time of review for a provisional operating license. The rest of the Committee's concerns can be resolved by further Regulatory Staff works or appropriate limits in the Technical Specifications.

Following the discussion with the applicant, the ACRS decided it could issue a letter having many reservations. And it asterisked those items that were properly identified as industry-wide problems and not specific to Browns Ferry. The asterisked items from the Browns Ferry letter of March 14, 1967, are reproduced below.

EXCERPT FROM ACRS LETTER 3/14/67 - BROWNS FERRY ITEMS

WHICH HAVE BEEN NOTED WITH *

> The complex of emergency core cooling systems for Browns Ferry is similar to that proposed for the Quad-Cities reactors. Each reactor is provided with a high pressure coolant injection system; a low pressure coolant injection, or flooding, system; and two core spray systems. Because of the higher core power density and power level, substantial increases have been made in the flooding system and core spray system capacities. The Committee feels that the emergency core cooling systems proposed have a high probability of preventing core meltdown in the unlikely event of a loss-of-coolant accident. It notes, however, that although calculated peak fuel temperatures in such an accident are similar to those for the Quad-Cities reactors, the calculated number of fuel elements reaching undesirably high temperatures is greater. Also, the time margin available for actuation of the systems is less. Because of these factors and the importance of the effective functioning of emergency core cooling systems, the Committee believes the adequacy of these systems should be further corroborated by the following two measures:
>
> 1. Analysis indicates that a large fraction of the reactor fuel elements may be expected to fail in certain loss-of-coolant accidents. The applicant states that the principal mode of failure is expected to be by localized perforation of the clad, and that damage within the fuel assembly of such

II. Meeting with the AEC

The Committee was joined by Commissioners Ramey, Tape and Johnson, and by the General Manager, the Director of Regulation, and members of their staffs.

Safety Research Programs

Mr. Palladino noted that he wished to discuss with the Commissioners the implementation of work necessary to the solution of technical problems identified in recent ACRS letters as common to the nuclear industry. It is the opinion of the Committee that these matters need early attention so that they may be resolved in a timely fashion. Mr. Palladino thought, in particular, that it would be most helpful if the industry were informed of what could be expected in the way of solutions to such problems from AEC-sponsored research programs. He cited the example of the Oconee Nuclear Power Station review during which the applicant had indicated his dependence on AEC research programs for the solution of his problems. Mr. Palladino pointed out that the Committee was not aware of programs specifically aimed at providing such solutions, noting that he did not intend to imply whether or not the AEC should be sponsoring such research.

Commissioner Ramey thought that the Steering Committee should provide a mechanism for making such identifications.

Mr. Hollingsworth agreed that the direction of AEC-sponsored research should be identified for industry. He suggested that he and Mr. Price discuss a mechanism for doing this. Mr. Price agreed, but added that it was really necessary to indicate to the applicants that they alone must carry the burden of proposing a mechanism for solution of such problems. This mechanism could be their own research programs, AEC-sponsored research, research programs funded by other industrial concerns, etc..

Mr. Kavanaugh agreed that there was room for a better understanding of what is now being done.

Commissioner Johnson asked if the Committee had identified the problems of concern, specifically. Mr. Palladino replied that this had been done beginning with the Committee's letter on the TVA proposal. At that time, it was recognized that some of the outstanding problems were industry

When the ACRS completed its review of the Vermont Yankee BWR in June, 1967, it repeated several of the items from the Browns Ferry letter, once again labeling them with asterisks. And in July, 1967, when the ACRS wrote a report on the Oconee PWRs (supplied by Babcock and Wilcox), it tagged several reservations with asterisks, including two new ones, one concerning the possible effect ofthermal shock from ECCS operation on pressure vessel integrity and a second relating to the effects of blowdown forces from a large LOCA on core and other primary system components.

At its 89th meeting, September 7-9, 1967, the ACRS discussed the matter of resolution of asterisked items with the Commissioners, as per the excerpt from the minutes on the following pages.

After the meeting with the Commissioners, the ACRS discussed the matter further. There was some general unease about how to get the asterisked items dealt with on a timely basis. It was decided to send letters to Mr. Price, the Director of Regulation, and to Mr. Hollingsworth, the General Manager, asking how each was addressing matters raised by the ACRS as significant to all large LWRs. A copy of the letter to Mr. Price is reproduced on the following page.

At its December, 1967 meeting, the ACRS completed a construction permit report on the Diablo Canyon Unit 1 PWR (Westinghouse) which included several asterisked items, including one on separation of protection and control. At its 93rd meeting, January 11-13, 1968, the ACRS discussed how to proceed further with the matter of asterisked items, both procedurally and technically. There was sentiment among some Committee members not to continue repeating the items on all future letters. It was decided to include a detailed list on the upcoming Fort Calhoun report (a Combustion Engineering PWR), so that each LWR Reactor vendor had received at least one such list. It was also decided that a Subcommittee should be appointed to attempt to resolve the items, and that the Subcommittee should try to complete its work before completion of ACRS action on another reactor at a site which was similar, population-wise, to Indian Point 2. (The Zion Station was on the horizon for ACRS review.)

The ACRS began using a brief paragraph in succeeding letter reports on LWRs in the form "other problems related to large water reactors have been identified by the Regulatory Staff and ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to <u>the (   ) reactor."</u>

New items continued to be identified in specific letters, and at least initially, were asterisked if they were deemed generic; and sometimes an old item was specifically emphasized in the report on a particular case.

The Regulatory Staff responded in several months to the ACRS request of September 22, 1967, with a status report on the various asterisked or other generic items. And on July 28, 1968, the General Manager wrote the ACRS and provided a brief evaluation of how and to what extent the AEC Reactor Safety Research Program was responding to ACRS concern.

By the time that ACRS action was completed on the construction permit application for Zion Station Units 1 and 2, it was clear that progress was slow and was going to remain slow on most of the asterisked items. The Zion letter is reprocuded in Chapter 2. The list of items was continuing to grow (for example, the question of anticipated transients without scram, ATWS, was raised in 1969. See Chapter 4).

By joint effort of the Regulatory Staff and the ACRS, acceptable regulatory positions were developed on many of the open or unresolved topics wide and these were identified in the letter by asterisks. Mr. Palladino noted that he had been specifically addressing himself to those questions.

Mr. Shaw felt that the AEC had identified its R&D programs and the time scale on which they will be carried out. He suggested that even in the case of cooperative undertakings with industry, the Commission could not be put in the position of fulfilling license requirements for specific applications.

Dr. Okrent felt that, if the approach of making individual applicants responsible for indicating mechanisms of resolving the outstanding safety problems is followed strictly, the question will come to a head at the Committee's next meeting. Mr. Price agreed that that might well be the case and added that the Committee should get assurance from the applicants involved that there exists some R&D program which has a reasonable chance of providing solutions. He suggested that the feeling seems to be that the asterisked items in Committee letters call for information which is beyond, or which will be needed before, that to be derived from AEC programs. If this is, indeed, the case, the applicants should be made aware of it as soon as possible. Dr. Zabel noted that while the asterisks referred to a footnote on industry-wide problems, the content of the paragraphs was addressed to the applicant specifically.

Mr. Kavanagh noted that he had been trying to determine whether the total program was adequate based on the Committee's letter of October 1966, and other needs. He has been having some difficulty doing this.

Dr. Okrent pointed out that with two new reactor types either under construction or planned and with the indication that there are continuing safety problems in water reactor designs, more money might be needed for the research program. He pointed out that new reactor types bring with them whole new classes of problems. Mr. Kavanagh felt that it would not be possible to go much further in terms of additional funds. In answer to a question by Commissioner Ramey, Mr. Shaw noted that the 1968 budget has 60% of the safety research funds allocated for water reactor problems, about 15% for fast breeders, and about 1% for high temperature gas cooled reactors. Mr. Shaw pointed out that the present high temperature gas cooled reactor proposal carries with it its own funds for safety research. There is no way to obtain funds and solve problems on a short time scale.

and some of the asterisked, or generic items began to be resolved with the issuance of Safety (or Regulatory) Guides.*

And by late 1972, the ACRS wrote its first formal report on the status of generic items. The letter is reproduced on the following pages.

As noted in the letter, the Committee employed a limited definition of the term "resolved", namely that a specific conclusion or policy decision had been reached by the Regulatory Staff and ACRS. Most of the time, this meant that a modification acceptable to the Regulatory groups should be employed in future plants. The problem of how to backfit, if changes were desirable or necessary, was frequently a thorny one, and ended up a matter of engineering judgment.

Clearly, in 1972, many of the asterisked items identified in Browns Ferry and succeeding plants were still not resolved, even though these reactors were requesting operating licenses. The Regulatory Staff and the ACRS had to arrive at a judgment as to whether power operation was acceptable while a large number of generic items remained unresolved. The judgment made was that such operation would not pose undue risk to the public health and safety, and that when these items were resolved, a backfit decision would have to be made.

In a 1975 paper entitled, "The ACRS - Generic Issues and Standardization", (Bush, 1976) long-time ACRS member Spencer Bush, who devoted a considerable effort to the resolution of many such issues by the development of safety guides, provided comments on the question of generic issues resolved by that time. Twenty-one had been resolved by means of a Regulatory Guide, seven by the development of acceptable codes and standards,

* The first safety guides were issued in the Fall of 1970. A letter dated December 8, 1971, from L. M. Muntzing to E. J. Bauser, Executive Director, Joint Committee on Atomic Energy gives a definition of "Safety Guides".

> "Safety guides are not regulations nor are they intended as a substitute for regulations; therefore, compliance with safety guides is not required. These guides are used to describe solutions to safety issues in facility licensing cases where it has not yet been determined that a particular solution to a specific safety question should be made a Commission requirement and included in the regulations. Specifically, the guides serve to identify safety issues that should be considered in the design and in the evaluation of nuclear power plants and to describe a set of principles and specifications which, if satisfied, represent a solution of these issues acceptable to the regulatory staff and the Advisory Committee on Reactor Safeguards. Solutions other than those set out in the guides will be acceptable if they provide a sufficient basis for the findings requisite to the issuance of a construction permit or operating license by the Commission."

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 22, 1967

Mr. Robert E. Hollingsworth
General Manager
U. S. Atomic Energy Commission
Washington, D. C.

Dear Mr. Hollingsworth:

During the eighty-ninth ACRS meeting, there was some discussion of the Commission's reactor safety research program and the contribution it will make to the resolution of items identified in recent ACRS reports as matters which are significant for all large water-cooled power reactors.

As a follow-up to this discussion, the Committee has asked me to request, for use by the ACRS, a summary of the work related to these topics which the Commission is currently carrying on or has planned as part of its safety research program and an indication of the dates by which significant results on each item can be expected. The Committee would also be interested in a summary of the bases used in deciding what work is to be included in the Commission's safety research program for water-cooled reactors.

In this connection, it should be noted that the Committee plans to continue its investigation of each of these items in its review of specific reactor projects.

Sincerely yours,

/s/

N. J. Palladino
Chairman

COPY

possibly, implemented. In fact, requirements may differ for specific plants because of such factors as site characteristics and construction authorization dates, especially as they apply to backfit requirements.

Group I of the attachment includes generic problems that have been resolved together with the specific action that resulted in the resolution. Group II includes those items for which resolution on a generic basis is still pending. The ACRS and the Regulatory Staff will continue to consider Group II items and their significance to safety on a case-by-case basis until generic resolution is reached. Formal actions, such as issuance of Regulations or Safety Guides, are anticipated for many of the Group II items.

The ACRS expects to report to you from time to time on the status of generic items.

Sincerely yours,

/s/

C. P. Siess
Chairman

Attachment:
Generic Items

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 18, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS

Dear Dr. Schlesinger:

The Advisory Committee on Reactor Safeguards and the Regulatory Staff have identified a number of safety problems during recent years that are common to a specific type of light-water reactor or to all light-water reactors (LWRs). The "generic items" discussed herein have been cited in Committee reports pertaining to the construction or operation of LWRs; additional generic items not cited in the report have been identified by the Directorate of Licensing. The ACRS customarily has used a general paragraph to cover those generic items noted in its previous reports. This paragraph reads: "Other problems relating to large water reactors, which have been identified by the Regulatory Staff and the ACRS and cited in previous reports, should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed". The use of such a paragraph has permitted the ACRS to emphasize either new or particularly significant generic problems without listing in detail those problems cited in the past.

The Committee believes that many of the generic items cited in its reports have been resolved by actions of the applicants or decisions of the Regulatory Staff, in cooperation with the ACRS. Resolution of the remaining items are pending.

The Committee defines "resolved" to mean that a specific conclusion or policy decision has been reached by the Directorate of Licensing and the ACRS. Resolution of an item indicates that the Committee is satisfied in a generic sense; however, this does not mean that improvements should not be investigated and,

15. Pressure Vessel Surveillance of Fluence and NDT Shift:
    Covered by 10 CFR 50, Appendix A and proposed Appendix H; and ASTM Standard E-185-70

16. Nil Ductility Properties of Pressure Vessel Materials:
    Covered by 10 CFR 50, Appendix A and proposed Appendix G; and ASME BPV Code, Section III

17. Operation of Reactor With Less Than All Loops in Service:
    Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent.

18. Criteria for Preoperational Testing:
    Covered by Operational Guide for the Planning of Initial Startup Programs, December 7, 1970

19. Diesel Fuel Capacity:
    Covered by ACRS-Regulatory Staff position requiring 7 days fuel

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends:
    Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences. ACRS-Regulatory Staff position document will be prepared.

21. Operating One Plant While Other(s) is/are Under Construction:
    Specific requirements have been established by ACRS-Regulatory Staff. Position will be prepared.

22. Seismic Design of Steam Lines:
    Covered by Safety Guide 29

23. Quality Group Classifications for Pressure Retaining Components:
    Covered by Safety Guide 26

24. Ultimate Heat Sink:
    Covered by Safety Guide 27

25. Instrumentation to Detect Stresses in Containment Walls:
    Covered by Safety Guide 18

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps:
   Covered by Safety Guide 1

2. Emergency Power:
   Covered by Safety Guides 6 and 9 and portions of IEEE-308 (1971)

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA):
   Covered by Safety Guide 7 and Supplement

4. Instrument Lines Penetrating Containment:
   Covered by Safety Guide 11 and Supplement

5. Strong Motion Seismic Instrumentation:
   Covered by Safety Guide 12

6. Fuel Pool Design Bases:
   Covered by Safety Guide 13

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles:
   Covered by Safety Guide 14

8. Protection Against Industrial Sabotage:
   Covered by Safety Guide 17

9. Vibration Monitoring of Reactor Internals and Primary System:
   Covered by Safety Guide 20

10. Inservice Inspection of Reactor Coolant Pressure Boundary:
    Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI

11. Quality Assurance During Design, Construction, and Operation:
    Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971; Safety Guide 28; and Proposed Standard ANS-3.2

12. Inspection of BWR Steam Lines Beyond Isolation Valves:
    Covered by ASME BPV Code, Section XI

13. Independent Check of Primary Sytem Stress Analysis:
    Covered by ASME BPV Code, Section III

14. Operational Stability of Jet Pumps:
    Tests and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns for this generation plant.

10. Anticipated Transients Without Scram:

    Data provided by applicants. Evaluation is required to permit a decision to cover BWRs and PWRs operating and under construction.

11. Radwaste Management:

    10 CFR 50, Appendix I, covers in part: Formal rulemaking decision required to implement fully.

12. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock:

    Safety Guide 2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data on irradiated steels from the Heavy Section Steel Technology Program.

13. Instruments to Detect Fuel Failures:

    Instrumentation exists to detect fuel failures: ACRS-Regulatory Staff believes progress is satisfactory; however, continuing work is required.

14. Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel:

    State-of-the-Art results appear promising. More work may be required prior to decision as to installation of equipment.

15. Common Mode Failures:

    Requirements for diverse components should be established.

16. Emergency Core Cooling System Capability:

    Need for improvement cited by Regulatory Staff and stated in ACRS report of January 7, 1972. Further studies and evaluations are in progress.

17. Behavior of Reactor Fuel Under Abnormal Conditions:

    This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

Group II - Resolution Pending

1. Adequacy of Primary System Leak Detection and Location:

   ACRS-Regulatory Staff position requires systems in addition to sump measurements.

2. Positive Moderator Coefficient:

   One solution is use of fixed burnable poison in core.

3. Use of Sensitized Stainless Steel:

   ACRS-Regulatory Staff position is to minimize use.*

4. Protection Against Pipe Whip:

   Ultimate decision will depend on review of available information on pipe break probabilities.*

5. Turbine Missiles:

   Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed.

6. Fixed In-Core Detectors on High Power PWRs:

   Some information is available.

7. Performance of Critical Components (pumps, cables, etc.) in Post-LOCA Environment:

   Substantial information available in topical reports. Evaluation is required to determine if all necessary information is on hand.*

8. Effective Operation of Containment Sprays in a LOCA:

   Extensive documentation in topical reports. Review and evaluation are required.

9. Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments:

   Analyses made in topical reports. Evaluation required by ACRS-Regulatory Staff.

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* A Safety Guide is in preparation.

and twelve by the adoption of a staff position or other means. In a few cases, information coming from safety research programs provided the basis whereby a new requirement was developed or an item was judged to be a non-problem. On occasion, particularly ATWS, an item has been placed in the "resolved" category prematurely, because what was thought to be a regulatory position became "unstuck" and was not implemented.

The ACRS has reported periodically on the status of generic items in letters to the Chairman of the Commission. Reproduced on the following pages are portions of Report No. 6.

The attachment to this report includes a brief description of each unresolved item and a priority listing. A few descriptions are included for information purposes.

An examination of the generic items covered in the ACRS report of November 15, 1977, both resolved and unresolved, shows that they cover a wide range of topics, that they vary considerably in their specificity, and they vary considerably in their potential or probable impact on reactor safety. They varied as well in the way they arose.

Item 1 in Group I of resovled generic items relates to net positive suction head from ECCS pumps. This matter arose from philosophic considerations. During ACRS review of one or two specific BWRs during the late 1960's, it was ascertained that the designers assumed containment integrity in assessing the containment pressure head available to prevent cavitation of ECCS pumps. The containment is very much a safety-related structure and its ability to withstand LOCA forces is carefully analyzed. Hence, this assumption by designers ws not unreasonable. On the other hand, it could be argued that it was desirable that ECCS function not depend on containment integrity, so that some low probability event involving a major loss of containment integrity in a LOCA, e.g., gross failure of a large containment penetration, not lead automatically to core melt. Furthermore, it was clear that reactor design could equally well proceed without relying on containment integrity for adequate pump head, if the approach was adopted from the beginning. The ACRS flagged the matter as a desirable change, and it was adopted after not too long a time had passed.

Item 7 in Group 1 related primarily to the integrity of flywheels on primary system pumps in PWRs. The potential for missile generation from various sources had received review from time to time. During the ACRS construction permit review of Indian Point 3, the matter of the possible failure of pump flywheels and the potentially adverse affects on steam generator integrity and other safety-related components was raised. Experience with flywheels of that type had been good, and it was judged that with proper care in design, fabriacation and inspection, the probability of unacceptable affects from gross flywheels failure should be acceptably low. While the existing quality assurance efforts in the

18. Emergency Power for Two or More Reactors at the Same Site:

    Additional work is required on protection systems for multiple units.

19. Main Steam Isolation Valve Leakage of BWRs:

    A definitive position is required of ACRS-Regulatory Staff in the light of continuing experience.

20. Instrumentation to Follow the Course of an Accident:

    Some equipment exists; further analyses are required to establish equipment requirements.

21. BWR Recirculation Pump Overspeed During LOCA:

    Topical reports prepared. Decision required by ACRS-Regulatory Staff.

22. The Advisability of Seismic Scram:

    Further studies required to establish need.

"Resolved" as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, recognizing that technical evaluation and satisfactory implementation are yet to be completed. Anticipated Transients Without Scram represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, recognizing that further steps are desirable, as practical, or that different aspects of the problem require further investigation. Examples are the possibility of improved methods of locating leaks in the primary system, and of improved methods or augmented scope to inservice inspection of reactor pressure vessels.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachments:
(1) Group I; (2) Group IA; (3) Group IB; (4) Group IC; (5) Group ID; (6) Group IE; (7) Group II; (8) Group IIA; (9) Group IIB; (10) Group IIC; (11) Group IID; (12) Group IIE; and (13) Table 1, Priorities For Resolution of ACRS Generic Items.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 15, 1977

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS: REPORT NO. 6

Dear Dr. Hendrie:

The Advisory Committee on Reactor Safeguards has previously reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letters of December 18, 1972, February 13, 1974, March 12, 1975, April 16, 1976 and February 24, 1977. Since the Committee limits its definition of generic items to those cited specifically in its letters pertaining to projects and related matters, the attached listing is not all-inclusive; the Nuclear Regulatory Commission Staff has additional generic items.

Groups I through ID of the attachments are a reiteration of the generic items considered resolved at the time the Committee issued its Report No. 5 on February 24, 1977. Group IE includes those items resolved since February 1977. Following each resolved item is a brief statement of the specific action that resulted in the resolution. Groups II through IID include items previously listed as those for which resolution on a generic basis is still pending. Group IIE includes those added in the present report. The ACRS and the NRC Staff will continue to consider the safety significance of items in Groups II through IIE on a case-by-case basis until generic resolution is reached. Formal actions, such as issuance of Regulations or Regulatory Guides, are anticipated for many of these items.

Owing to questions raised concerning the scope and intent of various generic issues, the Committee has incorporated into the attachments a brief description for all unresolved items cited in this report.

With regard to the status of generic issues, as they apply to each plant, the NRC Staff addresses the status of the pertinent issues in the applicable Safety Evaluation Report. The ACRS identifies those that it believes relevant in its reports on individual projects.

The ACRS has received requests concerning the priorities to be placed on the resolution of outstanding generic issues. Such priorities are shown in Table 1, attached.

Group I Continued

15. Pressure Vessel Surveillance of Fluence and NDT Shift: Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185.

16. Nil Ductility Properties of Pressure Vessel Materials: Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; "Report on the Integrity of Reactor Vessels for Light-Water Power Reactors," (WASH-1285) by the Advisory Committee on Reactor Safeguards dated January 1974.

17. Operation of Reactor With Less Than All Loops In Service: Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent. Cited in Standard Review Plan Appendix 7-A, Branch Technical Position EICSB 12.

18. Criteria for Preoperational Testing: Covered by Regulatory Guide 1.68.

19. Diesel Fuel Capacity: Covered by ACRS-Regulatory Staff position requiring 7 days fuel (Standard Review Plan 9.5.4).

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends: Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction: Specific requirements have been established by ACRS-Regulatory Staff. Covered in Regulatory Guide 1.17, 1.70 Section 13.6.2; 1.101; ANSI N 18.17 and Standard Review Plan 13.3 Appendix A and 13.6.

22. Seismic Design of Steam Lines: Covered by Regulatory Guide 1.29.

23. Quality Group Classifications for Pressure Retaining Components: Covered by Regulatory Guide 1.26.

24. Ultimate Heat Sink: Covered by Regulatory Guide 1.27.

25. Instrumentation to Detect Stresses in Containment Walls: Covered by Regulatory Guide 1.18.

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps: Covered by Regulatory Guide 1.1.

2. Emergency Power: Covered by Regulatory Guides 1.6, 1.9, and 1.32 and portions of IEEE-308 (1971).

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA): ACRS concurred in proposed Staff position, covered by NRC Standard Review Plan for Nuclear Power Plants.

4. Instrument Lines Penetrating Containment: Covered by Regulatory Guide 1.11 and Supplement.

5. Strong Motion Seismic Instrumentation: Covered by Regulatory Guide 1.12.

6. Fuel Storage Pool Design Bases: Covered by Regulatory Guide 1.13.

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles: Covered by Regulatory Guide 1.14.

8. Protection Against Industrial Sabotage: Covered by Regulatory Guide 1.17.

9. Vibration Monitoring of Reactor Internals and Primary System: Covered by Regulatory Guide 1.20.

10. Inservice Inspection of Reactor Coolant Pressure Boundary: Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI and Regulatory Guide 1.65.

11. Quality Assurance During Design, Construction and Operation: Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971, Regulatory Guides 1.28, 1.33, 1.64, 1.70.6 and Proposed Standard ANS-3.2.

12. Inspection of BWR Steam Lines Beyond Isolation Valves: Covered by ASME BPV Code, Section XI.

13. Independent Check of Primary System Stress Analysis: Covered by ASME BPV Code, Section III.

14. Operational Stability of Jet Pumps: Test and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns.

Group IB - Generic Items Resolved Since February 13, 1974

1. Positive Moderator Coefficient: PWRs presently have or expect to have zero or negative coefficients. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into PWRs to reduce otherwise excessive positive coefficients to allowable values.

2. Fixed Incore Detectors on High Power PWRs: Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

3. Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment: Qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and IEEE Standards 382-1972, 383-1974, 317-1972, 323-1974.

4. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments: On designs prior to GE Mark III containment, resolution lies in surveillance and testing of vacuum relief valves. For Mark III containments, an additional requirement is that the design be capable of accommodating a bypass equivalent to one square foot for a given flow condition.

5. Emergency Power for Two or More Reactors at the Same Site: Resolved by issue of Regulatory Guide 1.81.

6. Effluents from Light-Water-Cooled-Nuclear Power Reactors: Resolved by issue of Appendix I to 10 CFR 50.

7. Control Rod Ejection Accident: Resolved for PWRs by Regulatory Guide 1.77.

Group IA - Generic Items Resolved Since December 18, 1972

1. Use of Furnace Sensitized Stainless Steel: Covered by Regulatory Guide 1.44.

2. Primary System Detection and Location of Leaks: Covered by Regulatory Guide 1.45.

3. Protection Against Pipe Whip: Covered by Regulatory Guide 1.46.

4. Anticipated Transients Without Scram: Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, September 1973.

5. ECCS Capability of Current and Older Plants: Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors," December 28, 1973.

Group ID - Generic Items Resolved Since April 16, 1976

1. Instruments to Detect (limited) Fuel Failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen, June, 1976 resolves issue for limited fuel failures, but not for severe failures (See II-4).

2. "Instrumentation to Follow the Course of an Accident" Regulatory Guide 1.97 Revision 1 resolves ACRS concerns.

3. Pressure in Containment Following LOCA - NRC document, "Containment Subcompartment Analysis" September 1976.

4. Fire Protection. Resolved by Branch Technical Position 9.5.1, and Regulatory Guide 1.120.

Group IC - Generic Items Resolved Since March 12, 1975

1. Main Steam Isolation Valve Leakage of BWR's: Covered by Regulatory Guide 1.96.

2. Fuel Densification: Covered by 10 CFR 50 Appendix K plus case-by-case review of vendor fuel models.

3. Rod Sequence Control Systems: Covered by NRC Staff Review and Approval of NEDO-10527 and Presentation to ACRS.

4. Seismic Category I Requirements for Auxiliary Systems: Covered by Regulatory Guides 1.26 and 1.29.

Group II - Resolution Pending

1. Turbine Missiles: Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problems.*

2. Effective Operation of Containment Sprays in a LOCA: Extensive documentation in topical reports. Review and evaluation are required.

3. Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock: Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data from the Heavy Section Steel Technology Program.

**4. Instruments to detect (severe) fuel failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen. Item ID covers limited failures. More work is required for the severe failure case to establish instrumentation criteria.

#5A. Monitoring for Loose Parts Inside the Reactor Pressure Vessel: State-of-the-Art results appear promising and some equipment has been installed.

#5B. Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel: Neutron Noise Analysis has been successful in detecting vibration of some components, however, additional work may be required concerning systems for detecting vibration in other components within the Reactor Pressure Vessel.

#6. Common Mode Failures: This heading covers a multiplicity of diverse components for which requirements should be established. Due to their diversity the ACRS feels that specific items should be separated into subsets under the general heading of common mode failures;

6A - Reactor Scram Systems
6B - Alternating Current Sources onsite and offsite
6C - Direct Current Systems

The above items are easily identified, other specific items may be added to this listing in the future.

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*Regulatory Guide is in preparation.

**Identified in the Committee's Report of April 16, 1976 as "Instruments to Detect Fuel Failures."

#These are a separation of items included under the same numbers in previous reports.

Group IE - Generic Items Resolved Since February 24, 1977

1. Control Rod Drop Accident (BWRs): Resolved through NRC review and documentation establishing such an event as not having severe consequences (Memorandum for M. Bender, Chairman ACRS, from Denwood F. Ross, Jr., Assistant Director for Reactor Safety, DSS, dated February 11, 1977.)

2. Rupture of High Pressure Lines Outside Containment: Resolved by positions in Standard Review Plan 3.6.1 and 3.6.2.

3. Isolation of Low Pressure from High Pressure Systems: Resolved by positions in Standard Review Plan 5.4.7.

Group IIA - Resolution Pending - Items Since December 18, 1972

1. Pressure in Containment Following LOCA: Further criteria and methods are needed to better evaluate local dynamic pressures in a LOCA to establish more definitive design margins.

2. Control Rod Drop Accident (BWRs): Calculations indicate that the reactivity response differs from earlier values. New analyses are required, including three-dimensional effects.

3. Ice Condenser Containments: Additional analyses are required to establish response during a LOCA, and to establish design margins.

4. Rupture of High Pressure Lines Outside Containment: The possibility exists that failure of a high pressure line such as a steam pipe can prevent operation of critical safety components.

5. PWR Pump Overspeed During a LOCA: Problem arises in similar manner to that of BWRs (Item 8 Group II).

6. Isolation of Low Pressure From High Pressure Systems: Assurance required that low pressure systems cannot inadvertently be interconnected with a high pressure system leading to failure. There are potential interaction problems between Class 1 and Class 2 or Class 3 pressure connections.

7. Steam Generator Tube Leakage: Partially resolved by issue of Regulatory Guide 1.83 which addresses the concern from a preventative point of view.

8. ACRS/NRC Periodic 10-Year Review of all Power Reactors: A more effective, continuous alternative approach to periodic reviews is being proposed. Pending ACRS review, this item is still considered unresolved.

Group II Continued

7. Behavior of Reactor Fuel Under Abnormal Conditions: This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

8. BWR Recirculation Pump Overspeed During LOCA: Decision required by ACRS-NRC Staff.

9. The Advisability of Seismic Scram: Further studies required to establish need.

10. Emergency Core Cooling System Capability for Future Plants: Partially resolved by amendments to 10 CFR 50 [50.34(a)(4), 50.34(b)(4), 50.46, and Appendix K]. LOCA evaluation model complete. ACRS feels new cooling approaches should be explored.

Group IIC - Resolution Pending - Items Added Since March 12, 1975

1. Locking Out of ECCS Power Operated Valves: The Committee suggests that further attention be given to procedures involving locking out electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS.

2. Fire Protection: The Committee recommends review of design features intended to prevent the occurrence of damaging fires and to minimize the consequences to safety-related equipment should a fire occur.

3. Design Features to Control Sabotage: Attention should be given to aspects of design that could improve plant security.

4. Decontamination and Decommissioning of Reactors: Specific plans should be developed, including definitive codes and standards covering plant decommissioning. Also experience should be gained in reactor decontamination so that such information is available when needed.

5. Vessel Support Structures: Questions that have arisen concerning the loads on pressure vessel support structures due to certain postulated loss-of-coolant accidents should be resolved.

6. Water Hammer: Several cases of water slugging or water hammer have occurred in both PWRs and BWRs. Corrective measures should be taken to minimize such events.

7. Maintenance and Inspection of Plants: Provisions should be included in the design of future plants which anticipate the maintenance, inspection and operational needs of the plant throughout its service life.

8. Behavior of BWR Mark I Containments: Various aspects relevant to the BWR Mark I Containment should be resolved. Included are such items as relief valve restraint, control of local dynamic loads in the torus, vent clearing and establishment of torus water temperature limits during a LOCA. This is an extension of Item 1 in Group IIA.

