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Conceptual designs of advanced fast reactors

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CONCEPTUAL DESIGNS OF ADVANCED FAST REACTORS
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FOREWORD

Liquid metal fast reactors (LMFRs) have been under development for more than 45 years. Twenty LMFRs have been constructed and operated. Five prototype and near-commercial-scale LMFRs (BN-350/Kazakhstan, Phénix/France, PFR/United Kingdom, BN-600/Russian Federation, Superphénix/France), with an electrical output of between 250 and 1200 MW(e), have accumulated more than 85 reactor-years of operating experience. Altogether, LMFRs have accumulated more than 280 reactor-years of operating experience. In most cases the overall experience has been satisfactory; however, some incidents or failures occurred in the operation of the first prototype reactors. These were thought to be due to faulty design or fabrication.

Large scale LMFR plants were planned at the time when uranium prices were high, rising, and expected to continue to climb. Under those circumstances it was expected that LMFRs, even with high capital costs, would eventually be able to compete with LWRs due to breeding of new fissile materials.

The nuclear community now faces conditions which were not foreseen two decades ago. The most important of them are stable, low prices for uranium, a lower electricity demand growth rate, and the improved fuel utilization in modern LWR nuclear plants. Although fast reactors will likely be necessary both to meet long term fissile materials requirements and to reduce the quantities of long lived radioactive waste, it is now recognized that FR technology must be developed further to make FRs more economically competitive with LWRs. This means that improved LMFR design drawing upon the lessons learned from existing plants, which are costly and take longer to construct than thermal reactors, should result in plants that are simpler and cheaper to construct while maintaining higher safety levels. In addition of being economically competitive with LWRs, fast reactors are beneficial from the environmental viewpoint, since they are capable of converting radioactive waste to more environmentally benign forms and significantly reduce the amount of uranium that must be mined.

Considerable efforts have been made in recent years in France, the Russian Federation, Japan, the USA and India to lower the capital costs of advanced LMFRs. It appears that the goals of LMFRs economically competitive with LWRs can be reached. In this context, a Technical Committee meeting (TCM) was held on Conceptual Designs of Advanced Fast Power Reactors to review the lessons learned from the construction and operation of demonstration and near-commercial size plants. This TCM focused on design and development of advanced fast reactors and identified methodologies to evaluate the economic competitiveness and reliability of advanced projects.

The Member States which participated in the TCM were at different stages of LMFR development. The Russian Federation, Japan and India had prototype and/or experimental LMFRs and continue with mature R&D programmes. China, the Republic of Korea and Brazil were at the beginning of LMFR development. Therefore the aims of the TCM were to obtain technical descriptions of different design approaches for experimental, prototype, demonstration and commercial LMFRs, and to describe the engineering judgements made in developing the design approaches.

The IAEA wishes to thank all the participants for their contributions to this publication.

EDITORIAL NOTE

CONTENTS

SUMMARY AND CONCLUSIONS OF THE TECHNICAL COMMITTEE MEETING

1. INTRODUCTION

The liquid metal reactor technology has demonstrated a number of effective designs, project realization and experience in construction and operation. Some incidents and/or failures have been experienced in the operation of the first prototype and demonstration reactors, but it should be noted that the mechanical failures in LMFRs were more a consequence of faulty detailed design decisions and/or fabrication errors than faults in the basic concept. The operating experience indicated that some LMFR design aspects should be improved.

The construction of BN-600, SPX and Monju have indicated that LMFRs are costly and take more time for construction than thermal reactors. Therefore, efforts are being made to reduce the capital costs and construction time.

The safety record of LMFR is good and many encouraging results have been obtained on controllability, shutdown systems, decay heat removal and other major systems and components. Certain improvements in safety are being considered, such as prevention, detection and mitigation of leaks and fires of sodium, passive shutdown and decay heat removal systems, core catcher and improved containment.

Fast reactor programmes have recently been initiated in the People's Republic of China, the Republic of Korea and Brazil. International exchange of information and experience is becoming of increasing importance to these countries.

In view of these circumstances the International Working Group on Fast Reactors at its Annual Meeting in May 1994, recommended to convene a technical committee meeting to review and discuss the following general topics: (1) lessons learned from experience with prototype and demonstration plants, applicable to design of advanced LMFRs; (2) developments in advanced LMFRs and their general design features; (3) anticipated performance of advanced LMFRs; and (4) status of R&D programmes, findings and conclusions, future needs and trends.

The Technical Committee Meeting (TCM) on Conceptual Designs of Advanced Fast Power Reactors, was hosted by the Indira Gandhi Centre for Atomic Research (IGCAR). About thirty specialists from Brazil, China, India, Japan, the Republic of Korea and the Russian Federation participated in the TCM and presented a total of 18 papers. A technical tour of the IGCAR facilities, including experimental liquid metal fast breeder test reactor FTBR took place during the TCM.

2. SUMMARY OF TECHNICAL SESSIONS

2.1. Russia's experience of LMFRs designing, construction and operation

The BN-350[1] prototype reactor. The BN-350 LMFR, the first nuclear reactor in the world producing heat for seawater desalination and electricity production, was commissioned in 1973. The BN-350 demonstrated reliable performance of reactor equipment and the plant achieved an average load factor of 85% at established power levels. The problem of deformation in core subassemblies, due to neutron irradiation experienced in the initial years, has been overcome by changing to new structural materials. The 12 control rod drive mechanisms (2 control rods, 3 shutdown rods and 7 reactivity compensation rods) have functioned normally. During the whole period of the reactor's operation, 56 planned refuelling cycles have been completed. Sodium aerosols deposition in shield plugs increased resistance to rotation. The intermediate heat exchanger has operated for more than 160 000 hours without any problems. The main factor limiting the reactor power level have been failures in the steam generators (SGs). The SG consist of 2 superheaters with U-shaped tubes and 2 evaporators with straight tubes, inside which water flows under natural convection. The initial years operation of SG were unreliable on account of frequent tube-weld failures. The radiological release since commissioning has remained within acceptable levels. Detailed life extension studies for the reactor are in progress and include ensuring reactor safety and structural integrity during seismic events.