Group IIB - Resolution Pending - Items Added Since February 13, 1974

1. Hybrid Reactor Protection System: Systems should be qualified for reliability, particularly through in situ tests and under various environmental conditions, prior to use in reactor system.

2. Qualification of new fuel geometries: The 16x16, 17x17 PWR and 8x8 BWR fuels should undergo testing to meet Item 2 in Group IC and Item 7 in Group II.

3. Behavior of BWR Mark III Containments: Various aspects, including vent clearing, vent coolant interaction, pool swell, pool stratification, pressure loads and flow bypass should be resolved. This is an extension of Item 1 in Group IIA.

4. Stress Corrosion Cracking in BWR Piping: Several failures have occurred in operating BWRs. The ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution and extensive programs are underway by industry, ERDA, and NRC.

II-3 - Possible Failure Of Pressure Vessel Post-LOCA By Thermal Shock

Earlier nuclear reactor pressure vessels subjected to fluences of $1\text{-}4 \times 10^{19}$ nvt, which are anticipated in the last 20 years of a 40-year life, may suffer severe radiation damage denoted by a pronounced shift in impact transition temperature at the inner surface. There will be a damage gradient which decreases sharply, so that the properties halfway through the wall are essentially those of the as-fabricated material. If a LOCA occurs near end-of-life, the injection of cold water on the region of degraded properties may initiate and propagate a crack because of high local stresses near the surface. Analytic procedures indicate the stresses drop rapidly with distance through the wall so the flaw should not propagate beyond some limiting point. The lack of experimental evidence and the relative width of the error band in the analytic results are such that some experiments are required to validate the analytic model. These are planned under the HSST program.

II-1 - Turbine Missiles

Turbine failures for the past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discuses the problem.

Three issues require answers to resolve the turbine missile problem: (1) The first relates to the appropriate failure probability value; based on historical failures the probability is about 10^{-4}. Industry predicts a much lower failure probability based on improvements in materials and design. To date the ACRS has accepted the more conservative value; (2) The second issue is strongly dependent on turbine orientation with respect to critical safety structures. Strike probabilities from high angle missiles are acceptably low for single units and may be acceptable for multi-unit plants, depending on plant layout; however, lower angle missiles with non-optimum (tangential) turbine orientation have unacceptably high strike probabilities; (3) The third issue is one of penetration and damage of structures housed in the containment. The limited experimental data pertaining to penetration of large irregularly shaped missiles are not sufficient to determine structural response to impingement of turbine disc segments. Most missile penetration formulas are not relevant to this case. Some experiments with irregular missiles might resolve this issue, particularly for older plants with non-optimum turbine orientations.

IIC-7 - Behavior Of BWR Mark I Containments

Recent tests on the BWR Mark I Containment design revealed phenomena not anticipated on the basis of earlier tests where pressure loads were imposed by insertion of air. Specific problems somewhat comparable to those under review for the Mark III Containment, include relief valve discharge, pipe restraints in the torus, local dynamic loads on the torus, vent clearing, and influence of torus temperature on the LOCA.

Ongoing experiments are expected to develop the necessary data to confirm the adequacy of the existing design or to permit necessary modifications.

IIC-2 - Design Features To Control Sabotage

Considerable attention has been devoted to control of industrial sabotage of nuclear power plants, particularly with regard to control of unauthorized access, and potential modes of sabotage by individuals or groups external to the operating organization. The ACRS believes that deliberate attention should be given to aspects of design that could improve plant security. With the emphasis being placed on standardized plant designs, it becomes especially important to introduce design measures that could protect against industrial sabotage, or mitigate the consequences thereof.

industry seemed to be adequate for the most part, there did not exist a code of accepted practice. Hence, the matter was identified as a generic item which again was to be resolved by issuance of a regulatory guide.

Item 14 of Group I on operational stability of jet pumps represented nothing more than a possible operational problem for BWRs with jet pumps. The question was raised when the first jet pump BWR (Dresden 2) was reviewed for construction. The designer was confident that no such difficultis would arise. However, to be conservative, the ACRS flagged the possibility as one to be alert for. Operational experience proved the designers to be correct.

Item 4 in Group ID, fire protection, was made a generic item after occurrence of the fire at Browns Ferry Units 1 and 2. Actually, the matter of how to deal with protection against fires had been a thorny issue for many years. A significant cable tray fire had occurred earlier at San Onofre 1, due to thermal overload (a design deficiency which was corrected at San Onofre and watched for routinely at all reactors thereafter). The difficulties in postulating the causes of realistic serious fires and in assessing their consequences left the subject in an ill-defined state. And while progress had been made in improving protection against fires via criteria on separation of redundant systems, progress had been slow. Opinions varied as to the adequacy of existing approaches, as the minutes of an ACRS Subcommittee meeting on Fire Protection, held on January 19, 1973, show. Ironically, representatives of TVA came to that meeting and discussed Browns Ferry Units 1 and 2 to provide background as to the adequacy of existing practice.

After the Browns Ferry fire, the Regulatory Staff mobilized a major effort on the matter of fires and fire protection. Identification of the matter as a generic item was "proforma". And its "resolution" came about with agreement on the pertinent Regulatory Guide. Actual implementation of the guide proceeded its adoption, in part. However, in this matter as in many, there are various possible approaches, each having its pros and cons, and differences of opinion have continued to arise with regard to acceptable detailed practice.

Item 1 in Group II on turbine missiles is a safety issue which was raised many times, beginning in 1965 or 1966, before it was identified as a generic issue. In some of the initial discussions on the subject, it was argued by reactor vendors that the existing design provided protection against missiles. Continued consideration indicated that this was not necessarily so for missiles generated at the design overspeed condition, and less so if higher energy missiles were generated by rutpure of the turbine at much higher speeds. A statistical analysis of existing turbine experience by ACRS member Bush indicated that historically the failure rate of turbines had been about 10^{-4} per turbine year. Reactor vendors argued that new turbines were of much higher quality and predicted very much lower failure rates.

IID-1 - Safety Related Interfaces Between Reactor Island And Balance-Of-Plant

Questions have been raised concerning both standardized balance-of-plant and nuclear steam supply systems on the one hand and custom-designed site-related structures and components on the other hand. The depth of detail required at the stage of Preliminary Design Approval may not be adequate for construction approval. Procedures for instituting quality assurance programs covering design, procurement, construction, and startup with emphasis on timely and appropriate interdisciplinary system analyses to assure functional compatibility across the interfaces as well as for other systems, are necessary to assure functional compatibility for the postulated design basis accident conditions.

Analytical studies indicated that the probability of serious damage to the reactor from turbine failure was reduced markedly if a "peninsular" arrangement, rather than a "tangential" geometric relationship existed between the turbine axis and the containment. With continued pressure by the ACRS, most new construction permit applicatons adopted the "peninsular" approach. Hence, while the matter has not been completely resolved, nor is there agreement that the 'tangential' layout used in earlier reactors poses a significant risk, this design change, as well as some improvements in turbine overspeed control, have taken place.

The generic items on ATWS and steamline breaks outside containment are discussed in detail in Chapters 4 and 3. They represent items which arose by other paths, and whose resolution (or lack thereof for ATWS) provide still other examples.

The Regulatory Staff, as a consequence of their more detailed review, developed a much longer list of safety related issues to be settled (NRC, 1978). In fact, it is a continuing list. As some matters are resolved, new ones arise, partly as a result of operating experience, partly from changes in design approach, and partly from the surfacing or resurfacing of questions about existing criteria, designs, etc.

An interesting perspective on generic items as their resolution relates to standardization of nuclear power plants was provided in 1978 by ACRS member Bender. (Bender, 1978). He distributed the unresolved generic items into five groups according to the nature of the item and the action needed to resolve it, and then discussed the different needs for resolution of each group and the possible impact of reactor standardization. Excerpts from this paper follow below.

> For the first group, the requirement is primarily to analyze the existing situation and sometimes perform supporting experimental work in order to establish whether the concerns arising from undefined consequences deserve attention. Final judgment is not available until the work is completed.
>
> For the second group, the interest is in providing warning of impending difficulty in order to allow opportunity for circumventing safety problems. This may require diagnostic capability through instrumentation or perhaps separate experimentation to provide a basis for using existing instrumentation as diagnostic tools.
>
> A third group applies to matters correctable by appropriate definition of operational limits intended to assure that the safety integrity of the system is not challenged excessively. Resolution of these may come through technical specifications or relatively minor design changes involving electrical interlocks and operational trips.
>
> The fourth group addresses materials degradation where the ultimate life of the installation is uncertain because

degradation controls are not well understood. Since the degradation is sensitive to lapsed time, the corrective measures are not normally of immediate concern, but continuing effort is anticipated to establish an understanding of the problem in order that safety controls can be exercised before public safety becomes jeopardized by the potential consequences of accidents related to the degradation.

The fifth group is associated with plant configuration including equipment arrangements, redundancy provisions, control schemes, physical separation of vulnerable safety features, and structural strength of the physical plant. The matters in this fifth category usually must be preestablished during design if they are not to cause costly backfit actions.

These five groupings provide a characterization different from that which the ACRS has used to address the progress toward resolution of safety issues. You can judge the resolution status from the most recent ACRS report on generic safety matters dated November 15, 1977. What I will attempt to do here is consider how each of these groupings interacts with standardization.

Engineering Analysis

The fact that a generic safety issue has been listed is evidence that one or more knowledgeable people are uncertain as to whether attention is needed to minimize threats to public safety. It often turns out that the uncertainty is a matter of risk judgment influenced by the state of knowledge. Hence, improvement in the understanding can eliminate the concern. A number of matters have been dealt with in this way, including fuel densification, ECCS capability, control rod ejection accidents, and containment resistance to LOCA loads. The issue may arise because procedures for analyzing the events have an inadequate relation to the safety circumstances and do not provide sufficient basis for safety judgment. The result of refined analysis in some cases leads to engineering changes, but in most cases affirms that practices selected arbitrarily have sufficient margin to satisfy the design needs.

Accident Diagnostics

The second grouping is directed to accident diagnostics. The interest is in determining whether a safety problem exists and making certain that there is enough information provided to address the problem. Two types of diagnostic capability are involved. The first type is directed to matters where damaging conditions exist or have occurred and there is a need to know the extent of the damage. The vibration detection devices for measuring in-core vibration of loose parts and the physical movement of the core structure are representative of this type. Provision of strong motion seismic instrumentation, in-service inspection capability for the coolant system pressure boundary and leak detection features of operating plants are similarly of interest. They indicate whether the installation is performing as intended by design and provide a basis for affirming the effectiveness of the quality assurance program.

The second type is intended to address the unknown elements of public safety. No one can say with absolute assurance that all accidents are known and there is a need to determine the nature of unexpected occurrences. The need for instrumentation to diagnose accident symptoms in order to follow the course of an unexpected accident has been identified by the ACRS as a generic safety issue providing confidence that contingent public safety actions to deal with unexpected events can be implemented promptly. In order to make such provision, some diagnostic approach must be identified. For example, the extent of fuel melting, if it occurs, might be determined by measuring the energy spectrum of radioactivity at one or more points within the coolant system so that specific radionuclides peculiar to such accidents could be detected. If the same activity were released to the containment system, detection devices within the containment might serve a comparable purpose. The need for such devices has not been adequately evaluated. A notable circumstance of the Browns Ferry fire was that the almost complete loss of instrumentation in the late stages of the fire left the operator with the uncomfortable obligation to guess at the safety adequacy of the installation. In a comparable circumstance at Ft. St. Vrain, the nature of the instrumentation made it impractical for the operator to distinguish the level of difficulty associated with a minor coolant system leak, causing unnecessary national public alarm. The actual public safety need for such diagnostics is not likely to be frequent and hopefully, is infrequent enough to justify some arguments that the capability is unnecessary. Nevertheless, the two cited incidents indicate the appropriateness of requiring such capability for public safety purposes. No standardized designs yet include complete treatment of these generic safety matters and the implementation actions associated with the resolution of the issue remains in doubt. The fault is not attributable to the standardization process but only to the lethargy of those promulgating standardized design approaches.

Materials Degradation

One of the bugaboos of nuclear power plant safety is materials degradation that may ultimately lead to operational demands on the engineered safety features. Of the matters which have drawn public attention, this is the dominant category. The defense-in-depth principle was advanced mainly in response to the anticipation that degradation may not be wholly predictable and that safety can be best assured by multiple lines of lines of defense. The principle is valid, however, only if the degradation rates are controllable and difficulties are anticipated. We are not expecting nuclear power plants to match the performance of the "one horse shay". Some portions of the system will degrade at different rates from others and degradation may occur in ways which we do not anticipate. In recent years, questions have been raised about radiation damage to pressure vessels and its effect on gross rupture, to stress corrosion of stainless steel components, to environmental damage to important instrumentation and power supplies by LOCA environments, and to the general problems of aging and related deterioration factors from the steady state working environment of safety related portions of the plant. Thus, many of the generic safety issues involve an understanding and control of materials degradation that neither analysis nor short range experimental work will provide. In relating such matters to standardization practice, one concern is that the standardization action will establish a position based on status quo conditions that may not allow for longer term difficulties.

If, for example, the selection of newer materials would eliminate problems of stress corrosion in boiling water reactor systems, it would hardly be good regulatory practice or even good engineering judgment to preclude the use of new materials by standardization restrictions. The evolution of pressure vessel materials practice has finally reached a stage where there is no longer concern in newer plants about pressure vessel radiation damage as a cause of pressure vessel rupture. If the thrust of standardization had begun prior to the implementation of this improved pressure vessel materials practice and applicants had been inhibited by standardization from making materials improvements, the value of the standardization process would be doubtful. Many people are still concerned about whether standardization will create such inhibitions since many manufacturing organizations would find difficulty in obtaining management support for materials improvements once the current practices were accepted if there were no profit incentive.

Using the recent experience with steam generator tubing as an example, one must conclude that not all of the materials degradation questions are known for nuclear power systems. The interactive effects of aging, radiation, temperature, humidity, chemistry, and intrusive containments all suggest caution in allowing standardization to interfere with improving the ability of nuclear power systems to face adverse circumstances. The standardization mode of regulation is certain to foster preference for the status quo and to the degree that it is overly restrictive, the standardization process could interfere with the resolution of such generic safety matters.

Configuration Changes

Undoubtedly, the most difficult group of generic matters to be

> treated during standardization are those requiring redesign. Some organizations have chosen to be innovative in selecting the standard design for which a PDA would be sought. It is reassuring to see that the concepts pursued by the TVA and Duke groups have both incorporated provisions for better separation of redundant safety related features contributing to both improved fire protection and sabotage resistance. The SWESSAR concept appears to be equally creative in providing a design that eases the problem of safety feature separation. Although innovative design decisions are not without peril because what appears straightforward during initial design may prove to be complex when the design detail is ultimately developed, deliberate decisions to improve safety features in the standardized designs will go a long way toward resolving generic safety concerns.

Bender went on to discuss the differing points of view on how to address, if at all, the long-standing ACRS recommendation for improvements in ECCS system performance and reliability. He also directly addressed the question of Class nine accidents, as follows:

> Ever since the completion of the WASH-1400 study, the arguments have ceased about whether core melts are possible. The discussion has shifted to the question of acceptable public consequences from core melt. Studies within the regulatory organization are addressing alternatives such as filtered containment as a means of providing ultimate containment pressure relief or other ways of limiting public exposure from radionuclides if the loss-of-cooling features should prove less effective than anticipated. None of these approaches negate the argument that the likelihood of a core melt is low and may for most purposes be ignored as a public safety issue. The question is only whether there are important values in further improving the ability to handle the ultimate low probability accident. If standardization interferes with the opportunities for such improvement, it would be hard to defend its value.

GENERIC SAFETY MATTERS RELATED TO PLANT CONFIGURATION CHANGES*

1.	FUEL STORAGE POOL DESIGN	1-6
2.	PROTECTION AGAINST PUMP FLY WHEEL MISSILES	1-7
3.	INSERVICE INSPECTION OR REACTOR COOLANT SYSTEM BOUNDARY	1-10
4.	SEISMIC DESIGN OF STEAM LINES	1-22
5.	PROTECTION AGAINST PIPE WHIP	1A-3
6.	ANTICIPATED TRANSIENTS WITHOUT SCRAM	1A-4
7.	EMERGENCY POWER FOR TWO OR MORE REACTORS	1B-5
8.	ROD SEQUENCY CONTROL SYSTEMS	1C-3
9.	FIRE PROTECTION	1D-4
10.	RUPTURE OF HIGH PRESSURE LINES OUTSIDE CONTAINMENT	1E-2
11.	TURBINE MISSILES	11-1
12.	COMMON MODE FAILURES	11-6
13.	BWR RECIRCULATING PUMP OVERSPEED DURING LOCA	11-8
14.	ECCS CAPABILITY FOR FUTURE PLANTS	11-10
15.	PWR PUMP OVERSPEED DURING LOCA	11A-2
16.	BEHAVIOR OF BWR MARK III CONTAINMENTS	11B-3
17.	DESIGN FEATURES TO CONTROL SABOTAGE	11C-2
18.	MAINTENANCE AND INSPECTION OF PLANTS	11C-6

*Roman numerals indicate ACRS Designation (11/77 Report)

GENERIC SAFETY PROBLEMS POTENTIALLY RESOLVABLE BY ANALYSIS AND EXPERIMENTATION

	ITEM	ACRS DESIGNATION (11/77 REPORT)
1.	TURBINE MISSILES	11-1
2.	POST LOCA VESSEL THERMAL SHOCK	11-3
3.	REACTOR FUEL BEHAVIOR UNDER ABNORMAL CONDITIONS	11-7
4.	NON-RANDOM FAILURES	11-6
5.	BWR PUMP OVERSPEED	11-8
6.	ADVISABILITY OF SEISMIC SCRAM	11-9
7.	PWR PUMP OVERSPEED	11A-2
8.	ICE CONDENSER CONTAINMENT	11A-1
9.	COMPUTER REACTOR PROTECTION SYSTEM	11B-1
10.	BEHAVIOR OF MARK III CONTAINMENT	11B-3
11.	LOCKING OUT OF ECCS POWER OPERATED VALVES	11C-1
12.	DECONTAMINATION OF REACTORS	11C-3A
13.	VESSEL SUPPORT STRUCTURES	11C-4
14.	BEHAVIOR OF MARK I CONTAINMENTS	11C-7
15.	SOIL STRUCTURE INTERACTION	11E-1
16.	QUALIFICATION OF NEW FUEL GEOMETRIES	11B-2

ACTIONS TO RESOLVE GENERIC SAFETY ISSUES

1. PERFORM ANALYTICAL AND/OR EXPERIMENTAL INVESTIGATIONS TO ESTABLISH SAFETY MARGINS.
2. PROVIDE FOR ACCIDENT DIAGNOSTICS TO WARN OF SAFETY JEOPARDY AND EVALUATE ACCIDENT PROGRESSION.
3. APPLY OPERATIONAL CONSTRAINTS THROUGH TECH. SPECS. TO ASSURE ACCEPTABLE PUBLIC SAFETY.
4. ESTABLISH MATERIALS DEGRADATION RATES BY OPERATIONAL SURVEILLANCE INCLUDING TESTS AND INSPECTION.
5. MODIFY OR ALTER DESIGN CONFIGURATIONS TO ATTAIN SAFETY IMPROVEMENTS.

GENERIC SAFETY PROBLEMS ADDRESSED BY TECHNICAL SPECIFICATIONS*

1.	A-C SYSTEMS (PARTIAL)	11-6B
2.	DIRECT CURRENT SYSTEMS (PARTIAL)	11-6C
3.	STEAM GENERATOR TUBE LEAKAGE (PARTIAL)	11-A3
4.	ISOLATION OF HIGH PRESSURE FROM LOW PRESSURE SYSTEMS (PARTIAL)	1-E3

*Roman numerals indicate ACRS Designation (11/77 Report)

GENERIC SAFETY PROBLEMS REQUIRING DIAGNOSTIC CAPABILITY*

1. INSTRUMENTS TO DETECT SEVERE FUEL FAILURE — II-4
2. LOOSE PARTS MONITORING — II-5A
3. EXCESSIVE VIBRATION MONITORING — II-5B
4. INSTRUMENTATION TO FOLLOW THE COURSE OF ACCIDENTS — IID-2

*Roman numerals indicate ACRS designation (11/77 report).

GENERIC SAFETY MATTERS RELATED TO MATERIALS DEGRADATION*

1. STEAM GENERATOR TUBE LEAKAGE — IIA-3
2. STRESS CORROSION CRACKING IN BWR PIPING — IIB-4
3. WATER HAMMER — IIC-5
4. DECONTAMINATION OF REACTORS — IIC-3A
5. MAINTENANCE AND INSPECTION OF PLANTS — IIC-6
6. ASSURANCE OF CONTINUOUS LONG TERM CAPABILITY OF HERMETIC SEALS ON INSTRUMENTATION AND ELECTRICAL EQUIPMENT — IID-2

*Roman numerals indicate ACRS Designation (11/77 Report)

6.2 Some Sidelights on LOCA-ECCS

The matter of loss-of-coolant accidents (LOCA) and emergency core cooling systems (ECCS) has probably been the major topic of public discussion in regard to light water reactor safety. And it is probably the topic which has received the greatest attention in licensing reviews. It has certainly received the bulk of the resources expended in nuclear reactor safety research. And it was the subject of a very long public hearing conducted by the Atomic Energy Commission in connection with the establishment of Acceptance Criteria for ECCS in 1972-73.

We shall not try to examine in detail this long and extensive history of LOCA-ECCS or even to cover all major developments up to the present time. Rather, we shall emphasize some of the earlier developments, provide a few selected sidelights, and look at some of the trends.

The first commercial LWRs, such as the Indian Point 1 PWR and the Dresden 1 BWR had very limited ECCS capability by current standards. As originally built, Dresden 1 had a low capacity, diesel-driven, emergency feed pump that took water from the condensate storage tank to the primary system steam drum (at full system pressure).

Indian Point 1 had its three normal makeup feed pumps. Analysis indicated that these three pumps could handle up to a two inch diameter break below the core or a four inch diameter break above the core.

The Connecticut Yankee and San Onofre PWRs, approved for construction in 1963-64, included substantially augmented ECCS, intended to cope with rupture of the largest pipe connected to the primary cooling system. On-site AC power was not adequate to power the ECCS in the originally designed plants. Oyster Creek, Nine Mile Point and Dresden 2, large BWRs approved for construction about 1965, each had an ECCS comprised of two core spray systems, with on-site AC power capable of operating at least one core spray. And the Brookwood PWR and Millstone Point Unit 1 BWR, approved in early 1966, were similar to their immediate predecessors with regard to ECCS design.

During the period through early 1966, the Applicants for LWR construction permits presented rather limited information pertaining to the performance capability of the proposed ECCS or the methods employed for performance analysis. The AEC Regualtory Staff itself did not possess the capability to evaluate the claimed performance. While the Regulatory Staff and ACRS were beginning to request that some experimental confirmation of ECCS be obtained, as is recorded in the ACRS minutes of the review of Oyster Creek in 1964, ECCS was not treated as a vital safeguard. The ACRS letter report to AEC Chairman Seaborg of November 18, 1964 on Engineered Safeguards summarizes the regulatory attitude of that era. This letter, which was prepared in response to a Commission request for a review of how engineered safety features were being substituted for distance in the siting of reactors, places primary emphasis on containment and on features to clean up the containment atmosphere, assuming the postulated fission product release of 10 CFR Part 100. While not crediting ECCS with preventing core melt, and hence the occurrence of so large a fission product release, the ACRS report of November 18 did not dismiss ECCS as unnecessary. The ACRS summary on ECCS are reproduced below.

> Core spray and safety injection systems cannot be relied upon as the sole engineered safeguards in a nuclear power plant. Nevertheless, prevention of core melting after an unlikely loss of primary coolant would greatly reduce the exposure of the public. Thus, the inclusion of a reactor core fission product heat removal system as an engineered safeguard is usually essential.

Another example of the relatively modest emphasis on ECCS in this period is available from the proposed "General Design Crieria for Reactors," published for comment by the AEC in November, 1965. Relatively little is stated concernng ECCS design in the 1965 Draft.

Then a revolution in LWR safety occurred in 1966. As has been discussed in detail in Section 2.8 entitled "Pressure Vessels, ECCS and the "China Syndrome" and Section 2.9, "China Syndrome, Part 1," the diret correlation between core melt and a loss of containment integrity was recognized and strongly influenced ACRS review of construction permit applications for Dresden 3 and Indian Point 2 in 1966. Emphasis shifted from containment to the prevention of core melt; and the LOCA received primary attention as the most probable source of an accident which might lead to core melt. The ACRS reports of August 16, 1966, on Dresden 3 and Indian Point 2 recommended much greater emphasis on the prevention of a LOCA, and greatly improved capability of the ECCS. Actually, when the core melt issue came to a head, General Electric modified its previously proposed ECCS for Dresden 3 and added a separate core flooding system to the two core spray systems. General Electric argued it also had redundant and diverse systems for small breaks in that its single high-pressure core injection system was backed up by an automatic depressurization system, which could open a set of valves, thereby reducing the primary system pressure sufficiently to permit the core spray or core flooding systems to function.

Westinghouse had not proposed an improved ECC system at the time of the August 16, 1966 ACRS report on Indian Point 2 and Dresden 3, and only preliminary examination of the newly proposed General Electric system had been possible. Hence, the ACRS had recommended that both the ACRS and the Regulatory Staff review the ECCS designs for these two plants before irrevocable commitments were made in fabrication and construction.

Actually, Turkey Point Nuclear Units 3 and 4 were the Westinghouse PWRs next in line in the regulatory review process, and in the fall of 1966, Westinghouse proposed a modified ECCS, employing accumulators containing large amounts of water under gas pressure, plus a system of pumps. This new ECC system, while not employing diverse principles (flooding and core spray as in the BWR), was supposed to be redundant, and was to meet the criterion of no clad melt for the rupture of any primary system pipe.

As can be seen from the excerpts from meeting minutes and the ACRS letter on Turkey Point, this approach was judged to be generally acceptable, and was implemented in similar although not identical fashion on all succeeding PWRs including those sold by Babcock and Wilcox and Combusiton Engineering.

> Mr. Beckjord reported regarding the engineered safeguards. Air coolers and containment sprays are to be provided to cool the containment building. Core cooling will be provided by the high head and low head safety injection systems and an accumulator system. The containment spray systems for Units 3 and 4 are not interconnected. The high head system is similar to that for recent Westinghouse reactors, except that three high head safety injection pumps are shared by Units 3 and 4. The containment spray pumps are not shared. Mr. Palladino inquired regarding Westinghouse's philosophy concerning sharing of components. Westinghouse representatives reported that two containment spray pumps had been provided for each of the Turkey Point reactors because of convenience in layout of the facility. They stated, however, that three shared containment spray pumps might well be acceptable. Westinghouse believes that the shared high head safety injection system is adequate for any interaction which might occur between the two facilities. The containment is not, however, being designed for simultaneous accidents of the same type. The low head safety injection pumps function for shutdown heat removal and may be required by both facilities at the same time, whereas operation of the high head safety injection system is anticipated for use only during accident conditions and for a short period of time.
>
> Westinghouse described a new accumulator core flooding system that was being contemplated for addition to the Turkey Point reactors. This system would consist of three large storage tanks which would inject water into each of the three main coolant loops. The system would act sufficiently fast and would have sufficient flow rate to prevent significant melting of the fuel cladding in the event of a loss-of-coolant accident. The system would actuate if reactor pressure decreased to the range of 400 to 600 psi. There would be approximately 800 cubic feet of water in each accumulator. In the event of rupture of a main coolant loop, water would begin to be injected within 10 seconds following the rupture, and injection

of water into the vessel would be completed in 40 seconds. At the end of that time, the pressure vessel would be refilled. The flow rate of this new system would be approximately five times the flow rate of the low head safety injection system. Mr. Beckjord indicated he believed that the main issue relative to the adequacy of core cooling is not whether there will be some clad melting but whether the core will remain in place and intact so that it can be adequately cooled.

The refueling water storage tank has approximately a 20-minute storage capacity. The low head safety injection system pumps are connected to the containment sump through a heat removal system. An operator must be present in the control room to shift the system from the refueling water storage tank to the containment sump source of water.

Mr. Beckjord stated that additional core cooling was needed on a previous Westinghouse reactor (Indian Point 2). He said that an accumulator system would be added to this previous reactor as well as to the Turkey point reactors. Although the design is not complete, the design objective (i.e., to hold the core intact in the event of the maximum size break) has been established. Mr. Beckjord indicated he believed there would be no problem in meeting the design objective if less than 1% of the core were to melt.

It was pointed out that there is a single line through the containment wall from the residual heat exchangers, and the system is vulnerable to a single component failure at this point. There is also a single line from the high head safety injection system pumps through the containment wall.

If there were a hole in the reactor pressure vessel, approximately 20 minutes would be required to flood the bottom of the containment and raise the water level to the height of the center of the core. Westinghouse believes that, even if the core were to melt and fall to the bottom of the pressure level, the core would not melt through the bottom of the vessel. Mr. Beckjord stated that he knew no reason why the core would melt through but that he could not guarantee that it would not. Dr. Okrent inquired as to what design consideration led to the proposed degree of flooding of the containment. Mr. Beckjord stated that consideration is being given to revising the plant layout in future facilities to provide for flooding of the pressure vessel to a greater height.

Mr. Beckjord stated that the proposed accumulator system is to be installed in response to a new problem, i.e., maintaining the core in place following the occurrence of a double ended pipe break. He said that it is a static system and that nothing should go wrong with the system. There would be a small, insignificant amount of core melting and metal-water reaction during the 30-40 seconds after a main coolant line break before the core can be recovered.

After operation of the accumulator tanks has re-covered the reactor core following a pipe break, boil-off can be made up with the residual heat removal pumps, which can operate at 150 psig. One of these is expected to more than account for decay heat boil-off and the water level in the pressure vessel will continue to rise until the entire core is covered.

The cooling system has been modified so that the residual heat removal pumps now discharge into the coolant loop cold legs. The high pressure injection system discharges into two of the coolant loop hot legs and two of the cold legs. The charging pump outlet is connected with the third hot and cold leg. This modification allows the cooling system to keep water flowing through the core in the direction of any small break which might occur.

The latest information is that in the event of a 29-inch break the accumulators will cover the core mid-plane in 16 seconds and that the low-head pumps will completely submerge the core in 78 seconds.

3.3 Emergency Core Cooling System

The applicant has provided a complex of emergency core cooling engineered safeguards in order to mitigate the course of the thermal transient following a loss of coolant accident and to prevent the occurrence of a subsequent core meltdown. These are the accumulator system (discussed in section 2 of this report) and the high head and low head safety injection system (discussed in detail in section 4.1 of Report No. 1). These systems with a brief description of their capabilities are listed in Table 3.5.

Because of the rapidity of the thermal transient as presented above, the sole use of a pump system, especially with an extra built-in delay when operated by auxiliary diesel power, is undesirable. The applicant has consequently proposed the passive accumulator system which is capable of very rapid additions of borated coolant to the cold leg of each coolant loop. In the case of the 29" double-ended break all the water is discharged in the time period of 5 to 19 seconds after initiation of the break. For this break size the accumulators have discharged their contents completely before a pump system operating on auxiliary power could be started. For a 3 square foot break the accumulators discharge in the period 17 to 34 seconds after the break.

The accumulators are sized to re-cover half the core column with one of three units discharging through the break. Some uncertainty that the remaining flow will reach the core remains. It is conceivable that in the case of a cold leg break significant flow from the two remaining accumulators could be led out of the vessel through the inlet plenum. Analysis is continuing in this area.

One interesting aspect of these records from 1966 is that the Regulatory Staff at that time stated "It is conceivable that, in the case of a cold leg break, significant flow from the two remaining accumulators could be led out of the vessel through the inlet plenum". This slight concern rose with a vengeance some years later following a series of experiments on the Semi Scale Facility in Idaho in 1971.

Florida Power and Light

Engineered Safeguards

The modifications to the emergency cooling system were described. Westinghouse had been concerned about methods of preventing any significant amount of fuel melting over the entire spectrum of break sizes. The prime necessity was a large amount of flow in the first few seconds following a break. The accumulator tanks are now considered to be reliable way of achieving this. There are to be three tanks (one per coolant loop) which discharge into the reactor cold legs(actually, the residual heat removal system discharge lines). The remainder of the system is relatively unchanged.

The criterion to which the accumulator tanks have been designed is that a sufficient volume of water be delivered into the two unruptured loops in a short enough time after a break that no "significant number of rods" will melt. The fuel would then be kept in place and only a small amount of molten fuel would fall to the bottom of the vessel and would not coalesce. There was not a specific limitation on the amount of metal-water reaction, but this accrues as a result of meeting the objectives outlined. In fact, for Turkey Point, calculations indicate that no clad melting occurs and the Zr-H_2O reaction is limited to less than 1%.

The size of the accumulator tanks was fixed by determining the amount of water necessary; the pressure at which they are charged by deciding the flow rate required to re-cover the core. The tanks will be statically charged with gas at about 600 psig and thus represent a passive engineered safeguard. The gas pressure is used to drive the coolant into the core.

The line connecting the accumulators with the coolant loop cold legs will be about 10 inches in diameter and will contain a check valve in series with the already existing check valve in the residual heat removal lines. There will be two isolation valves, also in series, available to isolate each tank when the reactor is depressurized. These will be normally open during operation.

In conclusion, we believe the proposed emergency core cooling system can be designed to perform its intended function. There are four areas for which additional information and analysis are necessary. These include:

1) Thermal-mechanical stability of the core during the temperature transient.

2) Mechanical stability of the reactor internals during the blowdown transient.

Late in 1966, the Quad Cities BWRs were reviewed for construction permits and the ECCS approach previously outlined for Dresden 3 (and backfitted to Dresden 2 and Millstone Point 1) was judged to be generally acceptable.

The revised draft General Design Criteria published for comment in 1967 reflected the greatly increased emphasis on ECCS. Criterion No. 44 specifically stated:

> At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling shall be provided. Each ECCS and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe.

Some side effects potentially associated with the sudden, complete break (and possible offset) of a large pipe included pressure waves leading to blowdown forces inside the pressure vessel, and dynamic forces in those sub-compartments within the containment which house major components such as the steam generator. (The latter effect received renewed attention after safety studies on the subject were reported by German research groups in the late 1960's.)

The matter of whether large primary system pipes of high quality had a significant probability of gross rupture, and especially of essentially instantaneous rupture, has been controversial. Although some simulated tests involving large pre-existing cracks have led to complete piping failure, and there is some history from the past of the gross failure of large pipes in non-nuclear systems, there existed a school of thought that the sudden "double-ended guillotine break was an inappropriate basis for sizing ECCS and the other safety features associated with this postulated break. In connection with the review of the Browns Ferry 1 and 2 BWRs, which represented a large increase in power over the previously reviewed Dresden and Quad City reactors, the double-ended pipe break controversy arose within the ACRS. After a long, difficult review which saw many issues, new and old, discussed at great length, the ACRS report of March 14, 1967 on Browns Ferry not only saw the birth of the "asterisked items," (the forerunner of the "generic items"); the letter also included a dissent by member Hanauer, who was dissatisfied with the proposed emergency AC power, among other things. And, during Committee preparation of the report, ACRS member Bush at one point indicated he would also have additional remarks to the effect that the use of the double-ended pipe break was inappropriate and was leading to less safety. Bush did not actually attach his remarks to the final letter, when the ACRS agreed to initiate a generic study of the double-ended pipe break question. This matter was pursued by a special Subcommittee over the next year, with the conclusion that the double-ended pipe break should remain among the spectrum of pipe breaks to be analyzed for ECCS performance. Actually, the "dominating" break in ECCS analysis was not

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 18, 1967

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON TURKEY POINT NUCLEAR GENERATING UNITS NO. 3 AND NO. 4

Dear Dr. Seaborg:

At its eighty-first meeting, January 12-14, 1967, the Advisory Committee on Reactor Safeguards completed its review of the application of Florida Power and Light Company for authorization to construct Turkey Point Nuclear Generating Units No. 3 and No. 4. This project had previously been considered at the seventy-ninth meeting of the Committee, November 10-12, 1966, and at Subcommittee meetings on September 7, November 9, and December 7, 1966, and January 7, 1967. Representatives of the Committee visited the site on December 16, 1966. During its review, the Committee had the benefit of discussions with representatives of Florida Power and Light Company, Westinghouse Electric Corporation, Bechtel Corporation, and the AEC Regulatory Staff and its consultants. The Committee also had the benefit of the documents listed.

The Turkey Point Units are to be located in Dade County, Florida, on the west shore of Biscayne Bay approximately 25 miles south of Miami. Each unit includes a pressurized water reactor to be operated at an initial maximum power level of 2097 MWt but designed to operate ultimately at a maximum power level of 2300 MWt.

The containment structure for each unit consists of a steel-lined concrete shell with shallow spherical dome and flat slab base. The shell and dome are fully prestressed, with steel tendon systems carrying the principal loads. Provisions are made for in-service inspectability, replaceability, and corrosion control of the tendons over the lifetime of the structure.

The complex of emergency core cooling systems provided for each unit includes a high head safety injection system and a low head residual heat removal system with an accumulator subsystem. The accumulators are capable of very rapid addition of borated water to the reactor in the unlikely event of a large scale loss-of-coolant accident, and increase the time

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margin available for initiation of emergency cooling flow by pumping. The high head safety injection system pumps (three) are shared by Units 3 and 4. These systems appear to be adequate for the Turkey Point reactors. The AEC Regulatory Staff should review carefully the final design of the emergency core cooling systems, including the analyses of system characteristics and the effects of blowdown on reactor internals.

The reactor is calculated to have a positive moderator coefficient during a portion of core life. The applicant will give careful attention to the influence of positive coefficients on reactor transients, including the loss-of-coolant accident, rapid control rod motion, and xenon oscillations. If necessary, the moderator coefficient will be modified by the addition of solid burnable poison to the core. The Committee feels that the Regulatory Staff should follow closely the status of this aspect of design. The ACRS would like to be kept informed with respect to both the emergency core cooling and the moderator coefficient studies.

The frequency and intensity of hurricanes at the Turkey Point site present problems of potential flooding and wind damage. The applicant has made preliminary estimates of wind forces, water levels, and wave heights associated with the maximum probable hurricane against which vital components of the plant are to be protected. Remaining questions on the appropriate degree of protection will be resolved between the applicant and the AEC Regulatory Staff.

The applicant desires to continue uninterrupted operation of the reactor in the event one of two or more redundant active components in an engineered safeguard system becomes temporarily inoperable. The associated operable components would be maintained in continuous operation, however, until the inoperable component is again ready for service. The Committee feels that this matter may require review at the time of application for an operating license.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactors can be built at the Turkey Point site with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Mr. Harold Etherington did not participate in review of the Turkey Point Nuclear Generating Units No. 3 and No. 4.

Sincerely yours,

/s/ by N. J. Palladino

N. J. Palladino
Chairman

References Attached.

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always the largest break. And interestingly, during the subcommittee review, representatives of industry generally agreed that the double-ended pipe break was applicable (although thre were differences of opinion concerning the time for actual rupture which influences blowdown forces in the reactor vessel for a PWR).

The ACRS report on Browns Ferry 1 and 2 are reproduced on the following pages.

Among other things, the ACRS raised questions concerning the potential consequences of fuel melting at power, and indicated strong reservations concerning the high fuel damage threshold limit proposed by General Electric. And the ACRS specifically identified the possibility that derating of the reactors might be necessary if important matters were not resolved satisfactorily. During the same time period in which the ACRS was completing its review of Browns Ferry, the final report of the Task Force on Emergency Core Cooling was received. (The detailed conclusions given in that report are presented in Sectrion 2.13 entitled "Chinese Syndrome Part 2".)

To a considerable extent, the Task Force report states the nuclear industry position of that time with regard to LOCA-ECCS. It was relatively optimistic that methods and knowledge were in hand for design and analysis of ECCS to cope with the full spectrum of pipe break sizes, and that suitable criteria existed.

However, with the passing months and years a variety of matters not anticipated in the Task Force report arose. One of the first was the observations by Ivins, et al. at Argonne that Zircaloy clad exposed to LOCA-like conditions and reaching peak temperatures in the vicinity of 2500°F (well below the Zircaloy melting point of 3310°F) embrittled and ruptured, or even shattered on cooling down. This threatened the integrity of the core geometry and hence its continued coolability. Therefore, instead of the criterion of no (or very little) clad melt, which had been proposed by the vendors and had been accepted for some months, a much lower limit on peak clad temperature was indicated, somewhere around 2200-2500°F. This change was formally recognized in the ACRS letter of April 29, 1968, on Surry Units 1 and 2, among others.

Prior to 1966, the Regulatory Staff had essentially no in-house (or consultant) capability to analyze the LOCA. Following the Dresden 3 and Indian Point 2 cases in 1966 and the ACRS safety research letter of October 12, 1966, the AEC began to institute a strong safety research program on LOCA-ECCS experiment and analysis, and a beginning was made on forming a LOCA analysis group among the Staff. The vendors all instituted substantial efforts on the development of LOCA-ECCS codes for their individual reactors, as well as limited experiments.

When Indian Point 2 was reviewed for an operating license by the ACRS in September, 1970, fairly detailed discussions of the anticipated ECCS behavior were held. The reactor vendor and the Regulatory Staff each expressed confidence in the acceptability of ECCS performance and their understanding of ECCS function. Following blow-down of the primary system from a postulated large pipe break, core reflooding rates of 5-10 inches per second were predicted, and peak clad temperatures of <u>about</u>

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C.

March 14, 1967

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON BROWNS FERRY NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its eighty-third meeting, March 9-11, 1967, the Advisory Committee on Reactor Safeguards completed its review of the application of the Tennessee Valley Authority for authorization to construct Browns Ferry Nuclear Power Station Units No. 1 and No. 2. This project was previously considered at the eighty-first and eighty-second meetings of the Committee, January 12-14, 1967 and February 9-11, 1967, respectively, at a special meeting on February 28, 1967, and at subcommittee meetings on November 26, 1966, January 4-5, and January 28, 1967. Representatives of the Committee visited the site on February 27, 1967. During its review, the Committee had the benefit of discussions with representatives of the Tennessee Valley Authority, General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Browns Ferry Units are to be located in Limestone County, Alabama, on the shore of Wheeler Lake approximately 30 miles west of Huntsville. Each Unit includes a boiling water reactor to be operated at a maximum power level of 3293 MWt, the highest power level for any reactor reviewed for a construction permit to date. The average core power density is about 40 percent higher than for the previously reviewed Quad-Cities boiling water reactors. The increase is achieved by flattening the power density distribution and employing an approximately 20 percent higher fuel element maximum linear heat rate. The margins between thermal operating limits and fuel element damage limits are thereby reduced. In relation to margin on critical heat flux, the applicant uses new heat transfer correlations developed from recent experimental data.

The complex of emergency core cooling systems for Browns Ferry is similar to that proposed for the Quad-Cities reactors. Each reactor is provided with a high pressure coolant injection system; a low pressure coolant injection, or flooding, system; and two core spray systems. Because of the higher core power density and power level, substantial increases have been made in the flooding system and core spray system capacities. The Committee feels that the emergency core cooling systems proposed have a high probability of preventing core meltdown in the unlikely event of a loss-of-coolant accident. It notes, however, that although calculated peak fuel temperatures in such an accident are similar to those for the Quad-Cities reactors, the calculated number of fuel elements reaching undesirably high temperatures is greater. Also, the time margin available for actuation of the systems is less. Because of these factors and the importance of the effective functioning of emergency core cooling systems, the Committee believes the adequacy of these systems should be further corroborated by the following two measures:

1. Analysis indicates that a large fraction of the reactor fuel elements may be expected to fail in certain loss-of-coolant accidents. The applicant states that the principal mode of failure is expected to be by localized perforation of the clad, and that damage within the fuel assembly of such nature or extent as to interfere with heat removal sufficiently to cause clad melting would not occur. The Committee believes that additional evidence, both analytical and experimental, is needed and should be obtained to demonstrate that this model is adequately conservative for the power density and fuel burnup proposed.*

2. In a loss-of-coolant accident, the core spray systems are required to function effectively under circumstances in which some areas of fuel clad may have attained temperatures considerably higher than the maximum at which such sprays have been tested experimentally to date. The Committee understands that the applicant is conducting additional experiments, and urges that these be extended to temperatures as high as practicable. Use of stainless steel in these tests for simulation of the Zircaloy clad appears suitable, but some corroborating tests employing Zircaloy should be included.

The applicant stated that the control systems for emergency power will be designed and tested in accordance with standards for reactor protection systems. Also, he will explore further possibilities for improvement, particularly by diversification, of the instrumentation that initiates emergency core cooling, to provide additional assurance against delay of this vital function.

Steam line isolation valves are provided which constitute an important safeguard in the event of failure of a steam line external to the containment. One or more valves identical to these will be tested under simulated accident conditions prior to a request for an operating license.

Operation with a fuel assembly having an improper angular orientation could result in local thermal conditions that exceed by a substantial margin the design thermal operating limits. The applicant stated that he is continuing to investigate more positive means for precluding possible misorientation of fuel assemblies.

The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly by inlet coolant orifice blockage or by other means, to be remote. However, the resulting effects in terms of fission product release, local high pressure production, and possible initiation of failure in adjacent fuel elements are not well known. Information should be developed to show that such an incident will not lead to unacceptable conditions.*

A linear heat generation rate of 28 KW/ft is used by the applicant as a fuel element damage limit. Experimental verification of this criterion is incomplete, and the applicant plans to conduct additional tests. The Committee recommends that such tests include heat generation rates in excess of those calculated for the worst anticipated transient and fuel burnups comparable to the maximum expected in the reactor.*

The Rod Block Monitor system should be designed so that if bypassing is employed for purposes other than brief testing no single failure will impair the safety function.

The diesel-generator sets for emergency power appear to be fully loaded with little or no margin (on the design basis of one of three failing to start). They are required to start, synchronize, and carry load within less than thirty seconds. The applicant stated that tests will be conducted by the diesel manufacturer to demonstrate capability of meeting these requirements. Any previously untried features, such as the method of synchronization, will be included in the tests. The results should be evaluated carefully by the AEC Regulatory Staff. In addition, the installed emergency generating system should be tested thoroughly under simulated emergency conditions prior to a request for an operating license.

The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system and of inspection during service life. Because of the higher power level and advanced thermal conditions in the Browns Ferry Units, these matters assume even greater importance. The Committee recommends that the applicant implement those improvements in primary system quality which are practical with current technology.*

The Browns Ferry Units have been designed to provide the same accessibility for inspection of the primary system as for the Quad-Cities plants. A detailed inspection program has not yet been formulated by the applicant. The Committee will wish to review the detailed in-service inspection program at the time of request for an operating license.

Considerable information should be available from operation of previously reviewed large boiling water reactors prior to operation of the Browns Ferry reactors. However, because the Browns Ferry Units are to operate at substantially higher power level and power density than those on which such experience will be obtained, an especially extensive and careful start-up program will be required. If the start-up program or the additional information on fuel behavior referred to earlier should fail to confirm adequately the designer's expectations, system modifications or restrictions on operation may be appropriate.

The Advisory Committee on Reactor Safeguards believe that the items mentioned above can be resolved during construction of the reactors. On the basis of the foregoing comments, and in view of the favorable characteristics of the proposed site, the Committee believes that the proposed reactors can be constructed at the Browns Ferry site with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

The following are additional remarks by Dr. Stephen H. Hanauer. "It is my belief that the substantial increase in power and power density of the Browns Ferry reactors over boiling water reactors previously approved should be accompanied by increased safeguard system margins for the unexpected. The emergency core cooling system proposed should in my opinion be redesigned to provide additional time margin and to reduce the severe requirements for starting of large equipment in a few seconds. The dependence on immediate availability of a large amount of emergency electrical power, using diesel generators operating fully loaded in a previously untried starting mode, is of special concern, as are the high temperatures and numerous fuel-element failures predicted even for successful operation of the emergency core cooling system in a large loss-of-coolant accident."

Sincerely yours,

/s/ N. J. Palladino

N. J. Palladino
Chairman

* The Committee believes that these matters are of significance for all large water-cooled power reactors, and warrant careful attention.

References Attached

$2000^{o}F$ were calculated for a proposed power level of 2758 MWT.

Unfortunately, the physical modeling of ECCS performance for both PWRs and BWRs in the mid and late 1960's was deficient, and even wrong, in many significant areas. Fortunately, there appears to have been enough margin in other areas to roughly compensate for the deficiencies.

In early 1971, some tests in the Semi-scale Facility at the Idaho National Reactor Testing Station showed that the modeling of water flow from the accumulators in a PWR ECCS had been deficient, and that much of this water would leave the reactor vessel during blowdown following a cold leg break, rather than reflood the core immediately.

During roughly the same time period it also became clear that previous analysis of core reflooding in PWRs had failed to allow properly for the backpressure in the core built up by the resistance to the escape of steam through the intact loops on the way to the broken pipe. This latter phenomenon had a very large effect of the rate of core reflooding, which now dropped to the vicinity of 1 inch per second (or less). This resulted in a much longer time during which the fuel elements were uncovered (and poorly cooled) and during which fission product decay heat raised the temperature of both fuel and clad more than had been previously allowed for during this regime.

Similarly, a very different concept of how the core spray system in a BWR actually behaves, compared to the traditionally proposed picture, evolved; and there were other significant deficiencies uncovered in the physical modeling of LOCA-ECCS for both the BWR and PWR.

In late 1970 an ACRS Subcommittee initiated a review of LOCA-ECCS for BWRs and in March, 1971, Mr. Price, the Director of Regulation, asked a Task Force under Dr. Hanauer to review the status of LOCA-ECCS for BWRs and PWRs.

The ACRS meeting summaries from February, 1971 thru June 1971, provide some background on the development of the Interim Acceptance Criteria by the Regulatory Staff.

<u>Meeting with the Chairman of the Atomic Safety and Licensing Board</u>

<u>Trend of Interveners" Questions at Public Hearings</u> - Dr. Buck reviewed the trend of interveners' questions at public hearings. Prior to 1967, intervention was based primarily on matters related to anti-trust issues. Some questions on iodine releases were also raised. During the 1967-1969 era, questions related to radiological monitoring and QA began to appear. In 1970, questions related to safety aspects of the reactor started to appear even though the major interventions were by environmentalists.

The recent (1970-1971) Monticello hearing (OL) was the first of such major cases to challenge "as low as practicable" limits, the QA program and the redundancy of the ECCS. Division of Compliance reports were also requested.

AS&LB hearings of the Palisades plant (OL) are still being conducted. The intervenor has produced an in-depth cross examination and has raised questions regarding QA, Compliance reports and daily work logs at the plant. The objective is to obtain a zero radiological release plant and to eliminate any thermal increase into Lake Michigan. He does not really expect to stop plant operation in the final analysis but has a sincere concern regarding environmental effects.

At Shoreham (CP hearing) the intervenor apparently is making an all-out effort to prevent the plant from being constructed or operated. A major attack has been made on the adequacy of the ECCS. The intervenor has referred to the ACRS LWR reactor safety research letter of 1969, WASH 740, LOFT scheduling, etc. to supplement in his argument. An effort is being made to find a credibility gap. Other questions have been raised regarding the containment, industrial sabotage, pipe whip, safe ends, metal-water reaction, hydrogen generation and other major concerns raised during the last five years. The intervenor has stated that he does not expect to win with the ACRS, AEC, AS&LB stacked against him.

Dr. Buck noted that the Dresden 3 and Midland interventions are taking on the same approach as has occurred at Shoreham.

The AS&LB is now having to ask for RDT witnesses to address R&D efforts in areas of concern.

Mr. Wells stated that he does not believe an early site review would be helpful unless a specific reactor was identified for use and the technical concerns were understood. Otherwise, another hearing would have to be held to address technical questions.

Mr. Wells did not know if such things as benefits versus risks could be admissible evidence.

Dr. Buck felt the best way to go was to recognize where the intervenors are heading and be prepared to answer their concerns. The AEC and industry have to be candid enough to place in the record any risks that exist.

Dr. Bush noted that the Committee was considering having a "classified" discussion on industrial sabotage and invited Mr. Wells and Dr. Buck to attend if a meeting is arranged. (They expressed interest in attending.)

<u>Turkey Point Nuclear Generation Plant Units 3 & 4</u> - The Committee reviewed the proposed repairs of the Unit 3 containment dome. The Committee informed the applicant orally that the proposed repairs appear to be a reasonable basis for initialing dome repairs and that the remaining unresolved matters for an O.L. will be considered at a Subcommittee and full Committee meeting in the near future (probably theApril ACRS meeting).

The applicant was asked by Dr. Okrent to consider the results of recent INC experiments on ECCS.

ECCS - The Staff reported that they have arrived at an agreement with the applicant regarding adequacy of the ECCS. The core peaking factors will be limited to values which would limit the maximum estimated peak clad temperature to 2100^{O}F. This provides a margin until the remaining uncertainties are cleared up to the Staff's satisfaction.

Mr. Zane, Inc. reviewed results obtained from experiments using the "semiscale PWR" at INC to investigate the effects of flask injection water bypassing the core during blowdown. A significant fraction of the water bypassed the core and was blown out of the break.

Regulatory "Task Force"- Mr. Price noted that the task force, established to review generic matters, has asked INC to provide assistance in the review of issues related to ECCS. The task force will develop a position paper and provide the paper to the ACRS for its comments at the next ACRS meeting.

ECCS Mr. Price stated that he has informed the AEC that he has asked the Staff not to say anything in public on PWR ECCS system until the Staff can agree where it stands on the issues raised. He hoped the ACRS would not write a report on Turkey Point this month (a PWR) unless the Committee can state the reasons why it believes ECCS is satisfactory for PWRs. It would be no help to state that the matter should be resolved by the Staff. He wanted another 30 days for the Staff to work on this matter. He did not think it was necessary to take any action at present on operating PWRs. He will convey the problems to FPL (Turkey Point) before the ACRS review if this is found agreeable to the Committee.

(In executive session, the Committee decided to ask Mr. Price to convey to FPL the message that the Committee would listen to the FPL position but that the committee could not write a report at this meeting because the Staff had not completed its review of ECCS.)

Meeting with Working Group on ECCS

Dr. Hanauer reviewed the status of the Working Group on ECCS. They have found no major questions for BWRs which need answers at the present.

Obtaining some suitable heat transfer coefficients is the tenure of concern for BWRs.

He identified three areas of concern for PWRs in the order of concern:

(1) Fate of accumulator water during blowdown.
(2) Cooling by core flow during blowdown.
(3) Steam binding after blowdown.

Steambinding calculations should satisfy this. This time to cool the core may be longer than before.

Core Flow Cooling: More tests will be necessary to answer this concern.

Accumulator Water Fate: This is intractable. The concern is whether enough water will flow down the downcomer to provide cooling. There is some probability that a large portion of the water will flow around the ring header and out the break. The net result is that some hardware may be needed to solve this problem.

The basic concern of the Regulatory Staff is that they are now not able to testify with reasonable assurance that the ECCS is adequate for Palisades, Midland, or Turkey Point. The Staff is working with INC to try and establish an interim position, e.g., derate the plants, install hardware. Mr. Price indicated that the Staff will reach a position hopefully within a month.

Meeting with the Regulatory Staff

Adequacy of ECCS for PWRs - Dr. Hanauer reviewed the status of ECCS for PWRs. Westinghouse has calculated peak clad temperatures using a number of "best estimate" parameters (e.g., C_D of 0.8 versus 1.0). This calculation indicates that for Turkey Point (2200 MWT) a peak clad temperature of 1917°F would be reached. If all accumulator water were dumped on the floor (during the blowdown phase), the peak clad temperature would be 2333°F. The Staff is evaluating the validity of these best estimate values.

Dr. Hanauer indicated that B&W has a number of sensitivity studies to complete. CE is apparently not planning to conduct any further studies. He indicated that Westinghouse has requested that the Committee hold a special meeting on Turkey Point as soon as this matter is resolved within the AEC. He believes a position can be reached for Westinghouse plants in about two weeks.

Emergency Core Cooling System

1. REG Task Force Recommendations - The Committee was informed that the Commissioners had met with each of the reactor vendors to discuss the REG Task Force's recommendations, which had been provided to the Committee in draft. The Director of Regulation indicated that REG was considering issuing the final evaluation techniques as an appendix to 10 CFR Part 50 in order to "stabilize the licensing process."

   The current REG position included acceptable evaluation models, and prescriptions for their use, for GE and Westinghouse and an AEC model based on INC-developed codes (RELAP 3 and THETA 1B) which could be used generally. No satisfactory models have been developed by Combustion Engineering or Babcock and Wilcox. The Task Force estimated that B&W could have an acceptable evaluation model within

2 or 3 months, but CE had only begun work which had even the potential for a satisfactory conclusion. For perhaps the next year, CE designs would have to be evaluated using the AEC model with the result that Palisades, for example, might not be licensable at more than half-power during that time.

The Task Force also recommended that plants currently in operation without "modern" ECCS and unable to meet the Task Force Criteria (Dresden 1, Big Rock Point, Indian Point 1, San Onofre, Yankee Rowe, etc.), be required to accelerate their schedules for the design and installation of adequate ECCS systems, and to perform increased in-service inspection, etc., in the interim.

Dr. M. Rosen presented a "minority report" in which he contended that the codes used in the evaluation models were developmental enough, and the recommended prescriptions for their use were arbitrary enough, that the evaluation models should not be used as the sole basis for licensing plants. Instead, he suggested:

1. A limit on peak linear heat generator rate to be applied to all plants.

2. A moratorium (short-term) on further increases in peak heat generation rate.

3. Cooperative AEC-Industry funding and utilization of resources leading to long-term solutions.

2. Meeting with the Commissioners - Commissioners Seaborg, Ramey and Johnson attended. After a brief summary of the current status of the commission review, Dr. Bush expressed the Committee position as follows:

"The Committee would like to emphasize that the following opinions are an initial reaction to the proposal, as presented orally by the Staff, because of the limited time we have had the draft paper, the limited discussion with the Staff, and the lack of substantiating documents. With these provisos, the following represents suggested positions concerning the specific regulatory problem with ECCS systems as well as longer term suggestions with regard to emergency core cooling:

(1) The Committee does not feel the document in its present form should be released as a part of 10 CFR 50 regulations.

(2) The Committee suggests that a preferred mechanism would be a two stage approach.

(a) The current paper, with appropriate modifications such as those suggested below, could be used as an interim staff position. This could be publicized as appropriate.

(b) A desirable second step would be to release the position as a safety guide after sufficient additional information and other appropriate modifications and clarifications are made. This second step might require several months, as was the case with problems of comparable magnitude such as hydrogen generation and anticipated transients without scram.

(3) Alternates to the suggestion of the Staff regarding reductions in power should be explored (for example, with respect to Palisades consider peaking factors, probability considerations, etc.). The Committee feels that increased levels of inspection and fixed deadlines to complete necessary system changes represent preferred routes.

(4) The Committee reiterates its belief that the ECCS problem is a very real one requiring substantially more analytic and experimental work by the AEC and industry. The ultimate solution may require revisions in hardware or in cooling concepts and too little attention appears to be directed in this direction, compared to analytic studies.

(5) Some reservations concerning the suggested AEC approach particularly with regard to publication as a 10 CFR Part 50 rule include:

(a) The acceptance of the AEC Codes Theta 1B and RELAP-3 as a benchmark is questioned on the basis of the limited number of test runs made.

(b) The approach suggested in the document appears to be conservative; however, past experience leads us to believe that this may not be the case over the spectrum of breaks. Alternatively, the approach may be excessively conservative.

(c) The approach is a "cookbook" one and does not really attack the basic problems. Regulation may require this but its a problem to the Committee.

(d) Plants having no or inadequate ECC systems should be given a fixed period to complete acceptable designs and a definite time schedule for installation.

(e) The document is not clear as to the times of implementation of the alternate proposals.

(f) Some criteria are mixed in the statement of objectives.

Mr. Price strongly supported the publication of the AEC position on ECCS as a 10 CFR Part 50 rule although the Commissioners did not seem as committed to this course of action.

Mr. Price maintained that it was necessary that a position on ECCS have Commission endorsement in order to stabilize the licensing process. He maintained that a Staff position on this matter would have little effect on AS&LBs who would decide matters on an ad-hoc basis without a policy or guidance from the Commission.

Various committee members expressed the following thoughts.

(1) The adequacy of the INC codes (RELAP and THETA-1B), used in the AEC Evaluation model has not been established.

(2) The degree of conservatism in the proposed prescriptions for evaluation models is not clear.

(3) It is disturbing that this ECCS review was not handled in a manner similar to the way in which ATWS and hydrogen generation were handled (e.g., without stopping the licensing process and the publicity which has accompanied this review).

(4) The derating of CE plants which may result is of concern. Plants that are already built (but not yet licensed) might be considered "grandfathers" as well as plants that are already operating.

(5) The instability in what is considered an adequate code is of concern (e.g., in 6 weeks the Staff has changed its position regarding the adequacy of CE low pressure injection flasks with respect to the large pipe break).

(6) The high degree of sensitivity of the analysis to small changes in assumptions and parameters is of concern.

(7) The use of a long list of conservative assumptions in the model may result in a conclusion that is unnecessarily restrictive.

Commissioner Johnson noted that the Commission must have a position on ECCS before the JCAE hearings begin on June 22, 1971. Dr. Hanauer noted that it is necessary for the Staff and the ACRS to agree on the mechanical aspects of the proposed position otherwise there will be disagreement regarding the adequacy of proposed plants.

It was noted that the ACRS is not necessarily in disagreement with the technical position proposed but believes that it has not received the review and consideration appropriate for a Commission rule.

The Commissioners noted that they would very much like to develop a position by June 19, 1971, which the ACRS can support. The Committee agreed to continue its consideration of this matter.

3. Meeting with the Regulatory Staff - Dr. Isbin gave a detailed report on his view of the Task Force recommendations. In particular he questioned the acceptance of the CE model without an independent evaluation since independent analyses of portions of the CE model have led to significant changes being required. He also questioned the AEC model as a basis for licensing plants (and that is all that is now available for use with B&W and CE designs) since the codes involved have not been used widely in the way now required. Consequently, there is no way to predict what the outcome would be when this model was applied. It is not even possible to predict that an evaluation of a Westinghouse design, which would be found acceptable under the approved Westinghouse code, would also be acceptable using the AEC model.

   After a detailed technical discussion of current Regulatory Staff approaches to BWRs and PWRs and a caucus, the Committee informed the Regulatory Staff that many members were particularly troubled by the suggestion that the draft Task Force proposal be published as a Commission regulation.

   Dr. Hanauer noted that a statement of Commission policy is needed but not necessarily a rule. The aim is to have each hearing board use an "identical yardstick" to measure emergency core cooling systems.

   With respect to currently operating plants, several members noted the inconsistency in allowing continued operation of plants which are currently licensed, but which have no adequate core cooling systems, while simultaneously prohibiting or severely limiting the operation of a plant such as Palisades with a much more complete core cooling system.