The BN-600 demonstration reactor. The BN-600 (1470 MWth/600 MWe, pool type) which was connected to grid on April 8, 1980, has successfully completed 15 years of operation. The commercial operation began in 1982. The initial UO_2 fuelled core consisted of two U-235 enrichment zones, i.e., 21% and 33%. In order to improve the fuel performance, the core was modified in 1987. The modifications included provision of three uranium enrichment zones with 17-21-26% enrichment, increasing the peak burnup from 7.3% to 8.3% and decreasing the linear heat rating from 51 to 47 kW/m. The decrease in linear heat rating increased the fuel column height from 75 to 100 cm. In 1993, with the use of advanced structural materials, the burnup was increased to the design value of 10% heavy atom. Efforts are under way to further increase the burnup to 11% ha. The life of control rod guide tubes was increased (from 200 to 640 ed) due to the use of advanced structural material. The operation of primary sodium pump was successful except during the initial periods in which the reactor trips were caused by failures of gear couplings and shafts. The origin of these failures was traced to the shaft torsional vibrations. The experience with intermediate heat exchangers(IHXs) has been trouble free. The reactor has 3 secondary loops each consisting of a SG, a secondary sodium pump and a turbogenerator. The SG is modular in construction with an evaporator, reheater (sodium reheat) and a superheater. The isolation valves in SG enabled to isolate a faulty module and continue the reactor operation. There have been 12 instances of water into sodium leaks in SG during 15 years of the BN-600 reactor operation and several sodium leaks from primary and secondary circuit. The large leaks of secondary sodium happened on May 6, 1994 in a drain pipe of an IHX during repair work when the reactor was shut down. The sodium leak was accompanied by burning and some equipment damage in the adjacent area. Approximately 1.2-1.3 m^3 of sodium were lost. The complex of measures used in BN-600 to detect sodium leaks, to localize them and to fight with sodium fires allowed to avoid serious fire incidents. The mass of the burned sodium was of the order of several tens of kilograms. The secondary sodium pumps have operated trouble free for about 105 000 hours. As regards the operation of TG, of late, the

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[1] Located in the Republic of Kazakhstan.

8

functioning of drain lines and valves are considered critical for the entire power plants' performance. The power plant performance has matched the performance of NPPs worldwide. The average plant availability is 76.6% with planned outages constituting 21.7% and unplanned outages constituting 1.7%. The 15-year operation of BN-600 has demonstrated reliable and safe operation of pool type sodium cooled fast reactors and successful commercial exploitation of LMFR.

Conceptual design of advanced reactor BN-600M. While the long-term objective of BN-600M is to provide nuclear power and fuel, the short-term objective is to burn the weapon grade plutonium, and minor actinides. The main concepts for BN-600M have been adopted from the BN-600 and BN-800 designs. However, there are modifications towards enhancing the safety and reducing the cost. The shape and support of the main vessel are similar to that of BN-600. The main vessel is cooled by sodium leakage from diagrid. Additional safety features such as negative sodium void reactivity, hydraulically suspended absorber rods, provision of core catcher, etc. are included. In order to avoid excessive heat generation in invessel storage subassembly boron carbide shielding is provided and storage subassemblies are cooled by natural convection. Four decay heat removal heat exchangers each of 15 MWth capacity are provided while two are sufficient to limit the sodium temperatures below the design basis value. Towards reducing the cost it has only 2 primary pumps and IHXs. The plenum for feeding the primary pumps is common. However, the number of secondary loops has not yet been decided, which may be either 2 or 4 depending upon the type of SG selected. The specific weight of BN-600M will be reduced by 1.25 (2 secondary loops with modular SG) or 1.58 (vessel type SG) times that of BN-600. Fuel burn-up will be increased by 12.7% ha (I phase) and by 15.8% ha (II phase). Future works will be devoted to the SG, secondary pipe work, development of bellows, testing DHRs in sodium and water etc.

Plutonium and actinide burner advanced reactor. The present strategy in Russia to solve power and ecological problems is to use FBRs to burn plutonium and minor actinides. It is feasible to develop a core with MOX fuel for effective plutonium utilisation on the basis of using of fuel pin structure with increased enrichment of plutonium and also using absorber material in core, which also provided safety sodium void reactivity effect negative or near zero. Advanced core of BN-800 type reactor for plutonium utilisation also allows spent fuel removed from WWER type reactors to be recycled and used leading to reduction in radiotoxicity. Considered scenarios in Russia in plutonium utilisation also ensure non-proliferation.

2.2. Recent progress in conceptual design of LMFR in China

The experimental reactor CEFR-25 has a capacity of 65 MWth/25 MWe and may use UO_2 fuel for first core and subsequent core will have MOX fuel. It is a sodium cooled pool type experimental fast reactor. There are 2 primary sodium pumps and 4 IHX in the main vessel, 2 secondary loops, single steam/water circuit and one TG set. The decay heat removal is provided by two independent (0.5 MWth each) passive residual heat removal systems. The main vessel is top supported and is provided with a guard vessel. Two rotable plugs with one straight pull machine are used for fuel handling. The primary sodium purification system is located outside the vessel. The paper also lists other major parameters that have been considered in the design of the reactor. The present planned burnup is 50,000 MWd/t with a future targeted burnup of 100,000 MWd/t. The concern has been to limit radioactivity releases during operational states (normal operation and anticipated operational occurrences) and accident states to within acceptable limits as the site selected is close to Beijing city (40

km). This project is the first step towards development of LMFR in China; development in LMFR technology is viewed from the angle of meeting the energy requirements in the next century.

2.3. An overview of Brazil LMFR conceptual design

In Brazil the primary motive in FR development is to gain design and operating experience in LMFR systems. The reference design for 60 MWth/20MWe REARA (REActor RApidos) is as follows. It is a pool type, U-10%, Zr metallic alloy fuelled reactor with active fuel height of 62 cm and having 30 cm Ni reflectors at top and bottom. There are 6 B_4C control assemblies (4 primary + 2 secondary) for controlling/shutting down of the reactor. The neutronic calculations have been done using EXPANDA and CITATION codes and the corresponding burnup figures are given. The main parameters for primary sodium pump and IHX have been selected based on EBR-II and PRISM designs and process design of IHX has been done with computer code TCIPRO. The purpose of establishing the reference design is to put forth the case of LMFR to the "decision makers" and convince them of the potentials of LMFR technology.

2.4. The conceptual design of fast breeder reactor in Japan

Demonstration reactor. The DFBR is being developed as a next step of MONJU towards commercialization of LMFR around 2030. It is a 1600 MWth/660 MWe, PuO_2-UO_2 fuelled, sodium cooled, top entry loop type reactor. The primary cooling system consists of a reactor vessel, three IHX vessels, three pump vessels and inverted U-shaped piping to connect these vessels. The argon cover gas system in these vessels is interconnected and maintained at 0.9 kg/cm². The diameter and height of the reactor vessel are 10.4 m and 16 m respectively and the thickness is 50 mm. The roof structure supports the vessel and is of box type construction (13.0 m dia, 6.2 m height and 60 mm thick plates). Both 316 FR and SUS 304 are used in construction of sodium cooling systems. The primary sodium flows on the tube side of IHX and the primary pump is of single stage single suction type. The design of SG has been based on "MONJU" design and is of integrated once-through type with helical coil arrangement for tubes. 9 Cr-1 Mo is used in construction of SG. The reactor has two types of shutdown systems to arrive at a level of safety equal to that of the LWR. The reactor containment building is designed to withstand a temperature of 150°C and a pressure of 0.5 kg/sq.cm under condition of sodium ejection into RCB after a core disruptive accident during which 100 Kg of Na is assumed to burn. The emergency power supply is provided by 3 independent diesel generator sets and the reactor protection uses a "2 out of 4 logic". The design of RCB incorporates suitable isolation in order to take care of seismic events. Rectangular and cylindrical reactor buildings were studied and the former is selected because of compact layout. In all the studies of the conceptual design phase of the DFBR, the aim has been to make the design harmonious with both safety and economic viability and to provide preparation for the basic design.

Advanced design of fast breeder reactor in PNC. Diverse needs including reduction of impact to the global environment and enforcing greater thrust on non-proliferation aspects were introduced. Nitrides are considered as promising candidate as fuel. Details of the core, the new fuel concept and related aspects of interface with plant design were described. It was mentioned that the design of an advanced recycle fast test reactor has just started which will facilitate to evaluate all the features considered in the advanced concept.

10

2.5. Conceptual design of Indian prototype fast breeder reactor

Main options. In India fast breeders are considered a key alternative to coal in meeting the energy demands. The significance of capital cost and construction period on economy was elaborated. Sodium coolant, pool type concept, oxide fuel, 20% cold work D9 for fuel clad and hexcan, SS 316LN for main hot leg components and sodium piping and modified 9Cr-1Mo steel (T91) for steam generator tubes have been selected for PFBR. Steam temperature of 763 K at 16.6 MPa and a single TG of 500 MWe gross output have been chosen.

Reactor assembly. The conceptual design of PFBR selected in 1995 was revised with the aim of simplification of design, compactness of assembly and ease in construction. The reduction in size has been possible by incorporating concentric core arrangement, adoption of elastomer seal for rotating plug, fuel handling with one transfer arm mechanism, incorporation of mechanical sealing arrangement for IHX at the penetration in inner vessel redan and reduction in number of components. Warm roof slab concept has been chosen to avoid sodium vapour deposition in the gaps between rotatable plugs. A single grid plate of fully bolted construction has been adopted to simplify manufacture. The main vessel cooling circuit consists of cooling pipes drawing coolant from leakage from core subassemblies and passing the coolant through thermal baffles on the main vessel. A core catcher is provided below the grid plate as defence in-depth philosophy. The sequence of erection of reactor assembly components was also explained.

Reactor core. The mixed oxide fuel care of PFBR is conventional, homogenous core with two enrichment zones and radial and axial blankets of depleted UO_2. Pin diameter has been chosen on lower side at 6.6 mm for reducing the core plutonium inventory below 2 t. Considerations for the choice of number of pins per subassembly, integrated vs separate axial blankets, and other pin and subassembly parameters were discussed. No special schemes have been explored for reducing the maximum positive sodium voiding coefficient as the core size is moderate and the coefficient is about 3.5$. Twelve control rods have been chosen in a layout optimised for maximum antishadow effect. The optimisation of layout of radial and axial shielding was also discussed.

Heat transport systems components. A single stage impeller design was chosen for primary pump. The non-return valve chosen for earlier design was eliminated to permit increased submergence under free sodium level. For IHX, straight tube design with primary sodium on shell side has been selected. Steam reheat cycle has been chosen for PFBR to effect cost reduction for SG and associated circuits, reduce construction time and provide ease in design and operation. Straight tube once through steam generator with expansion bend on each tube welded to tubesheet with raised spigot has been selected.

Core component handling system. In vessel handling in PFBR comprises of two rotatable plugs and a fixed transfer arm for handling subassemblies for in vessel transfer between core, internal, storage and in vessel transfer position. From IVTP, subassemblies are transferred to ex-vessel transfer position by inclined fuel transfer mechanism. For spent fuel cooling outside the reactor, water pool storage has been chosen. The economic considerations in the choice of the design were explained.

Safety instrumentation. The flux measurement for monitoring core power is proposed to be done in a location below the main vessel. However, for first startup, in-core fission chambers are provided. Provision has been made for monitoring outlet temperatures of all

fuel subassemblies. Global monitors for delayed neutron detection in sodium and fission gas activity in cover gas have been provided for detection of failed fuel. For localisation of failed fuel sodium from individual subassemblies are monitored. Primary sodium flow is measured by permanent magnet flowmeters provided in a bypass line. Sodium leak will be detected by sodium aerosol detector and spark plug detectors. Water leak in steam generator is monitored by mass spectrometer based hydrogen leak detection systems. Protection logics for scram of the two shutdown systems are based on different principles for diversity. To achieve required reliability it is proposed to couple the two shutdown systems optically.