   In closing, the Committee asked the Regulatory Staff to provide the following for its special meeting on June 16-17, 1971:

   (1) As much information as possible on the plugging phenomenon, particularly as it applies to Combustion Engineering reactors.

   (2) A written support paper with all technical material available, over and above what has already been provided by the Committee.

   (3) What the application of the draft task force recommendation will mean in terms of Combustion Engineering and Babcock and Wilcox plants which will be reviewed shortly as well as ice-condenser systems, to the extent that this is known.

(4) The degree of margin in power density between a calculated 2300°F peak clad temperature and clad melting following a LOCA for designs other than Westinghouse. (It had been noted previously that an 8% increase above the 15.8 kw/ft limit established for Turkey Point under the Task Force proposal would lead to calculated clad melting following a double-ended, cold leg break using the calculational methods of that proposal).

(5) Whatever RELAP 3/THETA 1B computer runs are available.

(6) The rationale behind the various choices which were made in establishing the prescribed assumptions for the Westinghouse calculations.

A discussion, to the extent possible, of the rationale under which systems which have no emergency core cooling are allowed to continue operation was also requested.

The detailed comments on the draft recommendation were also to be incorporated in a new draft in time for the Special ACRS meeting.

SPECIAL ACRS MEETING
JUNE 16-17, 1971
WASHINGTON, D.C.

Executive Session

Independent Evaluation of ECCS for Boiling Water Reactors

The Committee considered recommending to the Regulatory Staff than an independent evaluation be provided of the efficacy of ECCS for all classes of BWRs.

Comments by members included:

1. Additional consideration is also needed of the current AEC evaluation model for PWRs.

2. The Safety Research Subcommittee should have meetings on this subject so that the ACRS can have more effective input into the development of the model.

3. Independent analyses of systems important to safety could go a great deal beyond ECCS.

4. While this might be a desirable action, the Committee should consider whether it would be using the (always) limited available resources in the most effective way.

The subject was tabled until the 135th ACRS meeting. Members were urged to provide opinions, points of view, etc., in writing if possible.

Emergency Core Cooling Systems Review

The Committee had been provided with a new draft policy statement, dated June 16, 1971, incorporating the suggestions and comments made during the 134th ACRS meeting, a listing of reactor plants indicating the effect of various effective dates for implementation of the interim policy, and was later given a further revised draft of the policy statement dated June 17, 1971.

ACRS Consultants' Discussions

During the course of the meeting, the Committee held several discussions with its consultants in this area. Some of the points made included:

Dr. Ybarrondo

1. The blowdown and heatup codes used by the reactor vendors have been improved substantially during the past year, but since a substantial amount of technical judgment is still involved, they should be used with caution.

2. All of the codes currently in use are in a state of flux. Detailed and accurate system representation has only been attempted during the past year or so, and not all of the vendors have done this yet.

3. It is necessary to assure that solutions to the various equations being used are being approximated closely enough to assure stability of the codes.

4. The resulting programs are extremely complex (12,000 cards are used in RELAP) and must be checked very carefully.

5. Standard problems should be worked out as checkpoints for all the codes.

6. Simplified equations are solved in the codes and this requires the use of many simplified or empirical correlations. For example:

    critical heat flux
    post-critical heat flux
    reflooding heat transfer
    spray cooling (GE only)
    two-phase pressure drop
    critical flow at break (and orifice coefficient)
    pump coastdown representation
    phase separation
    fuel-cladding gap coefficient
    metal-water reaction rate (usually Baker-Just)
    thermal properties

In using these and others, it is necessary to compare their range of applicability with the range over which they are to be used, the geometry, and the aptness of extrapolations.

Dr. Ybarrondo thought that the application of safety factors to the results was appropriate because of the current insufficiency of data.

Dr. Ybarrondo also noted that the recent series of semi-scale blowdown and injection tests had been intended only to compare the code-calculated decompression with measurements, check on appropriate orifice coefficients, the amount of water left in the vessel, the nature of the initial core flow reversal, etc..

In general, Dr. Ybarrondo did not feel that the assumptions currently being used in the PWR calculations were too conservative; in some specific cases, they might not be conservative enough.

Dr. Carbiener

Dr. Carbiener felt that the Draft AEC Policy Statement reflected his feelings well. The approach is adequate, and may even be overly conservative, although this could not be proven now.

Given a double-ended cold leg break, an emergency core cooling system which met the criteria laid out in the policy statement would only have a one-in-a-thousand chance of failing. (Dr. Ybarrondo agreed with this conclusion of BWRs, but was not so confident of the situation for PWRs.)

Meetings with the Regulatory Staff

The Committee met with REG representatives several times during the meeting to discuss the revised draft of 6/16/71, the Commissioners' comments on the draft, the effective date of the proposed policy statement and its effect on operating and soon-to-be operating plants, the final revised draft of 6/17/71, etc.. Detailed, page-by-page technical comments were offered and, in many cases, incorporated into the final draft. Some of the items which were discussed at length or about which Committee members seemed to feel most strongly were:

1. Form of Policy Statement - With respect to the Committee's previous comments, Mr. Price indicated that the Commissioners felt that a statement of Commission policy, in the Federal Register, was necessary. It would not be an appendix to Part 50, however.

2. Criteria - The section on "Criteria for Specific Reactors" was too inflexible in the 6/16/71 version of the draft. It was later rewritten to indicate that the specific evaluation models listed were "examples" of acceptable approaches, allowing appropriate changes, additions, etc., without Commission approval and a notice in the Federal Register.

3. Effective Date - The alternatives which had originally been presented were to apply the policy to plants for which operating licenses were issued on or before either January 1, 1968 or July 1, 1972. Later in the meeting, it was reported that the Commissioners did not feel they could support any approach beyond exempting, for some interim period, plants with operating licenses issued prior to January 1, 1968. (This would include only eight reactors of relatively low power level.)

   Several members felt that extending the exemption date to July, 1972 would only mean exempting a few plants from any real requirements (such as reducing power or peaking factors) since most plants scheduled to be licensed for operation by that time are expected to be acceptable even under the new policy. Also, it seemed inconsistent to permit operation of older plants with essentially no emergency core cooling, albeit with augmented inspection and surveillance programs, while restricting operation of plants (e.g., Palisades) with a much more complete emergency core cooling system.

4. Margin Between an Acceptable System and Calculated Clad Melting

   Several members noted that, as the REG prescription is now written, there is a margin between a calculated peak clad temperature of 2300^{o}F and clad melting of only 8% in the peak linear heat generation rate (in Turkey Point, for example). The REG representatives indicated that they had been driven to the conservative prescriptions in the calculations largely through an inability to define a "realistic" computation.

5. Fraction-of-Clad-Reacted Criterion - Objections were raised to the inclusion of a 1% metal-water reaction limit in the criteria for acceptability. The Regulatory Staff agreed that an average of 1% of the core reacted implies a degree of local reaction which is clearly unacceptable. The suggestion of extending the criteria to include a limit on the length of time during which cladding can remain at high temperatures was made.

   In the 6/17/71 draft the implication that an average limit of 1% precludes local clad embrittlement had been softened and the limited-time concept included, but the criteria remained at least partly objectionable to some members.

6. Improvement for Future Reactors - Some members, and at least one ACRS consultant, were concerned over the possibility that a detailed prescription leading to regulatory approval would tend to "freeze" the technology at its current level.

   Others felt that, while the policy statement did not provide the maximum incentive for improvement, it did not preclude progress. REG felt that the policy statement was not the appropriate place to urge additional improvements.

7. Lack of an Accepted Evaluation Model for BWR LPCIS - Objection was raised to the lack, in the accepted GE model, of an evaluation model for the LPCI system. The Regulatory Staff felt that if evaluations for two different systems were required for BWRs, they could not license BWRs with only a single type of ECC system.

8. Requirement to Evaluate Effect of Nitrogen Gas in BWR Accumulators - The requirement, under both the AEC and Westinghouse evaluation models, to evaluate the effects of the nitrogen in the accumulators was objected to on the basis that such a calculation had never been done, and it was not clear that one could be done with any precision within a reasonable time. The requirement was modified in the later draft to indicate that only the effect on steam binding during reflooding need be considered.

Executive Session

The Committee held several discussions to arrive at a statement of Committee position for presentation to the Commission later in the meeting. Items which were of importance were:

1. Public Statement of AEC Policy - The Committee agreed to approve handling the interim policy statement as described by Mr. Price, i. e., publication, as Commission policy, in the Federal Register, but not as part of an AEC Regulation.

2. Effective Date of AEC Policy - Several members felt that to adopt an approach which implied derating Palisades by a large amount ("Alternate A" of the 6/16/71 draft) was not defensible without more justification than had been given; others felt as strongly about any approach which would exempt upwards of 50 plants for three years.

   Other comments included:

   a. It is not compatible with derating Palisades by 50% to allow operation of essentially unprotected plants for three years.

   b. There is no reason why CE can not do more on ECCS analyses and avoid the need to derate Palisades.

   The Committee voted to approve the application of the interim AEC Policy on ECCS to all plants for which operating licenses are issued after January 1, 1968.

3. Evaluation Model for GE LPCI System - Some members felt that, since the LPCIS was the basis, in part, of favorable conclusions concerning BWRs (e.g., on Shoreham), the Committee should advise the AEC that an independant evaluation model is necessary for LPCIS. Others could not defend requiring two diverse systems for BWRs, but only one for PWRs. It was also noted that silence on a point, in the policy statement,

does not mean it is not included in reviews of individual projects.

The Committee agreed that the GE evaluation model in the interim policy statement (6/17/71 draft) was acceptable.

4. Inclusion of Evaluation for Small and Hot Leg Breaks - It seemed to be the consensus of Committee opinion that mention of smaller breaks and hot leg breaks was desirable, but the Committee would not insist on it. The later draft of the policy statement covered this point by reference to analysis of "a spectrum of breaks."

5. Use of a Conservative Rather than Best-Estimate Fit for Deprivation of FLECHT Heat Transfer Coefficient - Some members were troubled because they had originally been led to believe that every choice in the evaluation models was the most conservative one possible, but had since learned some of the prescriptions were less conservative than was possible.

The Committee adopted the following statement of position on ECCS, for presentation to the Commission:

ACRS STATEMENT REGARDING ECCS

The ACRS believes that the draft, "Interim Acceptance Criteria for ECCS for Light-Water Power Reactors," dated June 17, 1971, is satisfactory for release subject to some clarification of the language.

The Committee concurs with the suggested approach of publishing the criteria in the Federal Register as an interim policy statement by the Commission. It agrees that Alternate A setting January 1, 1968, as a cut-off date is reasonable. It agrees with the relaxation in the criteria in Section IV B as promoting greater flexibility without freezing design and codes.

The Committee believes that the document represents an interim solution only, and that more work is required in code development, safety research oriented to LOCA-ECCS, and work on improved ECCS. The Committee is prepared to work closely with Staff and vendors through appropriate Subcommittees. We believe this approach may lead to an orderly solution of the outstanding problems.

With regard to the course of events of the last few months, the Committee now feels that there has been an over-all gain in that the level of work on ECCS evaluation has increased and more specific goals and objectives have been established. However, the Committee believes that more experimental work is needed to supplement the further analytical development of the evaluation models.

The Committee reemphasizes that the Commission should indicate publicly the need for continuing work on new ECC systems or significant modifications to current designs. It should be indicated that increases in power or power density will require larger margins.

The Committee believes its oral concurrence in the policy statement is the preferred mechanism of communication rather than a letter. We believe our approval of pending cases is the best mechanism for publicly indicating our approval of current designs of ECCS.

Some members wish to comment personally on this matter.

Meeting with Commissioners

Commissioners Seaborg and Larson attended.

Dr. Bush read the statement of ACRS position on ECCS.

Dr. Stratton added a personal comment to the effect that he had seen no evidence which supported a requirement to derate any plant, particularly Palisades. Mr. Price commented that he, too, did not want to require derating of any plant, but that it was difficult to defend the technical basis which CE and Consumers have so far provided in support of their request for a full-power operating license.

Dr. Okrent noted a potential omission in the interim policy in that it ignores any requirement for evaluating the adequacy for BWR LPCI systems. He also noted that he, personally, was less certain of the conservatism of the accepted evaluation models than the Regulatory Staff seemed to be. His own conclusion that plants could be permitted to operate was based not only on a high probability that ECC systems would cool the core after a large pipe break, but on the very low probability of such a break occurring. He suggested that the latter approach should form at least part of the basis for allowing continued operation of plants with little or no ECCS.

Chairman Seaborn expressed interest in the Committee's feeling that there had been an over-all gain through the events of the last few months. The added effort on ECCS evaluation was mentioned in response.

During the 134th meeting, the ACRS heard from Dr. Hanauer concerning the tentative Staff position on Interim Acceptance Criteria. It also heard a dissent from Regulatory Staff member Dr. Morris Rosen. In Dr. Rosen's opinion, computer codes should not form the sole criterion for licensing nuclear power stations, particularly with the kinds of assumptions in use. He discussed the sensitivity of the calculated results to more or less arbitrary assumptions inherent in the Task Force recommendations. He suggested for the short term on arbitrary limit on peak linear heat generation rate of 15-16 KW/ft.

The ACRS did not endorse either position. Rather the Committee gave only limited approval to the Interim Acceptance Criteria, and emphasized the need to work both on improved evaluation models and on improved emergency core cooling systems.

At the 139th meeting, November 11-13, 1971, the ACRS discussed at some length the matter of how to develop and implement improvements on ECCS, as the excerpts below indicate.

1.1.2.8 <u>Leadership in ECCS Research and Implementation of Improved ECCS Designs</u>

Shortcomings of the AEC interim evaluation model for ECCS, urgency of need for improvement of both the criteria and systems based thereon and alternate means for achieving desired ACRS objectives were discussed at length. Dr. Isbin expressed the opinion that priority attention should be given to improvement of the interim evaluation model and that two to six months would be a reasonable time for completing this.

Dr. Okrent proposed a possible approach regarding improved ECCS (see attachment 2) and the Committee approved a motion by Dr. Okrent that the Chairman initiate early action to accomplish the desired objective by assigning appropriate tasks to suitable existing or new ACRS groups or subcommittees.

(Note: Dr. Bush has appointed an ECCS Subcommittee to handle this matter).

1.1.2.11 <u>Publication of Paper on the "Calculated" Loss-of-Coolant Accident - A Review, L. Ybarrondo, C. W. Solbrig, and H. S. Isbin</u>

Dr. Isbin reported on recent developments regarding AEC release of the subject paper for publication.

DRD&T has turned over responsibility for release of this paper to Aerojet Nuclear with stipulations which, in Dr. Isbin's opinion, made it difficult for Aerojet management to actually approve release.

2.0 MEETING WITH THE REGULATORY STAFF

2.1 PROGRAM FOR DEVELOPMENT OF IMPROVED ECCS DESIGN

Mr. McEwen (DRD & T) reported on AEC accomplishments on ECCS projects since adoption of the AEC interim ECCS criteria. The following items were noted.

- New FLECHT tests
- GE Blow Down Heat Transfer (BDHT) Program
- 1-1/2 Loop Semi-Scale modification and issuance of a Preliminary System Design Description for the facility.
- RELAP AND THETA-1B improvements
- Multiple injection capability included in LOFT and Semi-Scale
- Preliminary PWR BDHT program initiated at ORNL
- Design modification of the PBF initiated.

RDT bases and plans for future programs and the scope of contemplated ECCS studies were also described.

Mr. McEwen noted that RDT has initiated only those programs which are considered adequately defined; that funding is a problem; that water reactor safety should be the concern of those who are building the plants; that RDT has entered into a joint program with industry, which is funded by AEC at a level of 25 percent; that none of the programs are directed toward development or testing of advanced ECCS concepts or investigation of metal/water reactions; and that the program includes attention to BWRs through code modifications.

Although the approach, scope and progress of the RDT program was viewed with disappointment by many members of both the Committee and Regulatory Staff, there was no unanimity of opinion. Major points of disagreement revolved around the issues of ECCS shortcomings and organizational responsibility for implementing programs for improvement of either the evaluation criteria or the systems. Viewpoints were too divergent to permit either resolution of the matter or adoption of a solution-seeking course of action agreeable to both ACRS and the Regulatory Staff at this meeting. (An ACRS Subcommittee has been assigned to consider this matter).

2.4.3 Plans for Implementation of Improved ECCS Designs in Accordance with AEC Interim Criteria

Dr. Hanauer referenced the DRD&T report by Mr. McEwen, which reflected lack of AEC progress in implementing programs and achieving timely results in this area, and he expressed the opinion that although it is possible that things may not be as bad as they seem, indications are that performance of DRD&T in this area could be improved. During discussion of alternate approaches to ensure maintenance of a viable water reactor safety program, Mr. Squires informed the Director of Regulation that the Committee had previously recommended that the Regulatory Staff should have its own budget for independent work in this important area. In response, Mr. Muntzing stated that he felt the recommendation had considerable merit, and that it would be particularly desirable for the Regulatory Staff to have such funds, particularly if separation of regulatory and promotion activities of the AEC should become a reality.

The following specific points were noted:

- Neither the AEC nor the NSSS vendors understand the problems of ECCS well enough to know what is an improvement in system design.
- Problems associated with the interim evaluation model must be cleaned up.
- The adequacy of existing ECC systems must be confirmed.
- It is not entirely reasonable to require NSSS vendors to comply with the interim criteria model and at the same time require them to improve the model and their systems.

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At the 140th meeting, December 9-11, 1971, the ACRS heard extensive presentations on proposed amendments to the Interim Acceptance Criteria and prepared a report on this matter for use by the Commission in the upcoming rulemaking hearing on Acceptance Criteria for ECCS. This version of the letter was not sent in order to take into account comments by the Commissioners regarding: (1) the desirability of including a more positive statement with regard to the safeguards adequacy of reactors designed in accordance with the Interim Criteria, and (2) the possibility of greater exposure of minority commenters to subpoena during rulemaking hearings.

1.2 SUBCOMMITTEE AND OTHER REPORTS

1.2.1 Report of Subcommittees on ECCS Analysis for CE and B&W Reactors

The Committee heard presentations by the respective Subcommittee Chairmen in preparation for joint meetings to be held with the Staff and each vendor on Friday, December 10, 1971. The purpose of the meetings was to evaluate vendor-proposed ECCS analytical methods and models against the requirements of the AEC Interim Acceptance Criteria for ECCS. Dr. Bush indicated that the Committee should be prepared to inform the Staff of its conclusions regarding the acceptability of the vendor proposals, but that a letter might not be required.

1.2.1.1 CE Subcommittee Report

Dr. O'Kelly reviewed the information developed by the Subcommittee concerning CE progress in achieving a satisfactory method and model for analyzing ECCS performance for their nuclear plants. In discussion, the Commission was informed that its consultant was not much impressed by CE's 1/5-scale experimental work in the sense that he did not think the model accurately reflected ECCS phenomena to be expected in the full-scale reactor plant. It was pointed out, however, that the work to date was directed toward achieving eventual solution of the problems, and that the possibility for existence of imperfections was recognized. It appeared that most of CE's effort since last June had been devoted to improving the model.

1.2.1.2 B&W Subcommittee Report

Dr. Hendrie reviewed the status of B&W's work in this area and observed that their proposed evaluation code applies only to the vent-valve plant. For these plants the accumulators inject directly into the reactor vessel annulus.

There was considerable inconclusive discussion among Committee members regarding 1) variations in PWR plant design among Westinghouse, B&W, and CE, and reasons for same, 2) shortcomings of the AEC Interim acceptance Criteria for ECCS, 3) differences in blowdown/injection phenomena among the plants of the various PWR NSSS vendors, and 4) B&W interest in continuing participation in the joint AEC/industry ECCS R&D program.

1.2.1.3 Meeting with Regulatory Staff

During the subsequent meeting with the Regulatory Staff, the following information was provided in the form of individual opinions regarding ECCS problems and uncertainties associated with the current evaluation criteria and the models, techniques and designs used by NSSS vendors.

- The proposed rule change will grant equal acceptability status to GE, B&W, CE, and W models.

- A standard is needed for determining the acceptability of ECCS evaluation codes,

- There can be more than one type of ECCS fix, but a design fix is required for every plant. ECCS fix problems are generic in nature.

- The Regulatory Branch is guided by a desire to avoid establishing obstacles to the licensing of plants.

- The reduction of data from the FLECHT program made no allowance for radiant heat transfer to the walls. Corrections are being made.

- The principal ECCS problem for PWRs is steam binding.

- Present reflooding rates are considered marginal.

- Better definition of PWR blowdown heat transfer characteristics is needed.

- A crash program to achieve the optimum resolution of all ECCS problems is not warranted in view of the extremely low probability of the LOCA.

- Some consideration should be given to the need for avoiding excessive margins of conservatism.

1.2.1.4 Meeting with CE

Spokesmen for CE reviewed their analytical and experimental work on ECCS in considerable detail. The presentations included information on their past and future experimental programs and on the essential features and uses of:

- The CE FLASH 4, STRIKIN, RELBOT, and PERC codes and hand calculations

- the 1/5-scale model tests

- brittle fracture tests of Zircaloy cladding

- tube-bundle flow and burst tests.

On the basis of CE studies and experience, Mr. West expressed the opinion that the AEC Interim Criteria peak clad temperature limit of 2300°F was too conservative; that emergency coolant injection would not be blocked by steam; that the present trend of the regulatory process was in the direction of licensing methods rather than plants; and that the objective of the presentations would be to show that injection blockage would not occur in CE systems.

The following major points of information were provided by vendor representatives in discussions with Committee members:

- CE injection tests showed no flow blockage indications and no pressure fluctuations.
- Test results will be exploited to the maximum practical extent for ECCS improvement.
- Important vendor incentives would be the potential for safe operation at higher core power densities or confirmation of greater safety margins in current designs.
- CE bundle-blockage tests resulted in only about twice the original (pre-blockage) Δ P.
- Bundle distortion is expected to have only a minor effect on peak clad temperature.
- External surfaces of cladding can be exposed to steam of 2300°F for 130 seconds without embrittlement exceeding the CE standard.

1.2.1.5 Meeting with B&W

B&W representatives presented a summary of their activities related to emergency core cooling, including their accomplishments and future plans related to code and model improvement.

Mr. Montgomery stated that B&W's evaluations are based on the equilibrium model of the CRAFT code, which was capable of following the entire course of the accident in one calculational sequency. Their analyses of ECCS show peak clad temperatures of less than 2300°F and less than 1% metal/water reaction. Although they feel that an improved CRAFT will become a major analytical tool of the future, their work on the non-equilibrium version of CRAFT has been suspended, because of B&W's feeling that the time required to obtain Regulatory approval would be too long to avoid plant licensing problems. In order to expedite code acceptance by AEC, they are working on the development of a code similar to RELAP by adaptation and modification of published codes.

Other vendor spokesmen provided the following information in discussion with Committee members.

- For B&W plants, assumption of a locked primary pump rotor during blowdown is not conservative, but the assumption is conservative during reflood.

- The B&W method of analysis correlates satisfactorily with results of the early semi-scale tests (through test 859).

(Note: The Committee position with respect to acceptability of the ECCS evaluation models, codes, program and plans proposed by these vendors was formalized in its letter report* to the Commissioners on the proposed public rule-making hearing regarding 1) the AEC Interim Acceptance Criteria for ECCS, and 2) the addition of the CE and B&W evaluation models to the present list of acceptable models.

1.2.2 Report of Subcommittee on ECCS - ACRS Position Regarding AED Interim Criteria on ECCS

Dr. Bush referred to the November 26, 1971 AEC public notice of an ECCS rule-making hearing to be held on January 27, 1972, and reviewed the chronology of past events and the projected schedule of future events with regard to this announcement. He pointed out areas of possible Committee concern and emphasized the need for prompt action by the Committee to establish a position on the total matter and to provide guidance with respect to implementing whatever actions might be required by that position. He stated that he had discussed the matter with the Director of Regulation and had advised him of a possible Committee dissent. There was considerable discussion among Committee members regarding:

- the poor timing of related past events and lack of sufficient advance notice.

- extent of AEC understanding and appreciation of the Committee position of last June on the AEC Interim Acceptance Criteria for ECCS.

- whether ACRS opinion or comment was desired and, if so, the proper form of comment.

- ways and means for conveying Committee comment if comments should be deemed appropriate.

- possible misinterpretation of phraseology of the announcement in regard to ACRS participation in the review of the interim Criteria.

The general consensus was that comments/recommendations should be made, and Dr. Okrent expressed the opinion that Committee advice on

*Letter, Spencer H. Bush to Dr. Schlesinger, REPORT ON INTERIM ACCEPTANCE CRITERIA FOR ECCS FOR LIGHT-WATER POWER REACTORS, dated 1/70/72.

significant matters should be in writing to avoid misunderstanding. During a meeting with the Regulatory Staff on the following day (December 10), Dr. Hanauer indicated that a letter would avoid any possibility of a misunderstanding of the Committee position.

Subsequent deliberations of the Committee resulted in a fourth and final draft of a letter* to the Commission which would correctly reflect the position and recommendation of the ACRS with regard to the ECCS criteria and the proposed adoption of evaluation models. A motion for acceptance of the final draft was unopposed.

With regard to the manner of transmittal, the Committee approved a motion by Dean Palladino which would require hand delivery of the letter to the Commissioners prior to December 20, with opportunity for discussion. If this could not be accomplished the letter should be sent. The motion was not opposed, but one member abstained. Dr. Bush noted that if the Commissioners identified serious problems with the ACRS report he would refer it back to the Committee for further consideration.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 7, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON INTERIM ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER POWER REACTORS

Dear Dr. Schlesinger:

At its 140th meeting, December 9-11, 1971, the Advisory Committee on Reactor Safeguards completed a review of proposed amendments to the Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors, published as an AEC Interim Policy Statement on June 29, 1971. The proposal was considered at a Subcommittee meeting held on December 4, 1971, in Washington, D. C. Further consideration was given to the interim criteria at the Committee's 141st meeting, January 6-8, 1972. The proposed amendments add new evaluation models by Combustion Engineering, Inc. and the Babcock and Wilcox Company to the existing list of acceptable evaluation models.

The Committee concluded that the proposed amendments were acceptable, on the same basis as the original Policy Statement evaluation models. Both the original and the amended criteria involve a number of provisions which are clearly conservative as well as some which may not be conservative, but on balance reflect adequate conservatism for interim use with plants similar in design to those which have been reviewed for construction permits. The Committee believes that each plant should be reviewed on a case by case basis to determine the extent to which these criteria are satisfied or modifications to the design or operation of the plant are required.

At the time adoption of the Interim Acceptance Criteria was under consideration, the Committee had opportunity to discuss the draft Criteria with the AEC Regulatory Staff and the Commission, and to offer comments. The Committee comments, made orally to the Commission on June 17, 1971 included the following:

"The ACRS believes that the draft, 'Interim Acceptance Criteria for ECCS for Light-Water Power Reactors', dated June 17, 1971 is satisfactory for release subject to some clarification of the language.

"The Committee concurs with the suggested approach of publishing the criteria in the Federal Register as an interim policy statement by the Commission. It agrees that Alternate A setting January 1, 1968 as a cut-off date is reasonable. It agrees with the relaxation in the criteria in Section IV B as promoting greater flexibility without freezing design and codes.

"The Committee believes that the document represents an interim solution only, and that more work is required in code development, safety research oriented to LOCA-ECCS, and work on improved ECCS. The Committee is prepared to work closely with Staff and vendors through appropriate Subcommittees. We believe this approach may lead to an orderly solution of the outstanding problems.

"With regard to the course of events of the last few months, the Committee now feels that there has been an over-all gain in that the level of work on ECCS evaluation has increased and more specific goals and objectives have been established. However, the Committee believes that more experimental work is needed to supplement the further analytical development of the evaluation models.

"The Committee reemphasizes that the Commission should indicate publicly the need for continuing work on new ECC systems or significant modifications to current designs. It should be indicated that increases in power or power density will require larger margins."

Some of the restrictive assumptions imposed by the Interim Criteria have been introduced because of uncertainties in the behavior of the ECCS. The Committee believes such restrictions should be removed, as_possible, either by design changes, or by demonstration that they are unnecessary. The Committee believes that it is timely for further implementation of the statement in the Commission's Policy Statement of June 29, 1971, that "in connection with the water power reactors yet to be designed and constructed the possibility of accomplishing by changes in design further improvements in the capability of emergency core cooling systems should be considered." Although the Interim Acceptance Criteria should be useful and helpful in the licensing process, the evaluation models prescribed in these criteria are recognized to have only limited usefulness as design tools for improving emergency core cooling systems. The nuclear industry should respond in a more direct fashion with realistic design methods, based upon additional scaled experiments and analytical studies. Design changes which would clearly eliminate any potential steam binding problem during a loss-of-coolant accident represent an example of the type of improvement considered to be of particular importance. The Committee recommends that design changes to improve ECCS capability should be sought and, to the extent practical, employed in plants for which construction permit applications are received in the future, irrespective of whether the plant design without such changes appears to meet the provisions of the Interim Acceptance Criteria.

The Committee believes that there is reasonable assurance that, with appropriate use of the Interim Acceptance Criteria and other applicable design and evaluation criteria, water reactors of current design can be operated without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

The ACRS reconsidered its letter at its 141st meeting, January 6-8, 1972, and arrived at a new consensus which incorporated the bulk of the minority opinions which had been appended to the December letter. The letter of January 7, 1972, and excerpts from the meeting minutes follow:

SUMMARY
141ST ACRS MEETING
WASHINGTON, D.C.
JANUARY 6-8, 1972

1.0 EXECUTIVE SESSIONS

1.1.2 Chairman's Report: Status of ACRS Report on the AEC Interim Acceptance Criteria for Emergency Core Cooling Systems.

1 reported on the December 15, 1971, meeting of ACRS representatives 1 with Chairman Schlesinger and other members of the Commission for the purpose of transmitting and discussing the ACRS letter report[1] from the 140th meeting.

The ACRS representatives agreed to withhold the report in order that the Committee might consider Commissioners' suggestions regarding (1) the desirability of including a more positive statement with regard to the safeguards adequacy of reactors designed in accordance with the Interim Criteria, and (2) the possibility of greater exposure of minority commenters to subpoena during rulemaking hearings.

Although it had since been determined that rulemaking hearing procedures would not include the right of subpoena, it was the consensus of the committee that an attempt should be made to prepare a revised draft of the letter to accommodate the considerations of Item (1) above for presentation to the Commissioners during the 141st meeting. A revised report[2] was drafted by the Committee and approved without opposing vote in Executive Session on January 7, 1972. This report represented a consensus of opinion so that no additional remarks by minority members were required (as was the case in the previous report).

3.0 MEETING WITH THE COMMISSIONERS

3.1 Certificate of Appreciation

Chairman Schlesinger presented to Dr. Bush a Certificate of Appreciation in recognition of his service as Chairman of the Advisory Committee on Reactor Safeguards during 1971.

3.2 ACRS Position on Interim Acceptance Criteria for ECCS

Copies of the ACRS report were distributed to the Commission and others at the meeting, including the General Manager, General Counsel, Directors of Regulation, Licensing and Standards and members of their staffs.

Dr.Hanauer and Mr. Muntzing noted that it indicated that the ACRS feels the Interim Criteria are inadequate to protect the public health and safety and that improved systems must be considered on a case-by-case basis, which is inconsistent with the implementation of criteria. Commissioner Johnson raised a question about proceeding with the rulemaking hearing with this ACRS position, and there was considerable discussion directed toward clarifying the philosophy of the ACRS with respect to ECCS and the Interim Criteria.

The following major points were covered in the discussion.

- The report does not negate the Interim Criteria
- The interim Criteria are not, per se, sufficient determinants of the adequacy of all safeguards required to protect the public health and safety.