Safety considerations in the design. The general safety approach in PFBR design is governed by principles of redundancy, diversity, independence, fail safe design, sound design and operating experience of various LMFBRs. The temperature and power coefficients of reactivity are negative. Subassembly blockage is prevented by multiple opening entries and an adaptor at the exit. Two independent fast acting diverse shutdown systems have been chosen. Curie point magnetic switch triggering scram on secondary shutdown systems is proposed to be provided to protect against abnormal coolant temperature rise. Four decay heat removal circuits each comprising a dip heat exchanger in the hot pool of the reactor have been provided. The circuit is passive ensuring removal of decay heat from the reactor by natural convection. In addition, battery power to drive the primary pumps at 20% of the speed for one hour is also provided. Measures to control sodium fire were also presented. Sodium void reactivity is about 3.5$. Studies made for loss of flow without scram for flow coast down times of 2 s and 10 s showed that reduction in sodium void coefficient has the effect of delaying fuel melting by 10 to 40 minutes. Guillotine rupture of pump to diagrid pipe is considered very remote as leak before break argument seems to be applicable to this component. However, the guillotine rupture is proposed to be monitored by primary coolant flow measurement, fast response thermocouples and eddy current flow meters at selected subassembly locations and reactivity. A core catcher below the grid plate for post accident cooling of molten fuel debris from 7 subassemblies has been provided to protect against total instantaneous blockage of a single subassembly. The main vessel has been designed for an energy release of 200 MJ from core disruptive accident. The containment building is designed for overpressure of 25 KPa arising from burning of 500 kg of sodium ejected from reactor vessel during CDA.

2.6. Korean advanced concept of LMFR

Reactor design features and core performance. KAERI is developing the first FBR in the Republic of Korea. The main options selected for the KALIMER (Korea Advanced LIquid MEtal Reactor) were discussed. It is a modular pool type sodium cooled LMFR and its construction is expected to be over by 2011. The electric power per module is 333 MWe. It is designed to use metal fuel, initially U-Zr and later Pu-U-Zr. It uses electromagnetic pumps for the primary as well as secondary circuits. The number of primary pumps in the primary circuit is 4 while that in the secondary circuit is 2. It adopts the integrated concept which combines SG, intermediate sodium pumps and sodium expansion tank in a single compact component. The other unique feature includes use of integrated SG elimination of rotating plugs and simplification of invessel transfer machine. Passive safety features of KALIMER design include the reactor vessel auxiliary cooling system which assures safety grade decay heat removal and the self- actuated shut-down system (based on Curie paint magnet) and gas expansion module. It is designed to provide a strong inherent negative reactivity feedback with rising temperature. This along with RVACS, makes it capable of safely withstanding severe under-cooling and overpower transient events without scram.

3. CONCLUSIONS OF TECHNICAL SESSIONS

1. The experience gained with fast reactors has proved their safe operation and reliability and has revealed no fundamental problems with reactor physics, operation of various equipment and sodium technology. This experience will be useful in upgrading the design of next generation LMFR.

2. Research on LMFR during the last decades has significantly improved understanding of LMFR designs, technology and safety. But there are small a number of safety concerns which influence fast reactor licensing and safety analysis. In this particular context there still is a sodium leaks and fires. According to the experience gained with long-operating LMFR, sodium leakage took place in sodium containing systems and components. The BN-600 reactor during 15 years of operation had several small leaks and one of $\sim 1m^3$ leaks in its secondary circuits. The potential risks associated with large sodium leaks include: loss of function in the system in which the leak occurs, sodium-concrete reactions, sodium fire and associated phenomena. Prevention, detection and mitigation of sodium leaks, improved resistance of nuclear systems to fires and development of reactor equipment and systems designs for elimination of leaks is still an important direction of LMFR technology research.

3. Two principal design concepts have been discussed at the TCM: the loop and the pool type. The pool concept was chosen for all small, medium and large size advanced fast reactors: REARA (Brazil), CEFR (China), PFBR-500 (India), KALIMER (Korea, Republic of), BN-600M (Russia) and EFR-1500 (European Fast Reactor), except for DFBR-660 (Japan), which uses the top-entry loop-type reactor. The pool reactor placed in a guard vessel has been shown to have very attractive safety characteristics, resulting to a large extent from a liquid metal cooled reactor being a low pressure radioactive system with large thermal inertia. This type of design practically excluded unfavourable consequences of failures in the external radioactive systems and loss of reactor coolant. The top-entry loop-type design was selected for the Japanese DFBR because of the following considerations: (a) major primary components such as the IHX and the pumps are outside of the reactor vessel, and this facilitates maintenance and repair; (b) the system has flexibility to introduce such innovative technologies as the electromagnetic pump integrated component, which is needed for commercialization of the FBR, and (c) experience gained at the prototype "Monju" must be fully utilized. Considering that the top entry system is quite a new concept, the conceptual design study, the evaluation study of commercialization prospects and the water hydraulic tests using models of thermal-hydraulic properties peculiar to the top-entry system were conducted.

4. Although differences in conceptual design approaches were discussed at the meeting, a number of common topics could be identified among the conceptual design approaches presented. These included improvements with regard to safety, design simplifications and reduction in cost. Safety improvements included among other the consideration of core catcher design (excluding of the recriticality, cooling capabilities), passive backup reactivity shutdown and decay heat removal systems, a strong inherent negative reactivity feedback with rising temperature. Economic competitiveness should be improved due to the optimization of the number of cooling loops and equipment and achievements of reducing its weights as well as the building volumes, findings of structural materials, fuel technology and core design to achieve high burnup, and limiting the number of safety graded systems.