- The Interim Criteria are minimum requirements and are not, per se, sufficient determinants of the suitability of ECCS design. Interpretive application of the Criteria is required to be made on a case-by-case basis.

- Programs for improvement of the Criteria and,where indicated, any designs based thereon should be accelerated.

- Apparent redundancies in the report were necessary to achieve a sufficient degree of consonance with the views of Committee members to eliminate the need for additional clarifying or dissenting comments by members.

- Although report statements in reference to future matters apparently go beyond what the Commissioners intended for the Committee to include in the letter, and while the Committee was previously unaware of the precise requirements of the Commissioners in regard to content, the Committee feels that these statements are not inconsistent with ACRS responsibilities.

- It is improbable that any attempts by the Committee to revise the report would be fruitful.

- Although the Committee report has emphasized the importance of ECCS R&D, the Committee also fully recognizes the importance of all other factors critical to the design and operation of plants for elimination of undue risk to the public health and safety. The Committee's concern for these other criteria is intended to be reflected in certain statements of the report, such as those referring to "case-by-case review" and "other applicable design and evaluation criteria."

- The Committee's recommendation that ECCS design improvements should be sought and employed, to the extent practical, for future plants, irrespective of whether the proposed plant design appears to meet the Interim Criteria,is based upon awareness of the uncertainties and weaknesses of the Interim Criteria and an appreciation of the statistical relationship between accumulated reactor years and the expectation of a LOCA.

- Chairman Schlesinger accepted the Committee's report and indicated that the documentation of the Committee's views on the subject would be useful and helpful and of value in the rulemaking hearing.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 7, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON INTERIM ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER POWER REACTORS

Dear Dr. Schlesinger:

At its 140th meeting, December 9-11, 1971, the Advisory Committee on Reactor Safeguards completed a review of proposed amendments to the Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors, published as an AEC Interim Policy Statement on June 29, 1971. The proposal was considered at a Subcommittee meeting held on December 4, 1971, in Washington, D. C. Further consideration was given to the interim criteria at the Committee's 141st meeting, January 6-8, 1972. The proposed amendments add new evaluation models by Combustion Engineering, Inc. and the Babcock and Wilcox Company to the existing list of acceptable evaluation models.

The Committee concluded that the proposed amendments were acceptable, on the same basis as the original Policy Statement evaluation models. Both the original and the amended criteria involve a number of provisions which are clearly conservative as well as some which may not be conservative, but on balance reflect adequate conservatism for interim use with plants similar in design to those which have been reviewed for construction permits. The Committee believes that each plant should be reviewed on a case by case basis to determine the extent to which these criteria are satisfied or modifications to the design or operation of the plant are required.

At the time adoption of the Interim Acceptance Criteria was under consideration, the Committee had opportunity to discuss the draft Criteria with the AEC Regulatory Staff and the Commission, and to offer comments. The Committee comments, made orally to the Commission on June 17, 1971 included the following:

"The ACRS believes that the draft, 'Interim Acceptance Criteria for ECCS for Light-Water Power Reactors', dated June 17, 1971 is satisfactory for release subject to some clarification of the language.

"The Committee concurs with the suggested approach of publishing the criteria in the Federal Register as an interim policy statement by the Commission. It agrees that Alternate A setting January 1, 1968 as a cut-off date is reasonable. It agrees with the relaxation in the criteria in Section IV B as promoting greater flexibility without freezing design and codes.

"The Committee believes that the document represents an interim solution only, and that more work is required in code development, safety research oriented to LOCA-ECCS, and work on improved ECCS. The Committee is prepared to work closely with Staff and vendors through appropriate Subcommittees. We believe this approach may lead to an orderly solution of the outstanding problems.

"With regard to the course of events of the last few months, the Committee now feels that there has been an over-all gain in that the level of work on ECCS evaluation has increased and more specific goals and objectives have been established. However, the Committee believes that more experimental work is needed to supplement the further analytical development of the evaluation models.

"The Committee reemphasizes that the Commission should indicate publicly the need for continuing work on new ECC systems or significant modifications to current designs. It should be indicated that increases in power or power density will require larger margins."

Some of the restrictive assumptions imposed by the Interim Criteria have been introduced because of uncertainties in the behavior of the ECCS. The Committee believes such restrictions should be removed, as possible, either by design changes, or by demonstration that they are unnecessary. The Committee believes that it is timely for further implementation of the statement in the Commission's Policy Statement of June 29, 1971, that "in connection with the water power reactors yet to be designed and constructed the possibility of accomplishing by changes in design further improvements in the capability of emergency core cooling systems should be considered." Although the Interim Acceptance Criteria should be useful and helpful in the licensing process, the evaluation models prescribed in these criteria are recognized to have only limited usefulness as design tools for improving emergency core cooling systems. The nuclear industry should respond in a more direct fashion with realistic design methods, based upon additional scaled experiments and analytical studies. Design changes which would clearly eliminate any potential steam binding problem during a loss-of-coolant accident represent an example of the type of improvement considered to be of particular importance. The Committee recommends that design changes to improve ECCS capability should be sought and, to the extent practical, employed in plants for which construction permit applications are received in the future, irrespective of whether the plant design without such changes appears to meet the provisions of the Interim Acceptance Criteria.

The Committee believes that there is reasonable assurance that, with appropriate use of the Interim Acceptance Criteria and other applicable design and evaluation criteria, water reactors of current design can be operated without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

In its letter of January 7, 1972, the ACRS reiterated the statements made orally on June 17, 1971; the Committee also recommended that design changes to improve ECCS capability be sought and employed in plants for which construction permit applications are received in the future, even if the plants appear to meet the Interim Acceptance Criteria without such changes.

A long and controversial rule-making hearing on acceptance criteria for ECCS was held during 1972-1973. In August, 1973, the Atomic Energy Commissioners requested the ACRS to comment on the matter, and at its 161st meeting, the Committee prepared the following response.

The Committee generally supported the new position taken by the Regulatory Staff in their "Concluding Statement of Position" (April 16, 1973). However, the ACRS reaffirmed its position that

> "in the future, design changes to improve capability should be sought and to the extent practical, employed, irrespective of whether plant design without such changes appears to meet the provisions of the Interim Acceptance Criteria and the proposed changes in these criteria."

In its decision on the Acceptance Criteria for ECCS for LWRs, the AEC made relatively modest modifications in the final Staff position. In an accompanying statement, the AEC included a recommendation for work on improved ECCS.

However, the reactor designs proposed for construction in the period 1973-1977 show rather little change in this regard, except for a general trend toward smaller diameter fuel pins, which permitted the same or higher power densities with lower peak linear heat generation ratings, and hence somewhat lower calculated peak clad temperatures in a LOCA.

Despite a rather considerable experimental and analytical effort in the ensuing five year period, the basis by which performance of an ECCS is determined has remained imperfectly defined; and for the PWR, for example, it has remained a controversial matter whether the possible effects of water carryover to the steam generator, with a resultant increase in steam binding, might not lead to a worsening of performance, if both cold and hot leg injection of ECCS water were used, as in the German LWRs.

In addition, the NRC Safety Research Program refrained from any aggressive pursuit of possibly improved ECCS features, taking the position that the law required that the NRC do only "confirmatory" research. And the Electric Power Research Institute, the research arm of the utilities, chose not to pursue such possible improvements.

What changes will be produced by Congressional action in 1977 requiring that the NRC develop a safety research program on new and improved safety concepts and features remains to be seen.

In the meantime, the Regulatory Staff has been applying the Acceptance Criteria as if they were "the law." It has permitted relaxation in restrictive features included in the evaluation models first accepted following the AEC decision in 1973, as specific experiments indicate a basis for such relaxation of an individual (or piecemeal) basis. It can be argued that the NRC safety research program or LOCA-ECCS, rather than "quantifying" the safety margins in the existing LOCA-ECCS, is being used to reduce these margins where the Staff believes such reduction is justified.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 10, 1973

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

Dear Dr. Ray:

In response to your August 30, 1973, letter, the Advisory Committee on Reactor Safeguards at its 161st meeting, held on September 6-8, 1973, submits the following comments with regard to Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light-Water-Cooled Nuclear Power Reactors. The Committee previously commented on the Interim Acceptance Criteria for ECCS in its report of January 7, 1972. The Committee has also addressed matters related to the Interim Acceptance Criteria in its reports on various reactor applications during the past two years and in its response to a set of questions addressed to it by the Hearing Board on Docket No. RM-50-1.

The ACRS believes that the Concluding Statement of Position of the Regulatory Staff's Public Rulemaking Hearing on Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors (April 16, 1973) represents an improvement over the Interim Acceptance Criteria. The Committee notes that some sections still require interpretation. The Committee believes that the Position should represent a living document which will be responsive to added inputs of substantive information acquired through both analytical and experimental studies. The Committee believes also that it is essential that the Position provide flexibility for changes on a reasonable time scale.

In general, the Position retains previous restrictive assumptions, and several others have been added. Although there is evidence to support less restrictive evaluations, the Committee believes that a more substantial demonstration in terms of analyses and experimental data is required before relaxation of the restrictions can be effected. In achieving these goals, the development of the needed technical bases would be considerably enhanced if the proprietary aspects could be minimized.

The Committee continues to believe that more work is required on code development, safety research oriented to LOCA-ECCS, and improved ECCS. The Committee wishes to emphasize its opinion that a major portion of current and future research on LOCA-ECCS should be oriented toward the development of significant improvements in ECCS.

The Committee reaffirms its position that, in the future, design changes to improve ECCS capability should be sought and, to the extent practical, employed, irrespective of whether the plant design without such changes appears to meet the provisions of the Interim Acceptance Criteria and the proposed changes in these criteria. In its approach to nuclear safety, the Committee has sought to make allowances for the state-of-the-art knowledge on issues, to encourage further acquisition of background knowledge, to determine whether added safety improvements or margins are achievable, and then to encourage incorporation of practical improvements. It is the Committee's judgment, that for an expanding nuclear industry, the cumulative effects of the added improvements represent prudent goals. In particular, reactors proposed for standardized units should incorporate design improvements.

The Committee views the following as examples of measures which contribute to design improvements: 1) improved reliability of the ECCS and system components, including approaches intended to minimize the potential for common failure modes; 2) reactor core designs and operating modes which reduce the potential for high temperatures, clad swelling or perforation in postulated LOCAs; 3) ECCS whose proper functioning is relatively insensitive to reactor or ECCS design parameters and to proper functioning of other components such as steam generators or reactor containment; 4) ECCS having redundancy, diversity and abundance of flow such that its adequacy is subject to evaluation without undue requirement for complex evaluation techniques; and 5) other measures which further reduce the probabilities and consequences of a LOCA.

For existing ECCS designs, the ACRS encourages the ongoing efforts to develop evaluation methods, which are subjected to rigorous quality assurance measures, which are validated by experiment and theory, and which receive independent confirmation by the Regulatory Staff.

Acceptance Criteria for ECCS are not viewed by the Committee in isolation from other applicable design and evaluation criteria and guides. The Committee seeks to identify and to evaluate all pertinent factors for each case reviewed, as well as to be responsive to new information and experience. As noted in the Committee's December 18, 1972, report, Status of Generic Items Relating to Light-Water Reactors, resolution of some items on a generic basis is still pending. Other items which have been identified, studied or are under study, include pressure vessel integrity, steam generator tube integrity, reactor coolant pump overspeed, independent capability to analyze the ECCS, evaluations of containment pressures, independent analysis of the ice-condenser containment time-pressure history, and structural responses to a LOCA. All such items are taken into account in arriving at Committee positions.

For those plants which are licensed to operate under acceptance criteria such as those of the Regulatory Staff's Position, the Committee recommends continued study of means to improve ECCS reliability and performance by such design and operational measures as appear practical and significant. When such studies indicate that a significant safety improvement can be achieved, it should be considered for backfitting on a timely basis in accordance with the Commission's backfitting policy.

The evaluation of ECCS capabilities must, of course, include the possible effects arising from fuel densification. Several restrictive assumptions have been introduced by the Regulatory Staff in their densification evaluation models that, when combined with the Interim Acceptance Criteria, result in more severe limitations being placed on maximum linear heat generation rates for PWRs and maximum average planar (bundle) linear heat generation rates for BWRs. These restrictions result in the requirements for more intensive incore monitoring. In some cases, such restrictions have also led to lower initial power ratings for reactors having operating licenses. In this new phase of bringing into operation reactors having both higher power and higher power density, the Committee agrees that some initial limits on power may be appropriate in achieving the following objectives:

a) prudent approach for gaining operating experience
b) more thorough demonstration of fuel behavior
c) more detailed measurements of maximum linear heat generation rates and peaking factors for PWRs, and maximum average planar (bundle) linear heat generation rates for BWRs
d) resolution of more generic items, and
e) implementation of the proposed new requirements for the acceptance criteria, including evaluation of analytical and experimental data relating to model and code developments.

The initial restriction on power ratings and/or on flexibility for power operations and the duration of these restrictions should be evaluated on a case-by-case basis until a standardized approach has evolved through implementation of the Position.

The Committee believes that the Concluding Statement of the Regulatory Staff represents an acceptable Position, that possible ECCS design improvements should be pursued for plants in operation or under construction, and that reactors filing for construction permits after January 7, 1972, should have significantly improved ECCS capability.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References:

1) Concluding Statement of the Regulatory Staff entitled "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Reactors," April 16, 1973

2) Final Environmental Statement concerning the Proposed Rulemaking Action: Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, dated May 9, 1973 (Volumes 1 and 2)

3) Supplementary Concluding Statement of Position of the Regulatory Staff's Public Rulemaking Hearing on Acceptance Criteria for ECCS for Light-Water-Cooled Nuclear Power Reactors, dated August 9, 1973

6.3 LWR SAFETY RESEARCH

Reactor safety research, in a general sense, began essentially as soon as nuclear reactors began. Safety studies were done by many of the groups working on reactor development. At the third meeting of the statutory ACRS in December, 1957, there was a major presentation made on a new, fairly comprehensive program on the safety of fast reactors. And prior to this time, there had been major experiments performed relating to the behavior of light water reactors during severe reactivity transients, for example, the BORAX 1 destructive experiment and the subsequent series of SPERT experiments (Thompson, 1964).

At the 11th ACRS meeting, November 6-8, 1958, Mr. Hembree of the Division of Reactor Development reported the total dollar volume of the Experimental Reactor Safety Program as being $7 million in fiscal year 1958, $8 million in fiscal year 1959, and $11 million in fiscal year 1960. This program covered three areas, namely rector kinetics and control, chemical reactions, and containment. During the meeting, a presentation was given on the work at ORNL on fission product release fractions (and behavior thereafter) from molten fuels.

At the 12th meeting, December 11-13, 1958, the representatives of Atomics International reviewed their safety research work on the development of a fuse (for backup shutdown of reactors).

At its 21st meeting, November 12-14, 1959, the ACRS wrote a letter report to AEC Chairman McCone in which it recommended that a comprehensive review be prepared of the available information on reactor safety. (This recommendation culminated several years later in publication of the book "The Technology of Nuclear Reactor Safety.")

At its 34th meeting, May 18-20, 1961, the ACRS wrote a letter report to AEC Chairman Seaborg calling attention to problems of reactor vessel embrittlement which could arise with high neutron irradiation doses.

In 1961 the post of Assistant Director for Reactor Safety Research was established within the AEC Division of Reactor Development, and at the 45th meeting, December 13-15, 1962, the ACRS sent a letter report to AEC General Manager Leudecke concerning aspects of the AEC safety research program. The letter is reproduced on the following pages.

The letter of December 31, 1962 shows the strong interest of the ACRS at that time in incresed knowledge concerning large destructive reactivity transients. Such transients were not a "design basis" or "maximum credible accident" for the light water power reactors then under regulatory review. However, a damaging reactivity transient had occurred in the small, experimental boiling water power reactor, SL 1, the previous year (Thompson, 1964); and reactivity transients had probably been the major safety issue during the 1950's.

At its 48th meeting, July 11-13, 1963, the ACRS completed a second report on reactor safety research to General Luedecke, as is reproduced on the following pages.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

December 31, 1962

A. R. Luedecke
General Manager
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REVIEW OF REACTOR SAFETY RESEARCH PROGRAM

Dear General Luedecke:

The Advisory Committee on Reactor Safeguards has completed the following stages of a review of the program in reactor safety research that is supported by the Division of Reactor Development:

1. A complete summary of the program by Dr. J. A. Lieberman and his branch chiefs. (Summary Report, Nuclear Safety Research & Development Program, Division of Reactor Development, June 1962, by I. E. Jackson, Jr.)

2. Review of the Spert and Step programs at the National Reactor Testing Station presented by the Phillips Petroleum Company's research group and others.

In addition, a number of reports on the Spert program have been made available to the Committee. Recently, information on tests of fission-product release and transport has been received but this has not yet been studied by the Committee.

The Committee commented on the safety aspects of the Spert I destructive tests in our letter to Chairman Seaborg dated August 30, 1962. The present letter presents further opinions on the conduct of individual tests at the NRTS, and provides the initial response to a request from the Director, Division of Reactor Development, for Committee comments on the entire reactor safety research program.

General Remarks on the Spert Program

The Committee believes that the present series of Spert destructive tests, which are carried out with small fission-fragment inventories, cannot constitute a serious hazard to the health and safety of the public. Such experiments provide information very useful to the understanding of reactor accidents, and any delay of such experiments increases the probability that safety evaluations of other reactor projects may be erroneous due to lack of the new information. It is therefore recommended that review of such experiments by the ACRS, and possibly by other safety groups within the AEC, be eliminated if the operator of the experiment files with the AEC a document showing that the radiation limits specified in 10 CFR Part 20 will not be exceeded for the general public, even in case of dispersion to the atmosphere of the whole fission fragment inventory of the reactor at any time throughout the test. This simple procedure should be modified at the request of the AEC, the ACRS, or the contractor, if unusual circumstances so warrant.

The Committee also suggests that a planned program of press releases be instituted to educate the general public that releases of fission products from tests of the Spert type will not constitute an undue hazard to anyone either on or off the site. The publicity should emphasize the point that such releases are an integral part of a reactor-safety research program whose objective is to protect the health and safety of the public.

Specific Recommendations on the Spert Program

The Spert group has built up experience and skill. There are many results which give a good basis for planning future tests of this type and this part of the whole program should grow. The analytical part of the work is making progress, but it would profit from further strengthening. For example, more adequate computer facilities should be provided.

The Committee recommends that destructive tests be performed as soon as possible on low-enrichment oxide cores, since such cores are used in the majority of power reactors. The tests on highly enriched metal cores have to be relatively widely spaced to allow adequate analysis between tests. Since oxide fuel for about two cores is available at the Spert facility, consideration might be given to performing destructive tests on oxide cores while the analysis of a metal-core test is underway.

The Committee suggests that the following areas be studied:

1. The influence of slow acting, small, positive temperature and void coefficients (in particular positive coefficients extending over a limited range of temperature and voids) on destructive or other severe transients. Such limited positive coefficients may prove to be without much influence on these transients; and, if this is the case, they may be used to reduce the reactivity change from cold to hot-operating. Such reduction would improve the reactivity lifetime and economics of the reactor, or it could be used to reduce the excess reactivity that has to be controlled.

2. The possible existence of mechanisms by which catastrophic local disturbances in a large reactor can propagate. Perhaps large reactors could be built in such a manner that destruction of more than a small part of the reactor is demonstrably impossible.

The Committee wishes to encourage a program consisting of destructive testing of fuel assemblies, and employing a re-usable reactor as source of the neutron burst in order to supplement the whole core destructive tests. Such a program for the testing of small fuel assemblies is in progress at the KEWB facility. The Fast Burst Facility, proposed by Phillips Petroleum Company, would allow testing of somewhat larger assemblies. The fuel assembly destructive testing program would have the following advantages:

a. It would save the cost of loss of material and cleanup involved in tests of whole-reactor destructions.

b. It could provide answers more rapidly than can be obtained in the case of whole-reactor destructive tests.

c. If tests were performed with "dirty" fuel, only a small amount of radioactivity would be dispersed.

d. It would extend the range of available reactor periods into the important region of fast transients by more than an order of magnitude beyond that available at the transient facilities which are now in operation and could accommodate large samples.

The recent Spert I destructive test seems to indicate that the destructive effect is separate from the reactivity-feedback effects, and it is essentially this feedback which requires whole-reactor tests.

The Step Program

The Step program is in the formative stage. It is our understanding that the loss of coolant accident will figure prominently in these tests. The Committee recommends that the investigation of this potential accident be carried out in two steps:

(a) Coolant loss from the reactor, at a controlled rate and correlation of this rate of loss with quantitative information on the behavior of the core;

(b) Study of the nature of piping failures and their effects on the rate of coolant; this study should take full advantage of and should not duplicate work done outside the Step group.

Inasmuch as accident analyses usually assume that the pressure vessel containing the reactor will not fail, and since brittle failure of this vessel may lead to catastrophies far in excess of the "maximum credible accident", the Committee recommends that additional support be given to the groups now investigating brittle failure starting at defects in pressure vessels. Of particular interest would be tests using pressurization by gases rather than liquids and the effect of environment on crack propagation. Continued attention should be given to radiation damage on pressure vessels and the study of the effects of the significant variables on radiation damage specimens.

The results of the Reactor Safety Research Program are important in the work of the ACRS. Quantitative data of general applicability are required to permit precise evaluation of reactor safety and for design and operation of economic nuclear power systems. The reviews to date have been extremely valuable to us.

Sincerely yours,

/s/ F. A. Gifford, Jr.

F. A. Gifford, Jr.
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

August 1, 1963

A. R. Luedecke
General Manager
U. S. Atomic Energy Commission
Washington, D. C.

SUBJECT: REVIEW OF REACTOR SAFETY RESEARCH PROGRAM

Dear General Luedecke:

The Advisory Committee on Reactor Safeguards has been reviewing the safety research program of the Atomic Energy Commission for several months. This review, undertaken at the request of the Division of Reactor Development, has been very timely because it has come during a period when engineered safeguards are increasingly used to justify sites that would otherwise be unacceptable. At the outset, the Committee wishes to thank the Division of Reactor Development for the opportunity to hear of the progress being made in reactor safety research and to comment on it. In a previous letter to you, dated December 31, 1962, some views of the Committee on the Spert and STEP projects were transmitted. In this letter additional comments are offered only on those portions of the program designed to provide further information on the release of fission products from fuel elements and the place and extent of their removal in passing through successive spaces and barriers.

The Committee would like to stress that dependence on engineered safeguards to reduce the effects of credible reactor accidents must be supported by confidence that the safeguards would act as expected. There must be assurance that the conditions to which the devices would be exposed are correctly foreseen. The effectiveness of the devices under these conditions must be established.

The safety research program devotes much of its attention to questions about the nature and magnitude of fission product releases by various mechanisms. Most of the research concerns the release of fission products by fuel that has been melted by afterheat. In most instances, the fission product heating is simulated by other means: plasma torches, electrical heating, induction heating. Several kinds of fuel are being investigated. Those being studied and proposed for study include the principal reactor fuels for the converter reactors.

In the spirit of the statement made above, that confidence in performance as planned is essential, the Committee would like to comment on aspects of these fission product release studies. Most of the comments simply reiterate views on which the research has been based, and these are stated again here only for completeness.

The two basic questions to be answered by fission product release studies are: (1) how much of what fission product of significance is released, and (2) in what form are the fission products released? There are no simple answers to these questions, because the answers depend on a variety of environmental conditions. These include: (1) the chemical composition of the fuel (e.g., uranium metal, uranium oxide, uranium carbide, alloying constituents), (2) the physical nature of the fuel (for instance, sintered oxide or vibratory compacted oxide), (3) the degree of burnup, (4) the temperature history of the melt, (5) the kind of cladding, (6) the kind of atmosphere in which the melt takes place (air, steam, air-steam mixture, noble gas). Amplification of the basic questions in the light of the environmental effects leads to such questions as: What fraction of the release of volatile fission products, particularly halogens, is in elemental form? What is the particle size distribution of released nonvolatiles? What is the expected degree of adsorption of volatiles on these particles? What chemical compounds are formed? What is the size distribution of the particles with which these are associated? The answers as functions of the environmental conditions must be known if the behavior of the engineered safeguards is to be assured.

In addition, the Committee would like to draw attention to the presence of large amounts of plutonium and other transuranic species near the end of reactor core life. The possible release of these, the effect of their release, and their effect on fission product release should be studied.

Throughout, care must be taken to assure that the history of significant fission products is followed. In circumstances where halogens are released in easily removable form, the effectiveness of the engineered safeguards will probably depend on other fission products.

It appears that some increase in the Atomic Energy Commission's safety research program will be needed if satisfactory answers to the above questions on fission product release are to be available for interpreting the consequences of integral experiments such as LOFT.

The release of fission products from fast reactor fuels should also receive growong attention. The fission product distribution curve differs somewhat with fission neutron energy. Fission product yields from plutonium fission are somewhat different from those from U^{235} fission. The fuels themselves will differ from those used in thermal converter reactors.

Before leaving the subject of fission product release research, the Committee would like to comment on the proposed studies of fission product release by methods other than simple fuel melting. The releases associated with nuclear excursions or chemical reactions (such as those between water and metals) will differ from the ones discussed above. The series of Spert destructive tests will shed some early light on the nature of such releases as well as on other questions. But the basic physical understanding of the releases will depend on research such as is projected for the Power Burst Facility. The Committee wishes to emphasize the need for the PBF, and to support its early construction and use.

The retention of released fission products on the inner walls and internals of the reactor vessel will reduce the magnitude of the release by an amount that is so far unpredictable. This reduction factor will depend on complicated circumstances: The geometry and composition of the surfaces, the form of the fission products (gaseous, elemental, particulate), the temperatures of surfaces, the size of the reactor vessel or pipe rupture, and the atmosphere in the reactor vessel. It will be necessary to identify the cause of the reduction, to establish the dependability of results. The Nuclear Safety Pilot Plant should help to answer a number of the questions influencing the expected retention of fission products in the reactor vessel, but it may be that the complication of structural members and fuel element surfaces will only lead to a lower limit on the advantage to be gained from vessel retention. The need for careful control in these experiments is stressed. It is noted that a plasma torch will be used to melt the fuel. This torch will be located in a separate chamber outside the simulated reactor vessel. Attention has been given to assuring that the release into the simulated vessel resembles that from an afterheat meltdown: this must be assured. The variable nature of the release as influences by features of the melting, discussed earlier, should be taken into account. The course of the deposition in the vessel should become well enough understood on purely physical and chemical grounds to permit mathematical justification of vessel retention factors that might be assumed in reactor plants.

Beyond the escape from the vessel, released fission products to be a major hazard must still escape whatever containment or confinement is provided. The tests of retention by containment or confinement, and the effectiveness of air cleanup devices under actual conditions, are planned for the LOFT facility. Releated test facilities have also been proposed: the Pressure Suppression Facility, and more recently, Spert-II.

It is difficult to specify the features important in finding the degree of retention in the containment or confinement building except by reiterating the need to justify whatever retention factors may be claimed in the future. This justification must rest on a foundation of physical and chemical understanding. The same sources of complication as pertain to vessel retention factors will also apply here.

In view of the recent finding of almost total release of several significant fission products, transport effects assume very great importance. The various engineered safeguards that have been proposed to reduce further the extent of final release should be tested under conditions under which they must be expected to perform. These engineered safeguards include spray washdown systems in the vessel and in the reactor building, building air recirculation systems, and final air cleanup systems. The variability of possible fission product releases will affect the performance of all of these. The temperature and steam content of the atmosphere will affect the performance of recirculating and final air cleanup systems. The possibility of saturation of air cleanup systems should be investigated. The rate of re-evaporation of halogens washed down by spray systems should be known.

The pressure suppression scheme that has been designed for some reactors bears further testing over a somewhat larger range of variables. In relation to the fission product retention problem, however, it would be useful to establish experimentally to what degree this scheme can be relied on for reduction of fission product escape.

The proposed Pressure Suppression Facility seems to be the one device that has been proposed for systematic study of the effectiveness of engineered safeguards such as building spray systems, air recirculation cleanup systems, and pressure suppression. The Committee wishes to encourage further development of this proposal, with emphasis on the goal of physical justification of the reduction factors to be assigned such engineered safeguards.

The Committee views the LOFT experiments as being in the nature of necessary full system tests. These would establish whether the more specific research on the individual and successive aspects of the core meltdown and fission product release have made it possible to predict accurately the complete sequence of events, and whether any effects of importance have been overlooked. As corollaries to this view, the Committee believes that the research that is to be correlated on a full system basis must be at an adequate stage for this test when it is performed, and that the LOFT experiment must be well instrumented to establish quantitatively the physical and chemical nature of the release from the fuel, the vessel, and the building, and the environmental features influencing the release. It is doubtful that a single LOFT meltdown will be adequate to provide the confidence in predictability of the magnitude and kind of fission product releases after core meltdown.

The proposed use of Spert-II to provide some information prior to the LOFT experiments would be of questionable value. Because the basic experiments needed for interpretation would almost surely not be finished in the two years before a Spert-II meltdown can be done, this test could not be considered as a systems test of the nature of LOFT. Without the physical understanding of the more elementary processes, any results achieved could not be depended on as guides to predicting fission product releases following meltdown of other reactors. It seems that at best a Spert-II meltdown might give some further guidance to the conduct of the later LOFT tests.

The Committee has been favorably impressed by the emphasis that the Division of Reactor Development gives to research on nuclear reactor safety. This research should be of real value in helping to ease the problems of reactor siting and the assured performance of engineered safeguards.

The Committee will forward further comments on research aimed at reducing or clarifying the possibility of serious accidents when the review of these portions of the research program has been finished.

It is clear that some facets of reactor safety are more important than others, and the degree of urgency in attaining useful results varies.

In the near future, the Committee will forward to you their views on the coverage of these facets, on the general scope of the program, and the relative emphasis that should be placed on various aspects of the program. This critique will be based on our opinion of the relative importance of safety problems being faced in the siting, design and construction of large power and test reactors.

Sincerely yours,

/s/

D. B. Hall
Chairman

COPY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 19, 1963

A. R. Luedecke
General Manager
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REVIEW OF REACTOR SAFETY RESEARCH PROGRAM

Dear General Luedecke:

In its letter of August 1, 1963, the Advisory Committee on Reactor Safeguards stated an intent to comment further on the Reactor Safety Research Program. Some additional comments, which are now transmitted, represent views of the Committee on the area of research that requires the greatest emphasis, with some discussion of the bases for these views.

The Committee believes that it is of primary importance to determine to what extent engineered safeguards can be relied on in relaxing reactor site restrictions.

In the light of present knowledge, it seems unlikely that general principles will render incredible the possibility that high power nuclear reactors can have large power excursions, or that they can have substantial core meltdown. Therefore, it must be expected that the safety analysis for locating and designing nuclear reactors will continue to assume such accidents to be possible, even if only remotely so.

Reactor accidents leading to temperatures and pressures representative of nuclear weapons can be considered incredible on physical grounds. Also, reducing the direct radiation effects of nuclear excursions to tolerable levels seems to pose no insuperable design problems. Primary attention will have to be given to potential release of fission products to the environment.

proof tests such as LOFT may make a contribution. It is doubtful that experiments done with the Nuclear Safety Pilot Plant will by themselves provide the upper limits relevant to large scale core meltdowns in real reactor vessels.

As indicated in our letter of August 1, 1963, the determination of decontamination factors for air cleaning systems and similar devices under actual conditions of release to the atmosphere is an important research area.

Some of the other major studies that will elucidate the probability and severity of accidents and hence the chances of fission product release are the following:

1. Research on the probability of gross rupture of primary pressure vessels and other pressurized components is needed. Information is needed on methods to protect the containment or confinement from possible missiles.