5. The development of simple, reliable, efficient and flexible systems and components is a primary objective in the design of advanced fast power reactors. Systems and components for developmental designs (advanced reactor designs which range from

moderate modification of existing designs to entirely new design concepts) in general require much extensive testing and demonstration to verify component and system performance. Key issues are scaling effects for simulation plant configuration, design life and interactions among different systems.

3.   SUMMARY AND CONCLUSIONS OF THE CLOSING SESSION

3.1.   Update on LMFRs progress

Significant technology development programmes for LMFRs are proceeding in several countries, notably Japan, India and the Russian Federation. Activities continue in a number of other countries at a lower level.

Brazil. Brazil foresees fast reactors as an efficient alternative source of energy and considers it as a component in the long term energy planning. A start is being made towards the design of a fast reactor REARA and construction of this reactor would be the aim in the coming years.

China. The basic research work of FBR was started in 1964. Since then and up to 1987, the major work was on neutronics, thermal hydraulic and sodium technology. Some small scale facilities have been built. During 1988-1990, the major work was to prepare design codes and select main options for CEFR. During 1991-1992, the conceptual design of CEFR was completed and during 1992-1993, the conceptual design was confirmed and optimisation studies were carried out. Since 1993 onwards, major work involved preparation of detailed design. Approval for construction of CEFR is under consideration.

India. FBTR is operating at 10.5 MWt power with MARK I small PuC-UC core. Power will be raised to 40 MWt next year with MARK II full size core. Fuel development and material irradiation apart from sodium technology would be the principal programme for the reactor. While the need for FBRs is well appreciated in the country, their introduction is linked to economic acceptability. In order that series construction provides better potential for realisation, it is important to incorporate in the "Head of the Series" all the design features that reduce cost and construction time. With basic design features selected now for PFBR, the emphasis in next two years will be on detailed design, engineering development, sodium technology and materials technology. Reduction in construction time is an important target.

Japan. In Japan, the fast breeder reactor development program symbolizes the national nuclear fuel recycling program, which was originally decided in the national "Long-Term Program for Research, Development and Utilization of Nuclear Energy", issued in June, 1995, by the Atomic Energy Commission of Japan. The experimental fast reactor "Joyo" reached initial criticality in April, 1977 and since then, has shown excellent performance for more than 18 years. The prototype reactor "Monju" with the capacity of 280 MWe also reached initial criticality in April, 1994 and has delivered electricity for the first time in August, 1995. Plant design is successfully proceeded by the design of DFBR, 660 MWe, which is expected to be constructed in the beginning of the next century. In addition to this main stream of development work, advanced studies, regarding technology capable of meeting diverse needs of future society including reduction of impact on the environment and assurance of nuclear non-proliferation, thereby widening the technological options, are under progress.

14

Republic of Korea. The Republic ofKorea plans to develop the conceptual design of its first FBR, KALIMER, by 2001. Construction of this reactor is planned for criticality to be achieved during 2011. The design of KALIMER will be refined and specified emphasizing excellent plant economy.

Russian Federation. Russia's experience in the operation of experimental and prototype fast reactors BR-10, BOR-60 & BN-600 has been very good and has vindicated the original faith in FBRs as the ultimate goal for efficient utilisation of uranium. The current efforts with regard to LMFRs in Russia are directed towards improving the safety and reliability of fast reactor and also making them economically competitive to other energy sources. While these efforts would take some time, an immediate use is foreseen for fast reactors as Pu and minor actinide burner from waste disposal and ecological considerations.

3.2. Conclusions of the closing session

- Fast reactors have demonstrated high level of operational reliability and safety.
- Fast reactor design and development programmes are in progress in Brazil, China, India, Japan, Republic of Korea and the Russian Federation. An important role is foreseen for fast reactors in the supply of energy in future.
- The next generation of fast reactors in India, Japan and the Russian Federation are being developed to further increase the reliability and safety, reduce costs and improve ecological characteristics of fuel cycle.
- The participants found the meeting very useful in knowing design trends in each of the participating countries and recommended that periodic (~ 3 years) meetings should be held to discuss in detail the progress made in the fast reactor designs.

LMFR DESIGN AND OPERATION EXPERIENCE

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Abstract

This paper presents general information on the designs of LMFR in the Russian Federation, especially devoted such basic problems as types of primary circuit arrangement and reactor vessel support systems. Operating experience of BN-350 and BN-600 components, (namely core, main circulating pumps, intermediate heat exchangers, handling systems and steam generators) are summarized.

1. LMFR DESIGNS

Intrinsic to the LMFR design is the option of primary circuit arrangement: either loop or pool type. Loop type arrangements is found more often for nuclear facilities and also used for the first prototype reactor BN-350 (Fig. 1.) This arrangement was based on the experience with the experimental reactors BR-5 and BOR-60. A typical example of LMFRs with integral arrangement is BN-600 (Fig. 2.). At the beginning the reactor was designed as loop type, based on design decisions for BN-350. But significant problems arose concerning reliable operation of the primary circuit large diameter pipes for BN-600 parameters. As a result of comprehensive optimization during the second stage of design, integral principle of the primary circuit arrangement was chosen. At present, reactors BN-350 and BN-600 with different conception of primary circuit are in successful operation. This operation experience has not revealed decisive advantages of any conception, proved operability of both variants and necessity to choice a type for arrangement, providing significant reduction of a cost. One problem which had to be solved with the BN-350 and BN-600 design was the problem with the reactor vessel support. The idea of a top support for the reactor vessel of BN-350 was born at the initial phase of the design development. A standard support system was the adopted being also in wide use for reactor facilities. A comparative analyses of top and bottom support systems were carried out in going to pool type arrangement. The main results of the analysis are presented below.

As a result, a bottom support for the main vessel was selected for BN-600. Basic reasons for the final option were more simple structure of the upper closure, lesser thermal extension of primary circuit pressure pipes relative to pumps and more preferable working conditions of the vessel support in normal operation and transients.
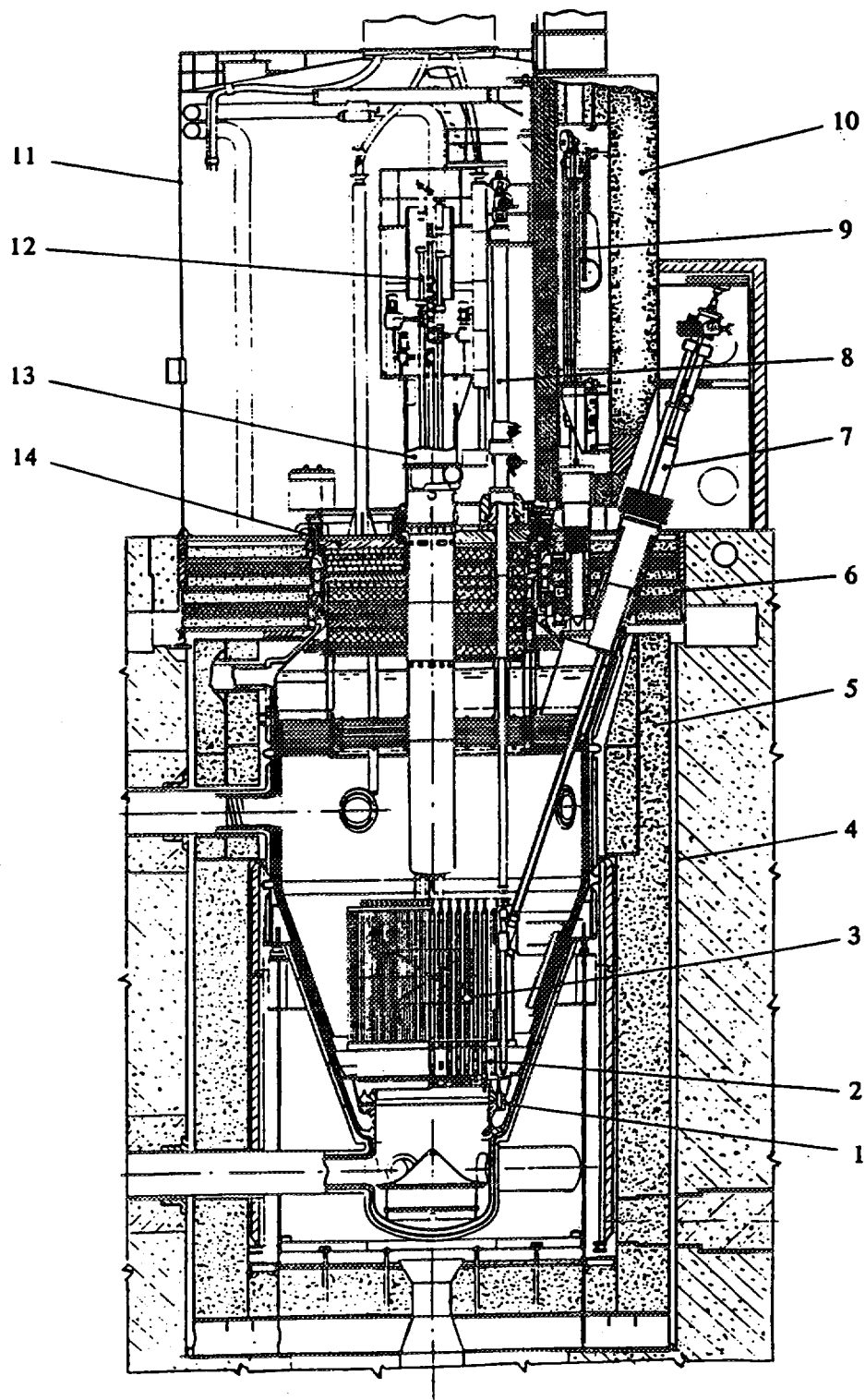
Fig. 1. BN-350 Reactor