2. Further studies of the brittle-ductile transition of steels are needed. The effects of radiation, radiation rates, radiation under stress, and welding variables on the brittle-ductile transition phenomenon need further exploring. The results need to be analyzed both in terms of fracture stress and energy absorption. More information on the change of energy absorption and crack propagation with irradiation rate would be useful.

3. The SPERT-I destructive test on November 5, 1962, showed evidence of an unexpected threshold phenomenon that increased the destructiveness of a nuclear excursion. The nature of this phenomenon should be clarified. The existence of other threshold phenomena should be watched for in subsequent SPERT-type destructive tests on water cooled systems.

Recent renewed emphasis on the long range role of large fast breeder reactors points up the need for a well developed, long term, comprehensive research program on the safety of such reactors. A strong research program started now should develop information very useful to the first generation of very large fast reactors. Some of the matters carrying special safety implication are as follows: The Doppler coefficient; reactivity effects due to coolant voids and fuel movement;

At its 52nd meeting, January 9-10, 1964, the ACRS heard encouraging results concerning the ability of the low-enrichment uranium oxide fuel and SPERT reactor to withstand a severe reactivity transient with essentially no damage. As expected, the transient was terminated by the Doppler reactivity effect; and the experience was directly applicable to the PWRs and BWRs under consideration in the regulatory process.

In a letter to AEC General Manager Hollingsworth dated September 17, 1965, the ACRS provided some detailed suggestions concerning the proposed LOFT core melt experiment, as well as comments concerning the development of improved containers for the shipment of highly irradiated fuel, improved methods of inspecting pressure vessels, and improved reliability for a large number of reactor components, including containment penetration seals and emergency power supplies.

In the summer of 1965, the AEC had formed a Steering Committee on Reactor Safety Research, composed of the members of the Regulatory and the reactor development sides of the commission with the charter (in part) of developing information and criteria pertaining to metropolitan siting of reactors. This group met with the ACRS at the 70th meeting, February 10-12, 1966 and the 72nd meeting, April 4-6, 1966, to discuss implementation by the AEC of recommendations made by the ACRS in its letter of November 25, 1965, on reactor pressure vessels. The AEC stated they were initiating a very considerable program in the pressure vessel area, aimed principally at gaining improved integrity, and that discussion papers on missiles, metal-water reactions, emergency core cooling, air cleaning systems, containment, and seismic effects were under preparation by ORNL.

At its 78th meeting, October 6-8, 1966, the ACRS completed a very significant letter rept to AEC Chairman Seaborg on reactor safety research. This letter, which was prepared after "resolution" of the "China Syndrome" issue for Dresden 3 and Indian Point 2, and agreement to the establishment of a task force (as discussed in previous chapter placed major emphasis on research on core melt phenomena, on LOCA-ECCS, and on improved primary system integrity. The letter is reproduced on the following pages.

2. Because of the importance of emergency core cooling as an engineered safeguard, studies on core cooling processes already underway within the AEC and industry should receive continued attention. Coolant distribution and heat transfer phenomena which could influence emergency cooling significantly should be examined to remove existing uncertainties, including those related to an assumed course of events where cooling is marginal or inadequate in sections of the core. Tests of actual spray cooling and core flooding systems under accident conditions warrant careful consideration.

3. Development of practical, effective methods for extensive periodic inspection of pressure vessels is of great importance. The current program in AEC and industry should be augmented, as necessary, to assure this. One or more practical systems for such inspection should be developed as soon as possible.

4. A strong program on the properties, homogeneity, and behavior of thick steel pressure vessel sections, including research areas described in the recently proposed program of the Pressure Vessel Research Committee, should be implemented by industry and the AEC. The work on thick-walled vessels should include a thorough study of potential failure modes under pneumatic loading for various flaw sizes and types, and the significance of the reduction in the energy absorption shelf as a function of neutron irradiation.

5. Because the Commission may be called upon to consider proposals to construct reactors utilizing prestressed concrete pressure vessels, the nuclear industry and the AEC should promptly institute a very active safety research program into such vessels, including their design for seismic effects. This program should include research into anomalous failure modes of such vessels, particularly under pneumatic loading. This work should encompass effects of potential structural defects or overloads and problems associated with closures, penetrations, and anchors.

6. The further development of advanced methods of calculating destructive reactivity transients in water-cooled reactors, including predictions of damage to the primary system, is recommended. Important phenomena, such as the mode, time-sequence, and effect of fuel element failure, should be identified and studied so that the phenomena are dealt with adequately in the over-all analyses. Also, the role of space-dependent kinetic effects should be fully identified.

When it reaches the operational stage, the planned PBF program should play an important part in identifying fuel failure modes. The large transient experiments in the Spert program have already been very useful. Further experiments with low-enrichment-fuel water reactors

Some computational methods are already being developed in connection with various safety research programs. Others could be developed. In expanding this program, the Division of Reactor Development and Technology should work closely with the Regulatory Staff to establish an appropriate series of standard methods or reference problems.

5. In view of the large amounts of recycled plutonium fuel that will probably be used in thermal reactors in the future, potential safety problems arising from the use of such fuel should be identified and appropriate information developed in timely fashion.

6. Since early detection of small leaks in primary coolant systems of reactors can provide considerable protection against more serious difficulties, existing leak detection methods should be evaluated from the safety standpoint and new techniques developed, if appropriate.

7. The dilution, dispersion, and transport of liquid radioactive wastes in surface waters (rivers, lakes, estuaries, bays and open ocean) are important factors in the siting of nuclear reactors. In addition to these phenomena, attention frequently needs to be directed toward biological concentration of radionuclides in aquatic life. It may be desirable to review previous work on this subject, including related reaearch on discharge of municipal and industrial liquid wastes. Preparation of a state of the art review of current knowledge, and delineation of areas where further research is needed, would be useful. A special evaluation of the impact of siting many reactors on the shores of the Great Lakes, in relation to retention and flushing characteristics and to accumulation of radionuclides in aquatic organisms, may also be desirable.

It should be noted that information developed in connection with several items listed above would not only help to enhance public safety, but would also contribute toward a more expeditious review of the large number of reactor projects anticipated for the future.

Sincerely yours,

Original signed by
David Okrent

Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 14, 1967

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: COMMENTS ON WATER REACTOR SAFETY RESEARCH PROGRAM

Dear Dr. Seaborg:

In response to a request from the Director, Division of Reactor Development and Technology, the ACRS has reviewed the draft document entitled "Water Reactor Safety Research Program, Summary Description" and has transmitted comments on this program in a letter to the Director, DRD&T, dated April 14, 1967.

A copy of this letter is attached.

Sincerely yours,

/s/

N. J. Palladino
Chairman

Attachment:
Letter from N. J. Palladino, Chairman, ACRS to Mr. Milton Shaw, Director, DRD&T, dated April 14, 1967.

The ACRS recommends that all of the above areas of safety research be prosecuted vigorously.

Sincerely yours,

/s/

N. J. Palladino
Chairman

Dr. Beck (REG) pointed out that at no time in the last 20 years has the AEC been able to answer all safety questions. He stated that in the past we have assumed pessimistic conditions that we felt would not occur. He asked the participants to identify the urgent problems that are amenable to resolution. He said that if the AEC must have all of the answers then we must stop building reactor plants.

Mr. Case (DRS) asked if the R&D proposal was required to assure safety or was it to determine the degree of safety.

The representatives of the reactor designers asked what the AEC wanted in the way of safety margin. They expressed their opinion that they have sufficient information for current designs and plan no major changes in the immediate future. They expressed their opinions that their designs are adequately safe without further major R&D. They said that if the AEC did not agree, then the AEC should provide information as to what it considered to be adequate safety.

COPY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

March 20, 1969

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: COMMENTS ON WATER REACTOR SAFETY RESEARCH PROGRAM

Dear Dr. Seaborg:

During the past year the ACRS has been reviewing various aspects of water reactor safety research, including the January, 1969 draft document from the WRSPO entitled "Preliminary Water-Reactor Safety Program Plan". Committee comments on the water-reactor safety research program have been transmitted to the General Manager in a letter dated March, 1969, a copy of which is attached.

The Committee wishes to take this opportunity to emphasize the continued great importance to the health and safety of the public of the Commission's Nuclear Safety Research Programs on water reactors and on other power reactor types, and urges continued vigorous support of this work.

Sincerely yours,

/s/

Joseph M. Hendrie
Acting Chairman

Attachment:
ACRS Letter to Mr. Robert E.
Hollingsworth, dated March,
1969

Consideration should also be given to research aimed specifically atimproving the potential for siting of large water reactors in more populated areas than currently being utilized. For example, studies should be undertaken to develop reactor design concepts providing additional inherent safety or, possibly, new safety features to deal with very low probability accidents involving primary system rupture followed by a functional failure of the emergency core cooling system.

The ACRS agrees with much of the general emphasis placed by the Plan on various facets of water reactor safety research. However, the Committee believes that, in view of overall funding limitations, the considerable expenditures projected for the LOFT facility and its nuclear experimental program warrant re-evaluation with the benefit of a careful review of the specific objectives and anticipated accomplishments of this portion of the program. The Committee also believes that more effort should be devoted to gaining an understanding of modes and mechanisms of fuel failure, possible propagation of fuel failure, and generation of locally high pressures if hot fuel and coolant are mixed, and that effort should commence on gaining an understanding of the various mechanisms of potential importance in describing the course of events following partial or large scale core melting, either at power or in the unlikely event of a loss-of-coolant accident.

The Committee urges that increasingly strong direct support be given the Regulatory Staff by capable, experienced personnel working in the AEC's safety research program. Quantitative evaluation of safety questions arising in construction permit applications will not only help ease the current workload problem of the Regulatory Staff but should make for more meaningful research efforts.

Comments on specific sections of the draft Program Plan follow:

Section 1. Accident Prevention

The ACRS believes accident prevention to be an important matter and generally supports the high priorities placed by the Plan on most parts of this section. Specific attention and considerable emphasis should be placed on questions relating to the integrity of core structural support members in operation or under accident conditions.

1.1.1 Material Properties

The current strong programs in industry indicate that the lead role in 1.1.1a, b and c probably should be shared by AEC and industry. Topic 1.1.1.d on the effects of cleaning agents warrants careful pursuit.

important that improved means be developed for testing and assuring the workability of these systems in the unlikely event of a serious accident. The safety research program should continue to probe for unexpected phenomena and to attack gaps in our knowledge, such as the possibly rapid mechanical interaction between molten fuel and water.

New means of assuring adequate core cooling under still more severe postulated accident conditions, such as large pressure vessel leaks or severe core deformation, should also be studied.

2.1 Decompression and Heatup Prior to ECCS Initiation

2.1.1 Behavior of Coolant

The Committee supports a priority of A to studies of coolant behavior, since such knowledge is essential to prediction of blowdown forces, heat removal, and other phenomena. Mechods of predicting flow behavior in the unlikely event of large pressure vessel leaks should be considered for potential future use.

2.1.2.2 Temperature of the Primary System and its Internals

Experiments which assure conservatism rather than precise prediction are needed.

Although there is a limit to the amount of detail that should be incorporated in the Program Plan, it seems advisable in 2.1.2.2b to indicate at least the approximate maximum temperatures, power densities, kw per foot values, and similar conditions to be investigated within the experimental efforts described. This would help to make clearer the intended extent of validation of analytical models anticipated, and would amplify recognition of the assertion that the need for improved predictions of thermal transients during blowdown and heatup arises from the sensitivity of ECCS performance to the potential for higher power densities in future reactors (bottom p. II-2-14 and top of p. II-2-15).

2.1.3 Mechanical Response of the Primary System and its Internals

The question of fuel rod response (both prior to initiation of and during ECCS injection) in respect to distortion, ballooning, perforation, etc. is one of the more critical questions still requiring resolution. The experimental work being undertaken to provide such resolution, as described in 2.1.3.1b (and 2.2.2.1b), appears inadequate in extent and urgency. Also, the lead role is assigned to industry. The AEC should at least share the lead role in this vital area.

3.3 Structural Design

No AEC safety research effort appears to be warranted with regard to current license applications.

3.4 Antiseismic Design

The Committee continues to support the need for research in support of antiseismic design.

3.5 Leak Tightness

The Committee believes that industry efforts are warranted to assure leak tightness under accident conditions, particularly to assure acceptable behavior of penetrations in the presence of steam and of accident temperatures.

3.6 Containment of Molten Core

No AEC work is currently planned on containment of a molten core. Interest in this question continues, however, and the problem may be of more critical concern for large reactors in much more populated locations than are used at present. Some work in this area, in the nature of scoping studies of possible solutions, is appropriate. Also, research aimed at providing a better understanding of the more important phenomena involved should be undertaken. A priority of B is warranted.

Section 4. Behavior and Control of Fission Products

4.1 Release of Fission Products from Overheated Fuel

The Committee concurs in the D priority.

4.2 Prompt Release of Fission Products

Some scoping effort may be of value, especially if the effort is aimed at bounding release fractions to be anticipated for the higher burnup, higher-rated fuel which is likely in the future. A major AEC effort to make prediction quantitative does not seem justified.

4.3 Transport and Deposition of Fission Products

A priority of C is generally appropriate for the AEC program.

4.4 Fission Product Trapping in Emergency Core Coolant

A priority of C is generally appropriate for the AEC program.

Finally, in connection with fission product transport, the Committee wishes to call attention to the possible long-range need for improvements in knowledge of and methods for controlling the routine release of radioactivity to the environment and suggests that consideration be given to possible research bearing on this matter.

Section 5. Power Excursion Accidents

5.1 Reactivity Insertions

Current LWR's are designed to limit the reactivity and reactivity insertion rates which might be inserted in the unlikely event of a rod dropout or rod expulsion accident. Analytical studies might be devoted to a preliminary investigation of reactivity insertion accidents in future, larger LWR's, also to other potential mechanisms of rapid reactivity insertion, such as positive pressure or void coefficients of reactivity.

5.2 Reactor Kinetics

The theoretical study outlined on p. II-5-15 appears to be worthwhile. It should be performed partly in direct support of the Regulatory Staff with regard to current reactors, and partly to assess potential problems of future larger LWR's. Other space-time kinetic effects having safety significance for larger LWR's, such as the potential for large distortions in power distribution, should also be given preliminary study.

Studies of this kind, coupled with an assessment of uncertainties in available calculational methods of the margins likely to exist to core damage thresholds in future reactors, should provide a basis for determining the need for an experimental program in space-time kinetics.

5.3 Transient Fuel Failure Phenomena

Work should be scoping or semi-quantitative in nature, except where needs for a more extensive, quantitative program have been demonstrated. Experiments should be aimed primarily at previously unexplored or poorly explored effects, such as high-burnup fuel and fuel-coolant interactions. These studies should be coordinated with those recommended in 6.1 below.

5.4 System Damage Effects

Work on this task should depend on previous assessment of the nature of a severe reactivity excursion, including its source (5.1). In connection with evaluations of the capability of the primary system of larger, higher power density reactors to withstand pressure pulses, etc., preliminary study of accidents involving considerable fuel melting at power is warranted.

The letter reflects the general emphasis on primary system integrity and LOCA-ECCS, and again asks for work on core melt. It also recommends work on new safety features to deal with the very low probability events. A low priority is placed on fission product release and transport under accidental conditions, in contrast to the pre-1966 days. The letter recommends the availability of increasingly strong support to the Regulatory Staff by experienced personnel working in the safety research program. Actually there had been a beginning of the Regulatory Staff's Technical Assistance Program in 1968, following earlier ACRS recommendations to this effect.

In a letter to Chairman Seaborg several months later, the ACRS expressed considerable concern about a reduction in reactor safety funding for FY 1970 and 1971. This letter is of historical interest also in that it provides a pro and con discussion of the continuation of the LOFT program under conditions of financial exigency, and indicates a major division of opinion among the ACRS members themselves.

In its comments of March 20, 1969, the Committee also recommended that ". . . considerable attention be given now to the potential safety questions related to large water reactors likely to be proposed for construction during the next decade. Larger cores, higher power densities, and new materials of fabrication are some of the departures from present practice likely to introduce new safety research needs or major changes in emphasis in existing needs."

The Committee further recommended that consideration be given to ". . . research aimed specifically at improving the potential for siting of large water reactors in more populated areas than currently being utilized. For example, studies should be undertaken to develop reactor design concepts providing additional inherent safety or, possibly, new safety features to deal with very low probability accidents involving primary system rupture followed by a functional failure of the emergency core cooling system."

It appears that, because of funding limitations and for other reasons, the recommendations of the ACRS will not be implemented at this time.

Liquid-Metal-Cooled Fast Breeder Reactors (LMFBRs):

The ACRS, in its report on safety research of November 19, 1963, stated that "Recent renewed emphasis on the long range role of large fast breeder reactors points up the need for a well developed, long term, comprehensive research program on the safety of such reactors. A strong research program started now should develop information very useful to the first generation of very large fast reactors". The Regulatory Staff and the ACRS have recently undertaken a preliminary review of a proposed site to be used for construction of a 500 MWe LMFBR. Construction permit reviews of one or more LMFBRs are anticipated in the next few years.

While an extensive LMFBR safety program plan has been formulated, and a growing program in LMFBR safety has been started, many safety-related design decisions will have to be made by applicants and the regulatory groups without the benefit of needed safety research, in part, because of a lag in the implementation of studies of high priority matters.

High Temperature Gas-Cooled Graphite Reactors (HTGRs):

In its safety research report of October 12, 1966, the ACRS recommended that "Becuase the Commission may be called upon to consider proposals to construct reactors utilizing prestressed concrete pressure vessels, the nuclear industry and the AEC should promptly institute a very active safety research program into such vessels, including their design for seismic effects. This program should include research into anomalous failure modes of such vessels,

Representatives of DRDT stated that the current LOFT integral test schedule provides for completing the reactor and facility by the end of calendar 1973 and shows the first experiments involving reasonably high fuel temperatures in 1975. This program will require a very substantial proportion of the limited funds currently available for reactor safety research. The ACRS believes that relevant arguments can be made both for and against continuation of the LOFT integral experiment program with the current budget limitations, and that a straightforward decision regarding implementation of this program is difficult.

Some of the arguments that favor the continuation of LOFT under current budget exigencies are as follows:

1. While the components of the LOFT integral facility are different from those employed in large water power reactors, and while there are large differences in scale that will be difficult to account for, the importance of proper function of emergency core cooling systems to the safety of large water-cooled reactors is sufficiently great that a full test through the entire accident sequence is important enough to warrant the considerable expenditures involved.

2. Confirmation of many out-of-pile, single and multiple effect measurements can be obtained; many effects, including nuclear heating, are simultaneously combined in the integral experiments, and some previously unanticipated events may be observed in this way. Also, the entire emergency core cooling system would be called upon to function in an accident environment.

Arguments against the continuation of LOFT under the current circumstances include the following:

1. Whereas the LOFT integral tests will provide some confirmatory results for the emergency core cooling system, no design-related or operation-related regulatory decisions for large water reactors have been identified that the LOFT integral experiments will resolve.

2. Reactor vendors have stated they will still have to rely primarily and directly on analysis to demonstrate the efficacy of their emergency core cooling system designs in large water reactors, even should the LOFT integral series be performed. The single- or few-effect, out-of-pile experiments concerning phenomena important to a loss-of-coolant accident have been stated to be more directly useful in confirmation of the analytical methods.

The period 1970-71 saw renewed examination and recommendations concerning possible measures to cope with core melt, as is discussed in Section 2.13, "China Syndrome Part 2".

During this period there was discussion with the ACRS of possible recommendations for changes in administration of LWR safety research in order to make it more responsive to regulatory needs. For example, the following memorandum was submitted by an ACRS member for Committee consideration at the 122nd meeting, June 11-13, 1970.

AN ARGUMENT IN FAVOR OF DIVERSE SUPPORT

OF SAFETY RESEARCH

The reactor safety research programs of the Commission have traditionally been administrated by the Division of of Reactor Development and Technology. In recent years, with the great increase in the number of reactors being reviewed for licenses, there has been an increasing need for technical support of the Regulatory Staff. To date, the Regulatory Staff has employed consultants for this purpose, with some help on reactor physics and loss-of-coolant accident calculations from RDT contractors.

The use of consultants is not effective on some of the major Regulatory problems in reactor safety where a substantial and sustained technical effort is needed to define the problem fully and to examine the ramifications of possible solutions.

On the other hand, the use of RDT contractors, supported by the RDT safety research program, suffers from the cumbersome administrative procedures that are required to establish a study effort and to maintain its direction in response to Regulatory Staff needs. This difficulty in the present arrangement arises because the RDT safety research funds, always in short supply, are fully assigned each year. The introduction of a new effort in response to Regulatory needs then means that some portion of the existing program has to be terminated to make funds available for the new effort. More often than not, the best contractor to do the new work is different from the terminated contractor, and all the time and dollar consuming difficulties of contract phase-out must be endured, together with the disrupting effects of the termination on the over-all RDT program.

There is, therefore, a very strong and understandable reluctance on RDT's part to respond rapidly to a Regulatory

on May 27, 1961.

REACTOR SAFETY RESEARCH SUBCOMMITTEE MEETING

O'HARE FIELD, CHICAGO, ILLINOIS

May 27, 1971

The Reactor Safety Research Subcommittee met at O'Hare Field on May 27, 1971, to discuss the AEC water reactor safety research program. Present at this meeting were the following:

ACRS

D. Okrent
J.M. Hendrie
H.G. Mangelsdorf
H.O. Monson
A.A. O'Kelly
M.C. Gaske, ACRS Staff

Regulatory Staff

B. Grimes
S. H. Hanauer
R. B. Minogue
J. A. Norberg
M. Rosen

DRD&T

G. Bright
G.M. Kavanagh
W. H. Layman
J. E. McEwen
J. L. Mershon
A. J. Pressesky
S. A. Szawlewicz
G. F. Brockett, Inc.
W. A. Carbiener, BMI
L. Ybarrondo, Inc.

Meeting with Regulatory Staff Representatives

Mr. Minogue said that some of the areas of safety research relate to reactors to be proposed in the future for difficult sites. Some safety problems can be resolved through use of acceptably conservative design approaches. Dr. O'Kelly inquired regarding the ACRS and Regulatory Staff's influence relative to the priority for reactor safety research by the AEC. Dr. Hanauer stated that the two groups have an influence and are listened to but that the safety research program has a certain amount of momentum and takes some time for changes to be brought about. Dr. Hanauer indicated that the Commission is short of safety research funds and that the situation is going to become worse. He said the Bureau of the Budget sees the large water reactor industry, and they question the need for the AEC to provide funds for water reactor safety research. Dr. Kavanagh stated at a recent Joint Committee Hearing that an additional $30 million/year is needed for reactor safety research. DRD&T has been attempting to obtain industry participation in a joint reactor safety research effort.

Dr. Hanauer thought that some difficult decisions may have to be made regarding water reactor safety research. He felt that people will really have to mean what they say if industry is to be told

addition. Mr. Pressesky stated he believed that this would be a reasonable value, since in his opinion, the number of changes in water reactors will peak and there will only be a small number of changes from year to year. Mr. Pressesky stated he believes that government spending for water reactor safety research should be decreased and industry spending increased. Dr. Okrent inquired whether the total amount being spent will decrease. Mr. Pressesky replied that, if the LOFT and PBF capita costs are neglected, the amount spent should remain about constant.

At the 139th meeting, November 11-13, 1971, the ACRS had considerable discussion on how to get improved ECCS. The excerpt below from the minutes indicates dissatisfaction with the approach taken by the Division of Reactor Development and Technology, as well as divergent points of view among the ACRS members and the Regulatory Staff on how to proceed in this regard.

EXCERPT FROM SUMMARY
139th ACRS MEETING
WASHINGTON, D. C.
NOVEMBER 11-13, 1971

MEETING WITH THE REGULATORY STAFF

2.1 PROGRAM FOR DEVELOPMENT OF IMPROVED ECCS DESIGN

Mr. McEwen (DRD & T) reported on AEC accomplishments on ECCS projects since adoption of the AEC interim ECCS criteria. The following items were noted.

- New FLECHT tests
- GE Blow Down Heat Transfer (BDHT) Program
- 1-1/2 Loop Semi-Scale modification and issuance of a Preliminary System Design Description for the facility.
- RELAP AND THETA-1B improvements
- Multiple injection capability included in LOFT and Semi-Scale
- Preliminary PWR BDHT program initiated at ORNL
- Design modification of the PBF initiated.

RDT bases and plans for future programs and the scope of contemplated ECCS studies were also described.

Mr. McEwen noted that RDT has initiated only those programs which are considered adequately defined; that funding is a problem; that water reactor safety should be the concern of those who are building the plants; that RDT has entered into a joint program with industry, which is funded by AEC at a level of 25%; that none of the programs are directed toward development or testing of advanced ECCS concepts or investigation of metal/water reactions; and that the program includes attention to BWRs through code

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 10, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON WATER REACTOR SAFETY RESEARCH

Dear Dr. Schlesinger:

In several previous reports the Advisory Committee on Reactor Safeguards has emphasized the need for high priority for safety research work aimed at gaining a better understanding of the phenomena important to the course of postulated loss-of-coolant accidents (LOCA). In connection with its review of the Interim Acceptance Criteria, the Committee has stated its belief that more work is required on code development, on improved emergency core cooling systems (ECCS), and on safety research oriented to LOCA-ECCS.

The Committee has recently reviewed the general plans of the AEC and the nuclear industry for water reactor safety research. In this review, the Committee had the benefit of a Subcommittee meeting held on December 7-8, 1971, with representatives of the Division of Reactor Development and Technology, the AEC Regulatory Staff, and the nuclear industry.

In this report, the Committee confines its attention primarily to safety research pertinent to LOCA-ECCS. Continuing progress must be made in improving our knowledge in these areas because of the increased number of reactors soon to be operating and because some of these reactors are to operate at higher power densities.

1. The Committee finds that a desirable increase in ECCS-related research by reactor vendors and the AEC has occurred during the past year. Also, it appears that the electric utility industry may in the future take an active role in funding, directing, and applying such research.

   However, the Committee finds that the relative roles and responsibilities of the utilities, the reactor vendors, and the AEC with regard to safety research have not been clearly defined. Further, the Committee finds that, while a Water Reactor Safety Program Plan

document has been published by the AEC, and there are activities in process or planned for modification and augmentation of the plan, there has not yet been formulated a sufficiently specific definition of the national safety research needs for water reactors, including the means and schedules to be used in resolving problems.

The AEC has frequently stated that the ultimate responsibility for safety rests in the hands of the owner, the utility. However, certain safety questions are generic, applying to a class of light-water reactors, or to all light-water reactors. The ACRS believes that responsibility for the solution of such generic questions should be assumed by the nuclear industry. The Committee recommends that the Regulatory Staff assure itself that the overall industrial program is well delineated, is funded and implemented, and is adequate to provide proper assurance of public safety. The ACRS believes also that the AEC should continue to support at a high funding level a continuing light-water reactor safety research program designed to provide independent confirmation of the adequacy of solutions of identified problems and improved engineered safety features. This program would also provide a valuable and important source of expert consultants to the Regulatory groups.

The ACRS recommends that, in the future, the AEC safety research program should reflect more directly in extent and in detail the recommendations and needs of the Regulatory Staff and the ACRS.

2. While the current programs give attention to many problem areas related to ECCS, including those which seek to substantiate more fully the conservative nature of the design bases, the Committee believes that special emphasis should be given to the following areas:

   a. Flow phenomena during system depressurization and emergency coolant injection. Particular attention must be given to the interaction of the injected coolant with the primary coolant in hot and cold legs, in the downcomer annulus and lower plenum, and in the upper plenum. Model tests of suitable size should define the phenomena involved and should provide improved criteria for design and evaluation.

   b. Reflooding rates as affected by steam binding. Model tests of suitable scale should bound system and component flow resistances important to steam binding. Experiments should also characterize designs which enable high reflooding rates.

c. Flow and heat transfer during blowdown. New AEC-funded programs are underway in this area. System effects should be given careful attention.

d. Improved ECCS computer codes. The mathematical modeling of the reactor system should include both improvement in current capabilities and the development of new and improved integrated computer codes which better represent the actual phenomena involved.

e. Fuel rod failure. Experimental programs should be continued to better establish margins of safety related to time-temperature exposures of cladding to steam environments. Experimental programs with bundles of fuel pins, heated electrically or, preferably, with fission heat, should be started to investigate: (1) the effectiveness with which extreme fuel rod temperature transients can be quenched, including transients in which cladding temperatures approach the melting point; and, (2) the extent of Zircaloy-water reactions during such extreme conditions. Additional understanding should be sought of the possible types and magnitudes of loadings that potentially could be imposed upon the cladding during the course of the LOCA. Further definition of the potential effects of extensive flow blockage resulting from rod damage should also be sought.

3. The Committee has in several reports recommended safety research on interactions between water and molten fuel in experiments which simulate the range of accident situations of interest. This matter has not been pursued in recent out-of-pile studies and, while the Power Burst Facility (PBF) may be used to perform some experiments relevant to this matter, the current PBF program is unclear as to the scope or scheduling of such work. Out-of-pile experimental bases should be established for interpretation of the more difficult in-pile experiments; a capability for theoretical analysis of such events should also be developed.

   Although the ACRS has recommended that research and design studies be undertaken on systems which might be capable of coping with a largely molten core, little such work appears to be underway.

4. The Regulatory Staff has need for independent analyses of the various accident conditions considered in safety reviews, including ECCS performance and accident loads on containments and other vital structures.

Implementation requires a stronger program, which would permit an accelerated development of computer codes for use by the AEC. The ACRS recommends that a substantial increase in funds be made available for regulatory support of these activities and for reactor safety experiments which can be initiated in prompt response to items identified in regulatory reviews.

Sincerely yours,

/s/

C. P. Siess
Chairman

References:

1. Water Reactor Safety Research Program Plan, dated February 1970 (WASH-1146)

2. Augmentation Plan for Water Reactor Safety Research Program, dated November 1971

3. Preliminary System Design Description 1-1/2 Loop Semiscale System, dated September 24, 1971

The ACRS made several important general recommendations in the above letter, including the following:

(1) The AEC safety research program should reflect the needs of the Regulatory Staff and the ACRS.

(2) The AEC should continue LWR research at a high funding level to "provide independent confirmation of the adequacy of solutions of identified problems and (to provide) improved engineered safety features".

(3) There was need for a national program of research in which the role of the reactor vendors and utilities was broad and well-defined.

An independent comment on the AEC's water reactor safety program during the late 1960's and early 1970's was given by Ralph Lapp in testimony to the AEC Rule Making Hearing in Interim Acceptance Criteria, which is reproduced below.

STATEMENT OF RALPH E. LAPP, Jan. 27, 1972
AEC RULE MAKING HEARING ON INTERIM ACCEPTANCE CRITERIA

Nuclear Power Safety and Emergency Core Cooling

My name is Ralph E. Lapp. A Statement of Qualification was filed with this Board on Jan. 7, 1972.

I appreciate this opportunity to appear as a participant in discussing an issue of serious concern to the future growth of nuclear-electric power in this country. My studies in the fossil fuel sector of our energy economy convince me that new power sources are essential to the nation's well-being and that uranium-based power represents the only available option for the United States in the near decades. Thus I appear before this Board as an advocate of nuclear energy. However, I am deeply concerned that the present light water generation of nuclear power reactors may not employ an adequate margin of safety to create public confidence in them.

As of Jan. 1, 1972, a total of 23 nuclear generating units, rated at 10 mill kilowatts of electrical power, are in operation. Industry has accumulated about 100 reactor-years of experience with power reactors. It might be thought that this record, laudable as it is, should instill confidence in the safety of this new power source. However, this experience has been primarily with reactors of modest power; Shippingport (90 Mw), Yankee (175 Mw) and Dresden-1 (200 MW) for which emergency core cooling is less of a challenge than in the 500-800 Mw class, for which about a fifth of the experience applies. Of course, there is no experience with the 1,000 Mw and larger nuclear units. If we reckon reactor experience in 1,000 Mw units, then we have about 18 reactor-years of record, i.e. 18 years of operation of all reactors normalized to a 1,000 Mw level.

The fact that there has been no major thermal emergency (ECC accident or LOCA, loss-of-coolant accident) in the past is of little statistical significance. An accident probability of 0.01 per reactor per year or one chance that a reactor will exhibit an ECC accident in 100 years' operation is not accept- as a public risk. This would mean a 25% chance of accident over the 25 year life of the power unit. Life time operation of four such units would obviously be hazardous in the extreme. Deployment of large numbers of reactors requires that the individual reactor risk must be extremely small since the total risk is summed.

The ECC challenge is greatest in reactors exhibiting the highest power densities and I believe it is instructive to reproduce here the AEC charts displaying the time sequence growth of power plant size.

The dramatic increase in power levels imposed upon reactor design and concomitant rise in power density and this, in turn, posed an emergency core cooling problem which, apparently, presented itself to the AEC's Regulatory Staff with the Consolidated Edison application for a construction permit (Indian Point 2 application of Dec. 1965, awarded Oct. 14, 1966, for an 873 Mew PWR).

It was on October 12, 1966, the Director of Regulation appointed a task force "to conduct a review of power reactor emergency core cooling systems and core protection." A year later, the Ergen task force filed its first report "Emergency Core Cooling" containing 12 conclusions. The Ergen report defines a large number of technical unknowns in the ECC field and makes many recommendations. The report could scarcely be regarded as a confidence builder for the nuclear industry. It is significant ACRS (letter of Feb. 26, 1968) found itself in "substantial agreement" with some conclusions of the Ergen report. It is even more significant that, thereafter, in letters dated March 20 and Nov. 12, 1969, ACRS reemphasized its recommendations on additional safeguards and research on ECCS. This repeated emphasis, including reiteration on June 22, 1971, in testimony before the Joint Committee on Atomic Energy, supports my view that safety research in the ECC sector is lagging behind time goals consonant with the safe deployment of large power reactors.

This safety gap which has opened up between the accomplishments of AEC-industry research and development and the reality of the AEC's Regulatory approval of nuclear-electric stations appears to me as most significant and for this reason I would like to direct my testimony to it.

As a specific example, I call attention to the long-time delays which have been occasioned in the LOFT, loss-of-fluid test facility at the National Reactor Testing Station (NRTS). I wish to emphasize this particular facility since LOFT essentiality to reactor safety is described by the AEC as follows:

> "LOFT is the only integral test in the world planned to carry out a major loss-of-coolant accident experiment which integrates all of the accident initiation, response, and consequence phenomena into a single test with engineered safety systems in operation.
>
> Additionally, it can be noted that
>
> 1) LOFT is the focal point which provides a fundamental sense of direction to water reactor safety investigations.
> 2) as a live reactor in an accident mode, it makes investigators face reality and
> 3) it provides a central vehicle to build and hold a competent technical staff in a vital national program."
>
> (source: Joint Committee on Atomic Energy hearings, FY70, pt 2. p. 957.)

The conceptual design of LOFT was completed in 1963 and Kaiser Engineers was the firm selected as architect engineer for the facility. AEC testimony in 1964 stated: "This experiment is scheduled for test operation in late 1966." (JCAE Authorization FY65, p. 764.) Testimony last year (JCAE Au. FY72 Pt. 2, p. 855) stated that LOFT was 60 percent complete in its construction and that initial operation was scheduled for late 1973.

Thus it appears that LOFT is seven years behind schedule and that high temperature operations will be delayed until 1975. Given the present timetable for deployment of nuclear stations, the LOFT experiments take on the character of a *post facto* safety program.

LOFT is not an isolated example of slippage in time-framing of the reactor safety research program. If we look at the AEC's WASH-1146 "Water Reactor Safety Program Plan" (Feb. 1970) we find a tabulation of 50 individual programs in reactor safety, 15 of which are classified as Class A Priority defined (p. 1-14): "A: This is applied to very urgent, key problem areas, the solution of which would clearly have great impact, either directly or indirectly, on a major critical aspect of reactor safety." (Emphasis as in original document). In addition, 20 of the programs are stipulated as Class B Priority meaning, "This is applied to problem areas which are demonstrably of high interest due to their potential effect on reactor safety." Many of these programs relate to the ECC issue, yet the document shows them extending over a 4 to 5 year time span, beginning in fiscal year 1970.

It is difficult for me to reconcile the fact that much AEC safety research is in the future tense, whereas power reactors are in operation. It seems to me that this situation places the AEC's Regulatory Staff in an awkward position when it is called upon to approve new plant construction and operation. The position will be very much more awkward when utilities come in with reactor designs employing higher power densities. It is pertinent to note that the ACRS (JCAE au Hrg FY71 Pt. 1 p. 133) stated:

"While the resolution of the ECCS issue is believed acceptable for most present reactors at listed powers and power densities, the Committee is not now prepared to advise on the acceptability of ECC systems for higher power density cores. Experimental work is required to provide a bsis for evaluation of operating and accident behavior at higher power densities. The ACRS also notes that more experimental work is required to establish the degree of safety and conservatism in current ECC systems."

I note that the Dec. 28, 1971, statement of Aerojet Nuclear Corporation (pg. 11-22) states:

"The development of analytical models used to analyze and predict the events of loss-of-coolant accidents in water-cooled reactor systems should be complemented with experimental tests which provide data to evaluate and verify the solutions of the analytical models. Without such experimental tests and resultant data, meaningful confidence limits cannot be established for the analytical models."

When, over a year ago, LOFT semiscale tests indicated a deviation of experiment from the predictions of calculational codes, the Regulatory arm of the AEC established a task force to review the data and in June, 1971, Interim Criteria for ECCS were stipulated. In a number of public statements (New Republic, Jan. 23, 1971) I proposed application of limits to reactor power levels and conservative siting policy to reduce the population at risk to the radioactive consequences of a thermal catastrophe. The AEC Regulatory Staff (p. 1-32 of its January 27, 1972 testimony) rejected such proposals in favor of the evaluation model approach. But how good is such a model when it lacks experimental verification?LOFT experiments later in this decade will test the predictive capabilities of analytical models but complete verification may not be attained since LOFT is a 55 Mwt reactor and application of the results involves a scale-up of more than a factor of 60. Furthermore, a limited series of LOFT experiments can test only certain ECCS efficacies. Indeed, a full-scale test with an operating power reactor would provide only a single set of results applicable only to the specific accident mode and core history of the reactor. There will always remain unresolved aspects of verification and for this reason the AEC will have to exercise conservatism in its regulatory role. This is tantamount to saying that reactors can not be made <u>absolutely</u> safe and that there will always be some element of risk for people living in the vicinity of a power reactor.

The siting of nuclear electric stations adjacent to large populations imposes extraordinary responsibilities on the regulatory agency which must license these plants. A nuclear power plant constitutes a unique metropolitan hazard both in nature and in potential magnitude. I can think of only one parallel of comparable risk, namely, siting a large population in a valley directly below a high dam. In such a case, there is direct, line-of-vision perception of the threat and graphic comprehension of the consequences of a dam failure. It is, of course, a low probability event. Given no earth disturbance, such as a seismic shock, people could put confidence in the engineering record of the dam builders but rare events, such as earthquakes, can have high consequences and this is precisely the statistical situation posed by siting nuclear plants near metropolitan populations.

However, the layman is not apt to have line-of-sight perception of the nuclear risk. If he objects to the siting of a nuclear plant in his vicinity, he is at a disadvantage in intervening to oppose the nuclear action. In order to match wits with the nuclear utility, the intervenor needs time, money and availability of competent technical authority. I would estimate that perhaps $500,000 is the sum needed to fund an adequate intervention. Very often an intervenor finds it almost impossible to obtain the services of qualified persons to serve as experts. Too often, the intervenor has felt that he was in contest with not only the utility and the nuclear vendors, but also with the Atomic Energy Commission. If local intervention is to serve as a check on deployment of unsafe reactors or on unsafe siting, then intervenors must have access to some independent authority with which to challenge the organized technical resources of the utility.

As an example of the problem of democratic application of checks and balances in a nuclear issue, I cite the 163 page statement submitted here today by the AEC Regulatory Staff. On Page 2 of the forward it is stated:

> "As the testimony indicates, the technical data available today have been combined with complementary conservative assumptions and procedures in the eval-models. Together, these give reasonable assurance that a design meeting the criteria will provide adequate protection to the health and safety of the public."

But if we were to turn back the clock to the days before the Semiscale tests 845-851, and assume that the Regulatory Staff had then been required to prepare a statement like the 163 page testimony submitted today, would it not have been quite different in its character? Would "reasonable assurance" as now posited have been as conservative as today's? We are not given a definition of what the Regulatory Staff means by "reasonable". In any event the health and safety of a large population is being decided by a small population of experts who put their trust in evaluation models whose predictive capabilities are subject to future verification. It is significant that neither the statement of Aerojet Nuclear Company nor the testimony of Milton Shaw defines the time scale for LOFT. The long delays in bringing the LOFT reactor into operation constitute a serious deficiency in the AEC's reactor safety research program and point to a need for examination of the mechanisms by which the AEC Regulatory Staff coordinates its research needs with the AEC's Division of Reactor Development and Technology. It would be pertinent to know how the Regulatory arm of the AEC has expressed its concern over the long delays in the LOFT and other safety programs.

In his testimony (p.3) Mr. Shaw states:

> "The background and pertinent information pertaining to our safety R&D program and its relationship to other ongoing R&D programs and to the U.S. civilian power program have been presented in many official AEC publications and covered in depth in annual testimony before the Congress(7,10,11)."

I submit that the AEC's safety program is deficient in publication of up-to-date and critical evaluations. I admit that this deficiency is being corrected and the situation is improving. But I have taken the time to recheck the literature references cited by Mr. Shaw and I find that the AEC's safety program has been inadequately dealt with in Congressional hearings. In some years the Joint Committee on Atomic Energy glossed over the issue with no critical examination of the program. The most recent literature cited by Mr. Shaw (AEC Licensing Proceedûre and Related Legislation, Hearings before the Joint Committee on Atomic Energy, 4 parts,209 pages.) is almost exclusively devoted to regulatory matters with only little attention given to reactor safety. One would have thought that the Idaho Semiscale Experiments would have been treated in detail by the Joint Committee. Instead, the issue was dealt with by calling Mr. Shaw from the audience during a hearing to testify briefly on the problem. The Joint Committee concerned itself with studying means of expediting the licensing procedures. There exists an urgent need for the Joint Committee on Atomic Energy to hold public hearings on the vital issue of reactor safety.

I do not wish to appear to be unduly critical of the Atomic Energy Commission, but the nature and magnitude of potential nuclear risks demand a public accountability which imposes unusual responsibilities upon the AEC. Our democracy must invent mechanisms for dealing with technological risk so that checks and balances are applied to the decision making of the AEC. The emergency core cooling issue involves such complex technology, as illustrated by the technical details of the 163 page AEC Regulatory Staff tesimony, that I doubt if we have arrayed here today adequate talent to challenge the testimony. I note that a total of ten Regulatory Staff personnel sponsor the testimony. We have almost a full score of AEC or AEC contractor personnel of high technical qualification to back up the AEC's presentations before this Board.

Summarizing my position, I believe that the Atomic Energy Commission has allowed reactor safety research to lag so that its Regulatory Staff is called upon to judge reactor applications without an adequate experimental base which verifies the evaluation models and checks out the calculational codes of the safety statements submitted by the utilities. Furthermore, it is my opinion that new mechanisms are required to provide independent checks and balances for the protection of the public health and safety in areas where high power reactors are sited.

It appears to me that part of the explanation for the faltering AEC safety program in ECCS may be ascribed to the undefined role of the nuclear industry in this area. There was apparently a belief within the Atomic Energy Commission that it had fulfilled its promotional aspects of reactor development during the late 1960's and that it was up to industry to assume responsibility for the reactors which were being marketed. In this connection, the Advisory Committee on Reactor Safeguards (JCAE Au. Hrg. FY71 Pt 1 p. 115) commented.

> "We are unable to determine what factors determine industry vs. AEC funding of reactor safety research programs, other than in those cases where the AEC discontinues support. Then the decision is clearly up to industry."

It is my own impression that interest within the AEC shifted from safety research on water reactors to programs oriented toward the power-breeder and that this also accounts, in part, for deficiencies in the present water reactor safety program.

The proprietary nature of certain reactor safety information developed by nuclear vendors may be tantamount to a classification of data which denies intervenors access to information vital to their efforts. I believe that this point will be amplified by attorneys who are participating in this hearing.

In conclusion, I wish to summarize some suggestions and recommendations which may be constructive in increasing public confidence in nuclear power safety:

1. Require the Atomic Energy Commission to submit an annual report on progress in nuclear reactor safety programs. I would suggest that this report include the separate comments of the Regulatory Staff and of the Advisory Committee on Reactor Safeguards.
2. Amend the Atomic Energy Act to require biennial public hearings of the Joint Committee on Atomic Energy for the purpose of investigating the current status and adequacy of the AEC-Nuclear industry safety programs.
3. Direct the Atomic Energy Commission to issue specific criteria for the siting of power reactors, defining the allowable population of risk as a function of distance from the reactor site. (The absence of specific criteria has allowed escalation of the population at risk to a point

where the Newbold Island facility would, if approved, "see" 0.75 million people within a radius of 10 miles.)
4. Encourage the nuclear industry to redesign reactor cores to effect a reduction in power densities so as to ease the burden on the ECC system in the event of a coolant accident.
5. Require power derating of the 1,000 Mwe class reactors of the pressurized water type which are sited so as to have more than 10,000 persons at risk within a radius of 10 miles from the reactor.
6. Direct the Atomic Energy Commission to initiate a program to develop core restraint systems (i.e. "core-catchers") as part of a defense in-depth safety system to insure the public safety and protect the environment. (I would add that such a safeguard becomes essential for offshore reactors since a melthrough could result in extensive marine contamination).
7. Consider type certification of power reactors, treating the reactor core and primary coolant system with ECC systems as a unit, so as to facilitate licensing of nuclear reactors.

A very considerable controversy swirled around the management of LWR safety research under Milton Shaw during the early 1970's. Some of this was aired in a series of articles by Robert Gillette in Science (Gillette, 1971-75).

Among other things, Gillette reported allegations that:

1) Between 1968 and 1971, the Division of Reactor Development and Technology bootlegged money from water reactor safety to accelerate the breeder, and in the process killed or cut back a number of key research projects that had begun to raise questions about nuclear plants coming up for licensing.

2) Shaw had shown considerable indifference toward urgent needs of the regulatory branch for technical help during this period, and for several years forbade direct contact between safety researchers and AEC's regulatory staff.

Gillette discussed major delays in the LWR safety research program and their controversial causes, as well as the trend to reduce the AEC program on LWR safety research. Gillette reported that the AEC under Chairman

Schlesinger had considered and rejected removing the water reactor safety program from the development side of the AEC and placing it under the wing of the regulatory and licensing authorities.

In 1973, Dixie Lee Ray, the new Chairman of the AEC, reorganized the safety research program, taking the LWR safety program away from Shaw and placing it in the hands of a new Division Director, Herbert Kouts, a former ACRS member. There was also a major expansion in the funding level for LWR safety. The ACRS letter of November 20, 1974, to Chairman Ray reflected satisfaction with this change, which led to an LWR safety research program which was much more responsive to the requests of the Regulatory Staff.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 20, 1974

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON WATER REACTOR SAFETY RESEARCH

Dear Dr. Ray:

At its 175th meeting on November 14-16, 1974, the Advisory Committee on Reactor Safeguards completed a report on certain aspects of the water reactor safety research programs and needs. In its review the Committee had the benefit of a Reactor Safety Research Subcommittee Meeting on July 23-24, 1974, of the Emergency Core Cooling System (ECCS) Subcommittee on September 28, 1974, and of Subcommittee meetings on generic matters and applications, and of an Information Meeting held by the Reactor Safety Research Division (RSR) on September 19-20, 1974. The ACRS has most recently reported on facets of water reactor safety research in its letters of February 10, 1972, and March 20, 1969, in its letters on ECCS Acceptance Criteria of January 7, 1972, and September 10, 1973, and in its testimony to the Joint Committee on Atomic Energy of September 27, 1973, June 22, 1971, and April 5, 1967. In this report, the Committee will go into some detail on matters pertaining to the loss of coolant accident (LOCA) and take up some other subjects more generally.

LOCA-ECCS

In its previous reports the ACRS has consistently emphasized safety research oriented to LOCA-ECCS and has strongly encouraged the understanding and development of improved emergency core cooling systems. In the period following our Report on Water Reactor Safety Research of February 10, 1972, the AEC reactor safety research program has been reorganized, redirected and augmented, and a substantial industry-sponsored program through the Electric Power Research Institute (EPRI) has been initiated. Vendor sponsored research and development programs have been continued. Problem areas and needs in safety research have been more effectively defined by the Regulatory Staff and RSR. Programs being undertaken are responsive to such needs, though some, because of their complexity and expanded requirements, now appear to have significantly extended schedules for completion. Other programs, such as testing a larger scale reactor coolant pump, have been discussed, but not yet funded. The programs for LOFT, Semiscale and Power Burst Facility (PBF) have encountered significant delays, and the plenum fill experiment will require a substantial increase in funding to achieve the current objectives.

A balance between research seeking basic understanding of the phenomena involved and research directed toward empirical correlations suitable for application in nuclear reactor power plants is being sought by RSR. It appears to the Committee that in the future greater emphasis on gaining a basic understanding would be appropriate. The Committee finds the programs of RSR to be well conceived and is impressed with the influx of technical experts into the management and programmatic work. The varied experimental programs will require periodic reviews so as to ensure that they are structured to obtain compatible and mutally supportive goals. The Committee supports the RSR requests for its expanding programs.

As noted in its February 10, 1972 report, the Committee finds that the relative roles and responsibilities of the utilities, the reactor vendors, and the AEC with regard to safety research have not yet been sufficiently clearly defined. More specifically, it is not clear that industry, as an entity, has developed a comprehensive program of scope and schedule commensurate with the need. The role of architect-engineers in safety research warrants examination. Clarification of the roles and responsibilities should be pursued to insure that needed programs will have the appropriate attention and funding priorities.

RSR has through its review groups and through its coordinators, a good representation of the expertise from the AEC, industry, national laboratories, and the universities. Since the ACRS has not seen reports of the review groups and coordinators, it is at this time unable to comment on the effectiveness of these bodies in shaping meaningful programs. For some of the programs, more attention should be given to assuring that a proper mix of analytical and experimental efforts will be brought to bear on specific problems, that the working groups will not be isolated from one another, and that the influence of personnel experienced in reactor systems will be included. The Committee emphasizes the importance of strong analytical support of complicated experiments.

The Committee also notes the progress being made by the AEC in furthering international exchange and participation programs. These are significant steps in advancing reactor safety and are to be encouraged.

The AEC safety programs include studies of alternate and advanced ECCS concepts. The flexibility of the LOFT and semiscale facilities provides some opportunities for experimental investigations. The Committee believes that much more should be done on improved ECCS concepts, including conceptual design work and analytical studies using improved codes, and the input of the experimental studies to investigate how ECCS performance might be optimized. The studies should also include assessments on the overall reliability of the ECCS and what additional measures, if any, need to be taken.

The RSR advanced code development program has been extended from one to three contractors. The Committee concurs in obtaining a broader-based participation for technical input into the AEC's advanced code, Reactor System Transient (RST). The RST code is to have the capability of utilizing inputs based upon realistic estimates, as well as conservative estimates, so that safety margins can be more quantitatively determined. Special effort is needed to develop analytical predictive methods for realistic evaluations. Major benefits from a successful RST code would be in resolving questions relating to the significance of modeling techniques and parameters, and in allowing the use of scaled experiments in place of full-scale tests.

The Committee emphasizes the need to assure the adequacy of the current LOCA-ECCS research plans in the following areas:

a. system effects in blowdown of BWR's
b. PWR pump overspeed for a postulated downstream break
c. heat transfer and flow during reflooding
d. processes involving counter flow of different phases

Also, the effects that steam generator tubes with degraded properties have on the course of events in a LOCA-ECCS require further attention.

Additional Topics

The Committee has on several occasions in the past recommended a vigorous research program to investigate various facets of fuel element behavior, including power-flow mismatches, the potential for fuel-failure propagation, and the better establishment of acceptable limits. The Power Burst Facility has long been intended as a major in-reactor tool for such work and its proper use should receive priority. A necessary program of development of tools for theoretical analysis of fuel element behavior in transients and accidents has been initiated, but much additional progress is required to obtain the needed prediction capability. Out-of-reactor tests such as fuel-coolant interaction experiments should receive careful consideration as possible means of facilitating understanding of PBF experiments. Also, additional experimental and theoretical knowledge of the microscopic behavior and effects of fission gases in transients should be sought so that the understanding of the behavior of irradiated fuel in transients can be placed on a sounder footing.

The Committee supports the RSR program of continuing probabilistic accident studies. Consideration should be given to those risk assessments which may need further experimental and analytical study to more firmly establish a better understanding and to reduce uncertainties. The Committee reiterates its previous recommendations for research into phenomena involved in core meltdown, including the mechanisms, rate and magnitude of radioactive releases and the study of means of retaining molten cores or ameliorating the consequences. In this connection, more knowledge of the possibility and extent of steam explosions in the presence of large quantities of molten fuel and steel is of particular importance.

The Committee recognizes the value of the RSR programs in "Metallurgy and Materials," "Fuel Behavior," and "Environmental and Siting." The Committee acknowledges the contributions from the AEC Heavy Section Steel Technology Program in Fracture Mechanics, Fatigue, Irradiation Effects and Inspection and Monitoring. In this regard, the Committee notes its recommendations and interest in improved means for nondestructive testing, in increased understanding of pneumatic failure, in the adequacy of experimental information in all categories relevant to the safety of new pressure vessel steels, and in the study of concepts of withstanding or ameliorating the effects of pressure vessel rupture.

With regard to "Environmental and Siting," the Committee particularly wishes to emphasize the need for improved understanding of earthquake causes, probabilities, magnitudes, and effects.

The Committee recognizes there are three principal sources of funding for reactor safety research: The AEC, Electric Power Research Institute, and the reactor vendors. Within the framework of total available funds, it is essential that programs be optimized. Further efforts are desirable to assure that priorities on the use of available R&D resources are appropriately assigned and to establish whether additional funding is needed.

A final comment relates to the availability of safety-related information. AEC and EPRI programs are in the public domain. The ACRS recommends a policy of increased disclosure of safety-related information and suggests that reactor vendors reevaluate their proprietary categories in an attempt to place as much information as possible in the public domain.

Sincerely yours,

W. R. Stratton
Chairman

7. SOME FINAL COMMENTS

7.1 ON WASH-1400: SOME EFFECTS AND IMPLICATIONS

The Reactor Safety Study, WASH-1400, may have had part of its genesis in a letter from Senator Pastore to AEC Chairman Schlesinger dated October 7, 1971, which is reproduced on the following page.

It seems that one formal report produced by the AEC in response to this letter was a report entitled "The Safety of Nuclear Power Reactors (Light Water-Coled) and Related Facilities," WASH-1250 (FINAL DRAFT) dated July, 1973, an earlier draft version of which was circulated for comment in late 1972. However, this report did not provide a quantitative assessment of risk in a probabilistic fashion as discussed in the enclosure to Senator Pastore's letter.

In the summer of 1972, the AEC initiated a major study on LWR risk assessment with Professor Norman C. Rasmussen of MIT serving (half-time) as the study director. Mr. Saul Levine, a former member of the Regulatory Staff, served as full-time staff director. A separate group was established within the AEC which performed the study with the aid of many contractors and consultants.

The first draft report (Draft, WASH-1400, Reactor Safety Study) was issued by the AEC for comment in August 1974; the final report, WASH-1400 (NUREG-75/014), was issued by the Nuclear Regulatory Commission in October, 1975 and was comprised of a main report and eleven appendices.

When draft WASH-1400, Reactor Safety Study, was issued by the AEC for comment in August of 1974, it drew a very substantial response from many quarters of industry, government and the public. The principal architects of the study, Messrs. Rasmussen and Levine, sometimes summarized the results by noting that, while the probability of core melt was estimated to be higher than many people had expected,* the magnitude of the

*For example, E.G. Case of the AEC Directorate of Licensing, in testimony at the Public Hearing for Operating Licenses for Prairie Island Units 1 and 2, and in an article in Nuclear Safety, Vol. 15, No. 3, May - June, 1974, presented the Staff conclusion that the "likelihood of a sudden major LOCA accompanied by failure of the ECCS to cool the core to the degrees necessary to prevent breach of the containment is so extremely small - i.e., less than one chance in ten million per reactor year - that the environmental risk of such an accident can be considered to be negligible".

This probability appears to be the product of an estimate of 10^{-3} to 10^{-5} per reactor year of a sudden major LOCA, and 10^{-3} to 10^{-4} of the likelihood of ECCS failure severe enough to lead to containment breach.

Core melt associated with LOCA was estimated to have a much higher probability per reactor year in WASH-1400.

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October 7, 1971

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington,D.C. 20545

Dear Dr. Schlesinger:

I understand that when you met with Congressmen Price, Holifield, Hosmer, Hansen, and Senator Bennett, at the Geneva Conference in September, one of the subjects you discussed was nuclear power plant safety. I understand the members suggested that a comprehensive assessment of the safety aspects of nuclear reactors be made with the intent of setting down for the industry and public a clear-cut summary of what the facts are in this matter.

This idea has always appealed to me. It is my view that a basic document of a type similar to the 1962 Report to the President on Civilian Nuclear Power could be developed which would be just as valuable in providing the industry and public a basic understanding of our status and objectives in this critical field as the 1962 Report was to the formulation of the civilian power program. Accordingly, I join my colleagues in their suggestion and, as they promised, I hereby forward to you more specific suggestions prepared by the Committee staff.

If you think the Committee can be of assistance in this matter, please let me know.

Sincerely yours,

John O. Pastore
Chairman

Enclosure:
"Status of Nuclear
Power Plant Safety"

STATUS OF NUCLEAR POWER PLANT SAFETY AND ASSOCIATED SAFETY RESEARCH

There appears to be growing concern in the public arena, and certainly in AEC public hearings related to reactor licensing, regarding the safety of nuclear power plants and their associated safety research programs. The recent activities in regard to nuclear power plant emergency core cooling systems are an example of this type of development.

It is important for the AEC to be able to document in as clear terms as is possible the levels of safety it deems necessary in nuclear power plants as well as the levels of safety that are actually being provided in current designs. One way of accomplishing this objective would be to prepare a report which, by addressing the probability of occurrence and consequences of the spectrum of accidents which could befall a nuclear power plant, would represent an assessment of the risks involved in the use of nuclear plants. Of course, it would also be necessary to compare these risks with those involved in other fields of endeavor in our society in order to put them into proper perspective.

Such a report could also address the safety margins that are designed and built into plants, both in terms of engineering margins and in terms of specific safety features provided. For instance, the report could discuss in quantitative terms the probability of occurrence of a loss of coolant accident, the probability of the emergency core cooling system fulfilling its intended function and the consequences of the loss of coolant accident with and without emergency core cooling functioning properly. As another example, it could consider, under a number of appropriate conditions, the probability and related consequences of the failure of both normal and emergency electric power supplies.

Of course, any safety assessment such as this should address the adequacy of the technical substantiation being provided by the industrial and AEC water reactor safety research programs. This will require an examination of the adequacy of the progress being made as well as the reasons for lack of progress should it be determined to be inadequate. It should also include an examination of the relationship between the industrial and AEC water reactor safety research programs and of the relationship between divisions in the AEC that are involved in the establishment and approval of, the direction of, and the use of the results of these programs.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

OCT 18 1971

Honorable John O. Pastore
Chairman, Joint Committee
on Atomic Energy
Congress of the United States

Dear Senator Pastore:

Thank you for your letter of October 7, 1971 on nuclear power plant safety. As requested, we have taken steps to initiate a comprehensive assessment of the safety aspects of civilian nuclear reactors and a summary report will be prepared based on the outcome of this assessment. In this regard, we appreciate the suggestions by the Committee Staff enclosed with your letter and your offer of assistance on this matter.

Since an assessment of the scope suggested by your letter will undoubtedly require a considerable effort, we would plan, as we have done in the past, to summarize the status and plans for conducting it during the FY 1973 Hearings and to keep you informed thereafter of significant progress.

Sincerely,

James R Schlesinger

Chairman

potential consequences of core melt has a wide range of values, and the probability is high that the consequences will be modest compared to other risks, and small in absolute value.

Appendix XI of the final version WASH-1400 is devoted to a discussion of what the authors of the report considered to be the principal comments on the draft report, and their response to these comments.

In April, 1975, the ACRS sent a relatively concise letter report on WASH-1400 to NRC Chairman Anders, in which it noted it was giving primary attention to the implications of the report on the reactor licensing process. In July, 1976 and in December, 1976 the ACRS sent letters to Congressman Udall in response ot his request for comment on eleven issues. The letters to Anders and Udall are reproduced on the following pages.

One opinion given by the ACRS, which was common among many of those reviewing WASH-1400, was that the uncertainty in the results was larger than that assigned in the report.

The report of the American Physical Society Study Group on Light Water Reactor Safety (APS, 1975) and the comments by the Environmental Protection Agency are among the most interesting on draft WASH-1400. The record of the Oversight Hearing before the Subcommittee on Energy and Environment, Committee on Interior and Insular Affairs, House of Representatives, June 11, 1976 provides a good crosssection of comment on the final report.

The following paragraphs provide some of the effects, implications and issues arising from WASH-1400.

1. To the extent that WASH-1400 is correct, it appears that the large emphais placed upon improved primary system integrity and on improved ECCS from 1966 on, was effective, both in that the large LOCA was found not to be the dominate source of risk and that the oveall risk was low. In fact, the results of the study suggested that, if more effort was needed on reactor safety, reactor transients requiring reliable shutdown (residual) heat removal for extended periods of time were a strong candidate for attention. And the equipment needed for residual heat removal might be the focus of such attention.

Another candidate from the results of WASH-1400 for emphasis in any effort to reduce the probability of core melt lay in the removal of unanticipated adverse interactions between separate systems, a topic which the ACRS had earlier flagged as requiring attention. Also, as noted in the chapter on seismic safety, earthquakes appear to be a potentially significant contributor to the total risk, contrary to the results of WASH-1400. And as had been argued by Levine many years earlier in discussions of General Design Criterion No. 17, an extended loss of all AC power appeared to be an important contributor to overall risk.

2. WASH-1400 illustrated clearly that there are many different individual accident paths having the potential to cause core melt, only some of

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 16, 1976

The Honorable Morris K. Udall, Chairman
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, DC 20515

Dear Congressman Udall:

At its 200th meeting, December 9-11, 1976, the Advisory Committee on Reactor Safeguards (ACRS) continued its consideration of the points raised in your June 14, 1976, letter on the Reactor Safety Study (RSS, WASH-1400, NUREG 75/014). The ACRS had previously considered these matters at its 196th and 199th meetings and had responded to issues 1, 3, 4, 6, 8, 9 and 10 in its letter to you dated July 14, 1976. In its further consideration of the remaining four issues, the Committee had the benefit of meetings of its Reactor Safety Study Working Group with the Nuclear Regulatory Commission Staff in Washington, DC, on October 12, 1976, and November 10, 1976.

The ACRS is continuing to evaluate the considerable body of information presented in the RSS report, its appendices, and the comments received on it, giving primary attention to the potential implications of the report for the reactor licensing process. This letter provides the Committee on Interior and Insular Affairs a brief resume of current ACRS thought on issues 2, 5, 7 and 11.