1 - reactor vessel; 2- core diagrid; 3 - reactor core; 4 - reactor well liner; 5 - lateral shield; 6 - upper-stationary shield; 7- elevator; 8 - refuelling mechanism; 9 - FAs transfer mechanism; 10 - fuel transfer cell; 11 - protective dome; 12 - control rod drive mechanism; 13 - above core structure; 14 - rotating plugs
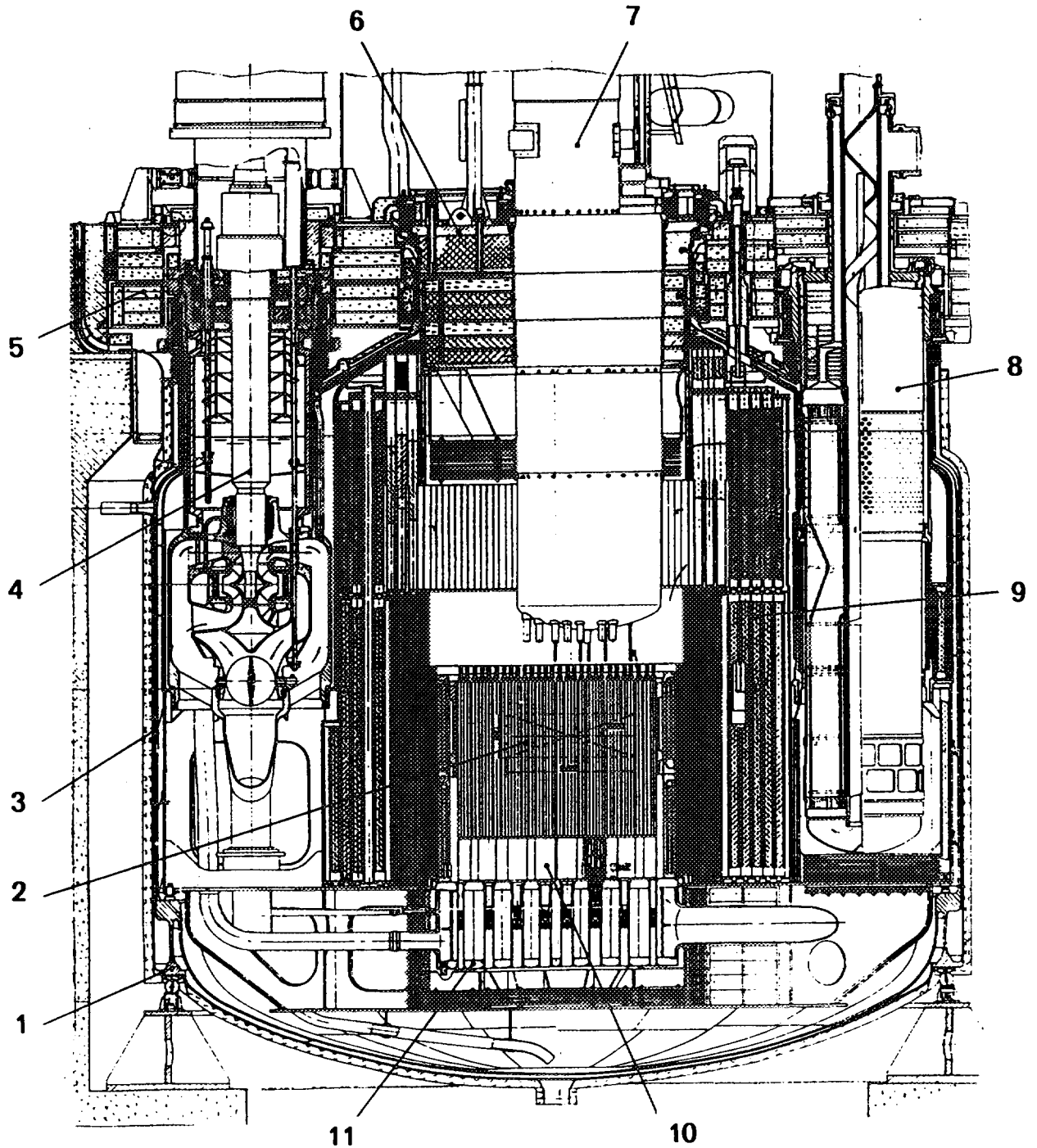
Fig. 2. BN-600 reactor cut-away view
1 - reactor support; 2 - reactor core; 3 - reactor vessel; 4 - reactor coolant pump; 5 - upper radiation shield; 6 - rotating plug; 7 - above core structure; 8 - intermediate heat exchanger; 9 - in-vessel radiation shield; 10 - core diagrid; 11 - pressure chamber

TABLE 1. COMPARATIVE ANALYSES OF TOP AND BOTTOM OF THE REACTOR VESSEL SUPPORTS

Items for comparison	Vessel with top support	Vessel with bottom support
Compensation of equipment thermal extention	It is necessary to compensate thermal extention of the primary pressure pipes. Lesser thermal extention of secondary circuit pipes	Problem of thermal extention of the primary pressure pipes, is easier to solve
Upper closure	More complicated structures, that is as load-bearing, as well as sealing the same time	More simple structure with function of only biological shielding, it is not load-bearing and sealing, located at a low temperature area and cooled by the reactor cavity ventilation system
Unit of the reactor vessel joint	Considerable bending stresses are in place from the vessel welding to the upper closure. Special measures are needed to provide acceptable working conditions for this unit (cooling, sharping, insulating)	Vessel support unit is located at the area with low temperatures. Load stresses are low. Support unit is designed in such a manner so that weight of internals does not give rise to additional load on the main and safety vessels
Main and safety vessels	More simple shape of the vessels. Accessibility for inspection from the outer vessel	More complicated shape of the vessels. Main and safety vessels have two joints (at the support unit and of the conical roof) bringing necessity to have bellows compensators

2. LMFRs OPERATION EXPERIENCE

Up to now, a considerable experience during LMFRs operation has been gained in Russia. This allows to give a judgment concerning correctness of design and conceptual decisions adopted during the LMFR development. Experimental reactors like BR-10 and BOR-60, prototype reactor BN-350 and demonstration reactor BN-600 are under operation. Russia has accumulated more than 100 reactor-years of LMFRs operating experience. In general, this experience is estimated as positive. Failures which occurred were not fundamental and were caused by exceptional unsuccessful design decisions. Of high interest is the analysis of BN-350 and BN-600 operating experience, which will lead to the main decisions concerning the for development of next generation LMFRs .

2.1. Reactor BN-350

The BN350 reactor was in operation during 1973. The heat generated by the reactor is used for electricity production and sea water desalination.

During the operation period the reactor had various power levels (Fig. 3.) The average load factor in respect to established power levels was 85%. The main factor limiting the reactor power level was failures of the steam generators. At the same time, the reactor equipment demonstrated stable fault-free operation. In the first years of operation the reactor core posed a certain problem due to excessive heat rating of fuel pins and a considerable shape deformation of the core units. The core modernization and the changing of structural materials solved this problem. Further increase of fuel burn-up is planned through utilization of the ferritic steel EP-450 for ducts and of improved austenitic steel in cold-worked state for claddings. Simultaneously, with the improvement of the core fuel operating performance, modifications were introduced into the design of the absorber rods and guide sleeves by the use of radiation-resistant structural materials.