"2. Adequacy and appropriateness of analysis used in NUREG 75/014 for purposes of estimating the likelihood of low probability, high consequence events."

The ACRS believes that the methodology of NUREG 75/014 is useful for purposes of identifying important accident sequences and for attempting to develop comparative and quantitative risk assessments for low probability, high-consequence accidents. However, the ACRS believes that considerable effort by more than a single group over an extended period of time will be required to evaluate the validity of the results in NUREG 75/014 in absolute terms. Among the matters which will warrant emphasis in such an evaluation are the following: improved quantification of accident initiators; the identification and evaluation of atypical reactors; the influence of design errors; improved quantification of the role of operator errors; improved quantification of consequence modeling; and the development of improved data for systems, components and instruments under normal and accident-related environmental conditions in a nuclear reactor.

4. "Sensitivity of NUREG 75/014 conclusions to differences in reactor design, in site characteristics, in local meteorological conditions and in population distributions."

All of the factors noted above will have some effect on the probability or consequences of a serious accident. The Committee has recommended that the methodology of the Study be applied to other types and designs of reactors, other site conditions and other accident initiators and sequences. If this is done, it will provide greater insight into the sensitivity of differing reactor designs and safety features.

6. "Adequacy of NUREG 75/014 methodology to take account of gradual degradation of plant safety over plant lifetime."

The Committee believes the methodology is capable of taking into account wear out of components and degradation of equipment over the lifetime of the plant but an appropriate data base needs to be developed.

8. "Need for periodic updating of NUREG 75/014 to take account of new data."

The Committee believes that a continuing effort is desirable in the application of the methodology developed by the Reactor Safety Study not only to factor in new data but also to consider design variations and new concepts.

9. "Need for continuing analysis of NUREG 75/014 for purposes of delineating areas of research and data collection."

The Committee believes that the NUREG 75/014 methodology should be used to aid in delineating areas for further research. Special emphasis should be given to quantification of the initiators, probabilities, and consequences of core melting.

10. "The extent to which NUREG 75/014 can be used to aid development of regulatory policies concerning design, construction, and operations."

The Committee has recommended to the NRC that many of the techniques used in the Study can and should be used by the reactor designers to improve safety and by the NRC Staff as a supplement to their safety assessment.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Attachment:
Ltr. to Hon. W. Anders from D. W.
Moeller, dtd 4/8/75 re: WASH-1400

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 8, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: REACTOR SAFETY STUDY, WASH-1400

Since the release of the draft Reactor Safety Study, WASH-1400 (RSS) in August 1974, the Advisory Committee on Reactor Safeguards has been reviewing the considerable body of information presented in the report, its appendices, and the comments received on it, giving primary attention to the potential implications of the draft report on the reactor licensing process. In its review, the Committee has had the benefit of Subcommittee meetings held on October 9, November 22, and December 20, 1974, and March 5, 1975, and of full Committee meetings held on October 10-12, October 31-November 2, November 14-16, December 5-7, 1974, and January 9-11, February 6-8, March 6-8, April 3-5, 1975.

The ACRS believes that the RSS represents a valuable contribution to the understanding of light water reactor safety in its categorization of hypothetical accidents, identification of potential weak links for the two reactors studied, and its efforts to develop comparative and quantitative risk assessments for accident sequences examined. The Committee believes that a continuing effort and better data will be required to evaluate the validity of the quantitative results in absolute terms. Special emphasis should be given to quantification of the initiators, probabilities, and consequences of core melting.

The Committee believes that the methodology of the RSS should be applied to other types and designs of reactors, other site conditions and other accident initiators and sequences, and that the current efforts to compile, categorize, and evaluate nuclear experience should be extended in breadth and depth to improve the data base for future studies of this type.

The Committee believes, further, that the RSS can serve as a model for similar studies of the failure probabilities, consequences, and resulting risks of other hazards (both nuclear and non-nuclear) to the health and safety of the public.

The Committee believes that many of the techniques used in the RSS can and should be used by reactor designers to improve safety and by the NRC Staff as a supplement to safety assessment.

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The Committee's review of the RSS has not caused the Committee to alter its judgement that reactors now under construction or in operation do not represent undue risks to the health and safety of the public.

The Committee will continue to review the RSS and will comment further on it in the future.

Sincerely,

Original Signed by:
W. Kerr

William Kerr

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 16, 1976

The Honorable Morris K. Udall, Chairman
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, DC 20515

Dear Congressman Udall:

At its 200th meeting, December 9-11, 1976, the Advisory Committee on Reactor Safeguards (ACRS) continued its consideration of the points raised in your June 14, 1976, letter on the Reactor Safety Study (RSS, WASH-1400, NUREG 75/014). The ACRS had previously considered these matters at its 196th and 199th meetings and had responded to issues 1, 3, 4, 6, 8, 9 and 10 in its letter to you dated July 14, 1976. In its further consideration of the remaining four issues, the Committee had the benefit of meetings of its Reactor Safety Study Working Group with the Nuclear Regulatory Commission Staff in Washington, DC, on October 12, 1976, and November 10, 1976.

The ACRS is continuing to evaluate the considerable body of information presented in the RSS report, its appendices, and the comments received on it, giving primary attention to the potential implications of the report for the reactor licensing process. This letter provides the Committee on Interior and Insular Affairs a brief resume of current ACRS thought on issues 2, 5, 7 and 11.

"2. Adequacy and appropriateness of analysis used in NUREG 75/014 for purposes of estimating the likelihood of low probability, high consequence events."

The ACRS believes that the methodology of NUREG 75/014 is useful for purposes of identifying important accident sequences and for attempting to develop comparative and quantitative risk assessments for low probability, high-consequence accidents. However, the ACRS believes that considerable effort by more than a single group over an extended period of time will be required to evaluate the validity of the results in NUREG 75/014 in absolute terms. Among the matters which will warrant emphasis in such an evaluation are the following: improved quantification of accident initiators; the identification and evaluation of atypical reactors; the influence of design errors; improved quantification of the role of operator errors; improved quantification of consequence modeling; and the development of improved data for systems, components and instruments under normal and accident-related environmental conditions in a nuclear reactor.

The ACRS believes that NUREG 75/014 represents a very considerable contribution to the understanding of reactor safety and provides a point of departure for quantitative assessment.

"5. Adequacy of NUREG 75/014 methodology to take account of multiple, correlated errors in procedures, design, judgment, and construction such as those leading to the Browns Ferry fire."

The ACRS believes that the methodology of NUREG 75/014 is useful in accounting for that portion of the risk resulting from identifiable potential common mode or dependent failures, and can be used to search out the possibility of multiple correlated errors. However, the methodology cannot guarantee that all major contributors to risk will be identified, and a considerable element of subjective judgment is involved in assigning many of the quantitative input parameters. Both for nuclear and non-nuclear applications, for complex systems, where multiple, correlated failures or common cause failures may be significant, the record shows that investigators working independently will frequently make estimates of system unreliability which differ from one another by a large factor. At this stage of its review, the ACRS believes that a substantial effort may be required to develop and apply dependable methods for quantitatively accounting for the very large number of multiple correlated or dependent failure paths and to obtain the necessary failure rate data bases.

Whether multiple, correlated errors will dominate the overall risk, however, is subject to question, particularly if simpler postulated accident sequences are generally the dominant contributors to the likelihood of system failure.

"7. Extent to which the final version of NUREG 75/014 takes into account comments on the draft version."

The ACRS is in the process of reviewing the disposition of selected comments received by the Reactor Safety Study Group, particularly as they have implications for short or long-term improvements in reactor safety. The ACRS plans to continue this type of activity; however, it is beyond the scope or available working time of the ACRS to review in detail the extent to which the final version of NUREG 75/014 takes into account the comments received.

"11. Validity of NUREG 75/014 conclusions regarding accident consequences."

As stated in its report to you of July 14, 1976 and as indicated in its response to other questions in this group, the ACRS believes that considerably more effort on the part of various contributors is needed to evaluate the quantitative validity of NUREG 75/014 conclusions regarding accident consequences. Based on information currently available, the ACRS would assign a greater uncertainty to the results than that given in NUREG 75/014.

The ACRS believes that the past and current practice of trying both to make accidents very improbable and to provide means to cope with or ameliorate the effects of accidents has been the correct approach to nuclear reactor safety.

The ACRS review of the Reactor Safety Study has not caused the ACRS to alter its judgment that operation of reactors now under construction or in operation does not represent an undue risk to the health and safety of the public. The ACRS believes that NUREG 75/014 has suggested many fruitful areas for study and evaluation for potential improvements in light water power reactor safety. The ACRS also believes that the extension of such risk assessment methodology to the total spectrum of activities involved in the production of nuclear power and in the production of electric power by other means, as well as to other technological aspects of society, could add significantly to our overall understanding of risk.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

which are peculiar to the design of a specific reactor. Hence, it becomes difficult to reduce the risk by a large factor, say 100, by trying to reduce the probability of those few events most likely to cause core melt for a particular reactor. If the few leading accident initiators are made far less probable by design changes, a large number of other initiators become candidates for importance. Also, because of the uncertainties arising from failures due to common or related causes, it is frequently difficult to accomplish major reductions in accident probability with high confidence, once the probability is already low.

3. WASH-1400 demonstrates fairly unequivocally that accident risks from reactors arise from Class 9 accidents, even though the AEC-NRC environmental assessment of radiological risks includes only accidents up to Class 8 on the basis that the probability of Class 9 accidents is sufficiently low to render their effect negligible and even though Class 9 events are, with rare exception, not treated in Safety Analysis Reports or Staff Safety Evaluations.

4. WASH-1400 vividly illustrates the inconsistencies in the methodology of the 10 CFR Part 100 Site Criteria, and suggests the possible use of probabilistic risk criteria. However, the potentially large uncertainties inherent in most estimates of low probability hazards makes it difficult to devise a workable and defensible quantitative probabilistic risk acceptance criterion which, in practice, would not require large elements of judgment. Some examples of the kinds of questions which arise are as follows:

If a best estimate value is to be used in meeting a risk acceptance criterion, should there not, nevertheless, be some limit on the uncertainty?

Should a probabilistic risk acceptance criterion include risk aversion? If yes, how much and why?

For low probability events, what confidence level is likely to be meaningful at successively lower return frequencies?

When experts differ markedly, as they do on the probability of large seismic events, how shuld such a situation be factored into decision making with a probabilistic criterion?

5. The containment failure made studies of WASH-1400, coupled with later studies, show that different containment designs (e.g., the large dry PWR, the ice condenser and the Mark I BWR pressure suppression types) may seem to lead to similar off-site doses for the "standard" design basis accidents, yet be quite different with regard to their protective capability against the family of core melt scenarios. WASH-1400 has led to a growing effort to study containment modifications which have the potential to provide a reduction in risk from core melt (Gossett, 1977). A dozen years after the "China Syndrome" came into focus, serious attention is being given to the possibility of mitigation of the consequences of core melt, in considerable part due to the action of the U. S. Congress (NUREG-0438).

6. The Department of Interior has raised questions concerning the potential effect on water resources of the solidified residue of the molten core which has penetrated the containment foundation in the unlikely event of core melt. This question has not been posed as part of the licensing process for land-based reactors* but has become part of the regulatory review of the Floating Nuclear Plant, on the recommendation of the ACRS (ACRS, 1972). A limited study of the question was performed for WASH-1400, and more extensive studies are underway. It remains to be seen whether new criteria for site criteria with regard to hydrological considerations will result.

7. WASH-1400 suggests a low average risk for U.S. sites. However, it did not provide specific evaluation of the risk for the most populated sites and of the very remote sites. Such a comparison may provide additional input into the judgment of site acceptability for future reactors.

8. WASH-1400 represents the most detailed risk estimate for any technology. The report exploited most existing techniques and data, and pioneered some new approaches, but it did not include sabotage,** various effects such as design and construction errors, or several forms of system degradation. There are also questions concerning its treatment of health effects and its estimate of risk from certain accident initiators such as earthquakes.

Neverhteless, WASH-1400 tends to indicate that nuclear power is safer than most, if not all, sources of central station electric power, as well as many other existing technologies. And, the uncertainties which apply to the risk estimation in WASH-1400 can largely be expected to apply directly or to appear in similar form for other technologies.

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* The effect of routine releases of radioactive liquid waste products or of small accidental spills on drinking water supplies or other aquatic resources has been reviewed routinely. However, only on rare occasion prior to 1974 was the potential effect of a core melt on a large water resource discussed even informally as part of a power reactor licensing review. One such occasion was in connection with the proposed siting of an LWR on the edge of a relatively small, poorly flushed, important lake; for other reasons this review was never completed and the issue did not receive great emphasis.

Presumably, the prevailing thinking was that to the extent that a risk existed from core melt, it was the airborne radioactivity, and not the liquid pathway, which was dominant.

** A separate section on sabotage considerations has not been included in this study for various reasons. In particular, it is inappropriate to discuss in such a document any specific avenues that a prospective saboteur might take. Although sabotage was called out as a potentially important safety consideration as early as 1950 in WASH-3, it began to receive continuing attention and emphasis in the regulatory process in the early 1970's, and has been a rapidly developing aspect of reactor safety. However, it remains a matter which is very difficult to quantify for purposes of overall risk evaluation.

9. It seems inevitable that Class 9 accidents will be introduced more and more into the licensing process. They already are entering in the sense that there exists an effort to reduce the probability of certain initiating events. And, it appears likely to enter from the consequence point of view, as well.

10. WASH-1400 reinforced an already existing point of view that the single failure criterion is useful but is not necessarily adequate. WASH-1400 suggested several specific systems for which the single failure criterion may have provided less reliability than desirable. It is to be expected that some modification of the single failure criterion will evolve, possibly in the form of supplementary quantitative reliability estimates, plus an evaluation of the consequences of system failure, in order to provide additional factors for judgment on the continued acceptability of the single failure criterion in each specific application.

11. There is a strong trend to use the probabilistic methods (and data) of WASH-1400 to make comparative studies of alternate designs and to judge the acceptability of existing situations subject to such analysis, even though an overall probabilistic risk criterion has not been formally adopted. The approach spelled out in the Regulatory Staff Standard Review Plan - namely, to neglect an initiating event having a frequency of less than 10^{-7}/year on a best-estimate basis or 10^{-6}/year on a "conservative" basis, will probably be subject to modification. There may be proposals to accept larger frequencies for individual initiating events on the basis that WASH-1400 suggests that a relatively high core melt frequency does not lead to unacceptably high risk. Some ANS standards groups have already suggested this; the Regulatory Staff has not accepted the proposal, but has not categorically rejected it. And the Staff sometimes appears to lean in that direction in individual scenario evaluations. However, the Commission itself has not defined "acceptable risk."

There may also be a modification which requires that some evaluation of the uncertainty accompany a best-estimate evaluation, and than an "expected value," rather than a best estimate, may become the preferred route.

7.2 CONCLUDING REMARKS

In writing this history, it was hoped that by examining several aspects of light water reactor safety in varying detail, the way in which safety evolved and continues to evolve would be illustrated by example. Such a treatment inevitably leads to a discussion of differences of opinion, of changes in specific policy, of errors and omissions in technical judgment, and of imperfections in the reactor licensing process. It must be recognized that such phenomena are a normal part of any regulatory process for a complex system. It is the borad record of accomplishment and the overall integrity of the process which should provide the basis for evaluation of its success.

In retrospect, the conscious policy of trying to make nuclear power reactors safer than other industrial or technological enterprises, a policy which was adopted in the 1950's and which was spelled out in the ACRS letter to AEC Chairman McCone of October 22, 1960, was particularly vital and long-lasting. Over the years the detailed aproach toward implementation of this policy varied among the regulatory groups and even within a single entity such as the ACRS. However, this continuing policy provided sustained general guidance.

The complex history of reactor siting records several major changes in technical approach by the AEC. However, despite several efforts in the 1960's by the nuclear industry to introduce urban and even truly metropolitan siting in the United States, the Indian Point site, which was approved in 1955-56, remains the most densely populated site approved for a large power reactor. The reasons for failure to gain approval of more densely populated sites varied from case to case. Although prior to 1966 containment was viewed as an independent protective bulwark against most accidents involving core melt, the various regulatory groups were unwilling to approve metropolitan siting at that time, at least for the reactor proposed.

With recognition of the interrelation between core melt and containment failure in 1966, a revolution in LWR safety and licensing occurred. Although the 10 CFR Part 100 site criteria still remained as an AEC regulation to be satisfied, the major emphasis shifted from containment and its associated engineered safeguards to reducing the probability of occurrence of potentially severe initiating events, and to preventing core melt, should an event occur. Some interest in metropolitan siting continued to exist withinthe nuclear industry, but not as much as before. And, with recognition of the problems associated with core melt, the AEC position tended to move away from favorable consideration of such sites.

Within the AEC, some differences remained between the philosophic approaches of the ACRS and the Regulatory Staff on whether increased safety measures were appropriate for the most densely populated sites receiving approval (beyond the somewhat artifical requirements imposed by Part 100). The staff tended to treat all acceptable sites as equal, while the ACRS tended to try to balance the increased numbers of people at risk with additional safety features. The Staff's "black or white" approach may have arisen, in large part, from legal constraints, namely that all reactors which were approved had to meet the existing Rules and Regulations which differentiated among sites only via Part 100.

In retrospect, the results of consequence studies like those in WASH-1400 show large relative differences in the risk between sites like Indian Point/Zion and the more remote sites. Of course, if the risk is acceptably low at the most densely populated site which has been approved, then it is still lower at other sites, and one can argue that the Staff approach is appropriate. However, some philosophic questions then arise concerning the basis by which still more densely populated sites were rejected.

One trend that emerges from the historical review is the general reversal in relative conservatism between the ACRS and the Regulatory Staff with regard to seismic safety requirements. In the late 1950's conventional industrial seismic design practice was applied to the eastern reactors. In the early 1960's, the utilities and their seismic consultants first proposed seismic design bases for reactors in California similar to those previously used for fossil-fueled electricity generating stations in the same area. Then, during the review of Bodega Bay, Malibu and San Onofre, the applicant's originally proposed seismic design basis was made more conservative. This seemed to occur more because of the initiative of the Regulatory Staff and the advice of its consultants than because of the ACRS. Of course, the Staff ended up opposing Bodega Bay while the ACRS wrote a report favorable to its construction. (In retrospect, both were probably nonconservative in their decision on acceptable vibratory motion).

Similarly, for Connecticut Yankee in 1964, the ACRS seemed to feel that the seismic requirements recommended by the Staff and its advisors might be excessive.

After 1966, to the extent that differences on seismic safety design basis arose between the Staff and the ACRS, the ACRS was usually, though not always, on the more conservative side. Why this change occurred is not obvious. It might be due to changes in personnel both on the Staff and on the ACRS. It might be due to the fact that the Staff formed an in-house capability and called less often on the USGS for advice. Or it may be that before 1966, containment was receiving ACRS emphasis rather than accident initiators, while after 1966 the emphasis changed to reducing the probability of an event which could lead to an accident involving core melt.

The various generic safety issues have exhibited very different courses toward resolution. The stemaline break issue, ATWS, and the matter of fires provide three illustrative examples. Fire had been recognized as a potential safety concern of considerable importance for at least a decade before occurrence of the Browns Ferry Fire in 1975. There were large differences in opinion concerning the magnitude of the threat to safety and in the measures needed to make the probability of a serious accident from this cause acceptably low. Significant fires did occur at San Onofre and Indian Point, and these led to some changes in specific requirements. In addition evolutionary improvements were being made in the LWRs under construction, and an industry standard (later shown to be inadequate) was developed. In retrospect, an early clean cut decision to take strong design measures with regard to limiting fire damage would have been preferable to the actual course of action. However, no harm to the public health and safety resulted from the semi-empirical process of learning which was followed, culminating in the Browns Ferry Fire, and the changed requirements which resulted from it.

The steamline break issue represents almost the opposite extreme in regulatory action. When it was recognized in 1972 that the existing design basis for high energy steam and other process lines outside containment did not include gross rupture of the largest pipes, the Regulatory Staff promptly initiated a program which required appropriate changes in existing plants and established new design requirements for those to be constructed. No such accident (that is, rupture of a large steamline) had occurred in a reactor. However, it was judged and judged quickly that all plants had to be protected against such an eventuality.

For obscure reasons, ATWS has had a very different history. Not long after the matter was identified, the Staff judged that scram unreliability was unacceptably high. A quick solution was indicated as needed at least for BWRs where the safety concerns seemed to be relatively well-defined. And, not too long afterwards, at least some of the PWR designs seemed to have unacceptably high primary system pressures for ATWS events.

In the period 1970-71, the ACRS exercised some delaying effect on regulatory action while more detailed technical information was being developed on both the effects of an ATWS event and possible remedial measures. And since 1972, the Regulatory Staff has been struggling to try to reach a firm position, while more and more reactors have gone into operation, and many more have continued to be designed and constructed without including measures to accommodate ATWS or even the flexibility to incorporate such measures readily.

Fires, steamline break and ATWS, all represent potential safety concerns whose probability is and was very difficult to assess, let alone quantify with confidence. It is possible that, in the future, essentially all the reactors will be judged to have been adequately designed without additional provisions for any of these matters. Or, it may be determined in the future that still more would have been preferable.

The absence of a quantified risk acceptance criterion, as well as the lack of ability to quantify the risk from such events with high confidence, leaves engineering judgment as the means for resolution of such matters.

That such problems are reviewed and judged in advance of the occurrence of an accident having severe effects on the public health and safety, is relatively unique in the regulatory field. Most technological ventures have approached safety empirically, with corrections made after the occurrence of one or more bad accidents. So, if the process has been imperfect, at least it has existed.

REFERENCES

1. ACRS, 1972, Letter from Siess to Schlesinger, "Platform Mounted Nuclear Plant", November 15, 1972.

2. Adams, C. A., and Stone, C. N., "Safety and Siting of Nuclear Power Stations in the United Kingdom", Proc. Symp. on Containment and Siting of Nuclear Power Plants, April 3-7,, 1967, Vienna (IAEA).

3. Anderson, D. L., Charwood, R. G., and Chapman, C. B., 1975, On Seismic Risk Analysis of Nuclear Plant Safety Systems: Canadian Society of Civil Engineers, v. 2, p. 558.

4. Ang, A. H-S., and Newmark, N. M., 1977, A probabilistic seismic safety assessment of the Diablo Canyon nuclear power plant (report submitted to the U. S. Nuclear Regulatory Commission in the matter of Pacific Gas & Electric Company, Diablo Canyon Nuclear Power Plant Units 1 and 2, Docket Nos. 50-275 and 50-323; November, 1977): Urbana, Illinois, N. M. Newmart Counsulting Engineering Services, 54 p.

5. Beattie, J. R., Bell, G. D., and Edwards, J. E., "Methods for the Evaluation of Risk", UKAEA Report AH SB(s) R 159, U. K. Atomic Energy Authority (1969).

6. Bender, M., "Impact of Standardization on Resolution of Generic Issues," paper presented at AIF Workshop in Phoenix, Arizona, April 1978.

7. Blasser, G., and Wirtz, K., "Determination of Reactor Location and Requirements for the Pressure Shell on the Basis of the Maximum Credible Accident and the Population Dose Limit," Proc. Symp. on Reactor Safety and Hazard Evaluation Techniques, IAEA, Vienna, 1962 p. 339.

8. BOREX I Destructive Experiment, T. J. Thompson, "The Technology of Nuclear Reactor Safety", Vol. 1, Chapter 11, pg. 622, MIT Press, 1964.

9. Brunot, W. K., Analysis of relative risk associated with operation of the Diablo Canyon nuclear power plant Unit 1 for an interim licensing period (report submitted to the U. S. Nuclear Regulatory Commission in the matter of Pacific Gas and Electric Company, Diablo Canyon Nuclear Power Plant Units 1 and 2, Docket Nos. 50-275-OL and 50-323-OL, September, 1977): San Francisco, Pacific Gas and Electric Company, Department of Engineering, 10 p.

10. Bush, S. H., The ACRS - Generic Issues -and Standardization, unpublished, 1976.

11. E. Case, Status Report to Commission on Single Failure Criterion, 1977.

12. Cave., L., and Illberg, D., "Relative Hazard Potential--The Basis for Definition of Safety Criteria for Fast Reactors", UCLA-ENG-7692, February 1977.

13. Charlesworth, F. R., and Gronow, W. S., "A Summary of Experience in the Practical Application of Siting Policy in the United Kingdom", Proc. Symp. on Containment and Siting of Nuclear Power Plants, April 3-7, 1967, Vienna, (IAEA).

14. Clinch River Breeder Reactor Plant Project office, 1977, CRBRP Safety Study - An Assessment of Accident Risks in the CRBRP: Clinch River Breeder Reactor Plant report CRBRP-1.

15. Davis, W. K., and Robb, J. E., "Nuclear Plant Siting in the United States of America", Proc. Symp. on Containment and Siting of Nuclear Power Plants, April 3-7, 1967, Vienna (IAEA).

16. Epler, E. P., 1969, Common Mode Failure Considerations in the Design of Systems for Protection and Control: Nuclear Safety, v. 10, no. 1, p. 38-45.

17. Ergen, W. K.,"Site Criteria for Reactors with Multiple Containment", Proc. Symp. on Siting of Reactors and Nuclear Research Centers (Bombay), IAEA, 1963.

18. Farmer, F. R., "Reactor Safety Analysis as Related to Reactor Siting", Proc. 3rd Int. Conf. on Peaceful Uses of Atomic Energy, Geneva, 1964, (United Nations).

19. Farmer, F. R., "Siting Criteria -A New Approach", Proc. Symp. on Containment and Siting of Nuclear Power Plants, April 3-7, 1967, Vienna (IAEA).

20. Gillette, Robert., News and Comment, Science, vol. 177-1972.

21. Gossett, B., et al., "Post-Accident Filtration as a Means of Improving Containment Effectiveness, UCLA-ENG-7775, Dec. 1977.

22. Hanauer, S. H., Chairman, "Recommendations Related to Browns Ferry Fire", Report by Special Review Group NUREG-0050, February 1976.

23. Hsieh, T., and Okrent, D., 1976, "Some probabilistic aspects of the seismic risk of nuclear reactors: University of California, Los Angeles, School of Engineering and Applied Science Report UCLA-ENG-76113, 75 p.

24. Hsieh, T., and Okrent, D., 1977, On design errors and system degradation in seismic safety, in Jaeger, T. A., and Boley, B. A., general eds., Transaction of the International Conference on Structural Mechanics in Reactor Technology, 4th.

25. Johnson, W. E., "Principles and Practices in 'Consequences Limiting' Safeguards in Facility Design", Proc. Symp. on Reactor Safety and Hazard Evaluation Techniques, IAEA, Vienna, 1962, p. 349.

26. Kellerman, O., "Accident Analysis as the Basis of Technical Examination for the Safety of Nuclear Reactor Power Plants", Proc. Symp. on Reactor Safety and Hazard Evaluation Techniques, IAEA, Vienna 1962, p. 483.

27. Kellerman, O., and Seipel, H. G., "Analysis of the Improvement in Safety Obtained by a Containment and by Other Safety Devices for Water-Cooled Reactors", Proc. Symp. on Containment and Siting of Nuclear Power Plants, April 3-7, 1967, Vienna (IAEA).

28. Keoing, J. B., 1963, The geologic setting of Bodega Head: Mineral Information Service (California Division of Mines and Geology), v. 16, no. 7, p. 1-10.

29. Levy, S., "A Systems Approach to Containment Design in Nuclear Power Plants", Proc. Symp. on Containment and Siting of Nuclear Power Plants, April 3-7, 1967 Vienna (IAEA).

30. Lewis, H. W., Chairman, 1975, Report to the American Physical Society by the Study Group on Light Water Reactor Safety: Review of Modern Physics, v. 47, supp. 1, p. S1-S123.

31. Marshall, W., 1976, An Assessment of the Integrity of PWR Pressure Vessels, Report by Study Group: United Kingdom Atomic Energy Authority Report.

32. Morris, P. A., and Waterfield, R. L., "Site Evaluation and Diffusion Calculation Procedures in the USA" Proc. Symp. on Containment and Siting of Nuclear Power Plants, April 3-7, 1967, Vienna (IAEA).

33. NRC, 1976, Evaluation of the Integrity of Reactor Vessels Designed to ASME Code Sections 1/8 NUREG 0081, June 1976, K. Hoge.

34. NRC, 1978, NRC Program forthe Resolution of Generic Issues Related to Nuclear Power Plants NUREG-0410, January 1, 1978.

35. NUREG-0438, Plan for Research to Improve the Safety of Light Water Nuclear Power Plants, April 12, 1978.

36. Record of the AEC Regulatory Staff Symposium held on April 29, 1965, at Germantown on Possible Zirconium-Water Reactions in Water Reactors, Division of Safety Standards, USAEC, June 16, 1965.

37. Revision I of Phase One Report by the Study Group on Metal-Water Reactions in Nuclear Reactors, February 1966.

38. Russell, Charles R., Reactor Safeguards, Oxford, New York, Pergamon Press, 1962.

39. Schlocker, J., and Bonilla, M. G., 1964, Engineering Geology of the Proposed Nuclear Power Plant on Bodega Head, Sonoma County, California: U.S. Geological Survey, Administrative Report to U. S. Atomic Energy Commission, 1964.

40. Shibata, H., A Seismic Design of Nuclear Power Plants - Developments of these Ten Years, Japan Atomic Energy Research Institute, November, 1970 (NSJ-Tr 161).

41. Thompson, T. J., The Technology of Nuclear Reactor Safety, vol. 1, Chap. 7, 8, and 11, Thompson, T. J. and J. G. Beckerley, MIT Press, 1964.

42. United Nations, 1st International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955.

43. USAEC, 1950, U.S. Atomic Energy Commission Summary Report of Reactor Safeguards Committee, WASH-3, March, 1950.

44. U. S. Atomic Energy Commission, 1966, Atomic Safety and Licensing Board Initial Decision in the Matter of Department of Water and Power of the City of Los Angeles (Malibu Nuclear Plant Unit No. 1): (U.S.) Atomic Energy Commission Reports, v. 3, p. 122-142.

45. U.S. Atomic Energy Commission 1967, Decision in the Matter of Department of Water and Power of the City of Los Angeles (Malibu Nuclear Plant Unit No. 1): (U.S.) Atomic Energy Commission Reports, v. 3, p. 179-189.

46. U.S. Atomic Energy Commission, 1973, Technical Report on Anticipated Transients without Scram for Water-Cooled Power Reactors: U.S. Atomic Energy Commission Report WASH-1270, 92 p.

47. USAEC, 1974, Technical Report on Analysis of pressure vessel statistics from fossil-fueled power plant service and assessment of reactor vessel reliability in nuclear power plant service: U.S. Atomic Energy Commission Report WASH-1318.

48. USNRC, IE Investigation Report 50-338/76-28, 50-339/76-16, Subject: "Investigations of Alledged discrepancies in the construction and quality control program for piping installation at the North Anna Power Station."

49. Vesely, W. E., et al., The OCTAVIA Computer Code: PWR Reactor Pressure Transients, USNRC Report NUREG-0258, Mar. 1978, 46 p.

50. Vinck, W. F., and Maurey, H., "Some Examples of the Relationship between Containment and Other Engineered Safeguard Requirements, Accident Analyses and Site Conditions", Proc. Symp. on Containment and Siting of Nuclear Power Plants", April 3-7, 1967, Vienna (IAEA).

51. WASH-740, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, USAEC 1957.

52. WASH-1400, Reactor Safety Study (AEC 1974).

53. Windscale No. 1, Memo to Parliment, "Accident at Windscale No. 1 Pile, 10th October 1957, (Cmmd. 302) Her Majesty's Stationery Office, London, 1957.