There are twelwe absorber rods in the reactor: two control rods, three shutdown rods and seven reactivity compensating rods. For the whole period of the CRDMs operation they have been functioning without any significant abnormalities. In 1979 and 1980 there were difficulties associated with the disconnection of the absorber rods from their drive lines before the reactor refuelling. The analysis revealed a potential for seizing the CRDM moving parts in the guide tubes in a sodium-to-gas transition zone by solidified sodium if the coolant temperature was diminishing. To eliminate such events a sodium temperature under refuelling was set in the range of 230-240°C. After that minor modification there were no problems with
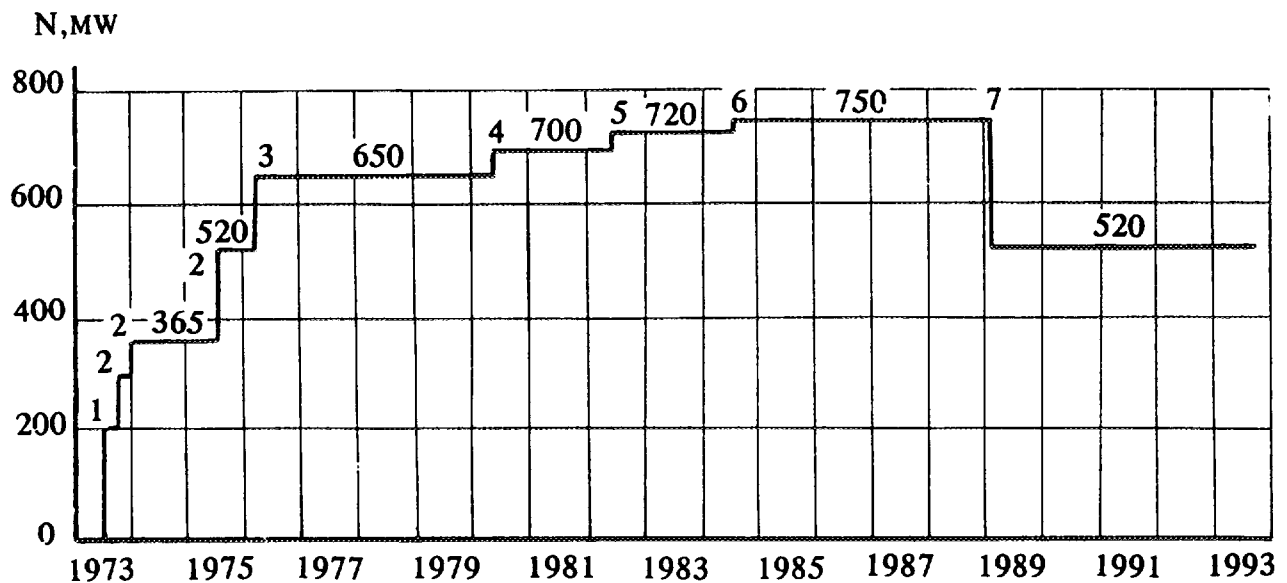
Fig. 3. BN-350 specified power variation during operation life
1. *July 16, 1973 - power start-up to 200 MW (th)*
2. *Power building-up as SGs were being repared and brought into operation*
3. *March 1976 - field-tube evaporators repair completion*
4. *May 1980 - first Czech-produced SG "Nadjozhnost" start up*
5. *June 1982 - second SG "Nadjozhnost" start up*
6. *July 1984 - specified power level increase on the basis of feedback*
7. *January 1989 - power reduction due to SG :Nadjozhnost" failure*

21

the absorber rods functioning. During the whole period of the reactor operation (1973-1995) 56 planned refuelling cycles have been fulfilled. The time spent for one cycle of FA replacement was one hour on average. The handling system provided the change of FAs and other core components without problems. The total of loading/unloading cycles for such mechanisms as ramp and fuel transfer machine amounts was about 3400. Due to failure of the spent fuel storage drum (1976) a special lead-shielded flask was designed and manufactured for spent FAs transportation from the transfer cell to the cleaning facility.

During the operation it was noticed that increased forces were required to rotate the shield plugs. Probable causes of the event could be sodium vapor condensation in gaps and non-uniform heating of the lead-hismuth seals. The fuel transfer machine, the ramp, the handling mechanisms of the handling cell and cleaning facility were run without substantial failure. Minor disturbances were eliminated through the replacement and modernization of individual items. Horizontal tube-and shell IHXs with three modules connected in series and made of U-shaped tubes are used in BN-350. Measurements of temperature and stresses in various parts of the IHX were carried out during the reactor operation. On this basis the requirements were formulated to limit the rate of the IHX heating-up in steps of 10% specified power with a delay of 5-10h on each step. By 1995, the IHXs had been in operation more than 160×10^3 h without any disturbances and failures.

The major source of the reactor radiological impact to the environment is gaseous discharge from the air cooling system of the equipment and from the reactor building through the vent stack. Resulting from improvements in the fuel design and associated reduction of failed fuel pins number in the core to single events, radioactivity of the plant discharged to the atmosphere was determined by the radiation-induced activation of air in the reactor cavity cooling system. Daily release of gaseous nuclides was 0.55-0.74 TBq while that for aerosols 0.9 E-6 TBq. Lasting observations on radioactivity of the flora and fauna, and on radiological conditions in the exclusion area and local populated territory showed that those characteristics are determined by natural and man-induced radiation sources and correspond to the radiation background.

The BN-350 reactor was developed on the basis of regulatory documents of 1960s and therefore it meets update requirements for safety not to a full extent. BN-350 design lifetime (20 years) expired in July 1993. The possibility of BN-350 further operation depends primarily on the reactor safety enhancement activity and substantiation of the reactor lifetime prolongation taking into account real operating conditions of the reactor and the main components. In 1992 the Design Institutions together with the Plant Operator completed a review on the reactor conformity to the requirements of safety regulatory documents. Resulting from the analysis a list of deviations was compiled, and indispensable measures were identified for the reactor upgrading, part of which has been already realized. The most important among those measures being performed in the last time in the framework of the reactor safety enhancement are the following: (1) Reaching seismic resistance required. Analysis of consequences of seismic impact on the reactor building structures, equipment and pipelines showed that the existing power supply system, the feed water system and the system for reliable service water supply to components of the systems important to safety would be destructed completely or partially under seismic impact of magnitude 6 (MSK scale) - Design Basis Earthquake accepted for BN-350. Taking into account this fact, the design was developed to arrange safety systems equipment in seismic resistant part of the reactor building for ensuring residual heat removal under seismic conditions. The design was developed and has been currently realized providing seismic protection for the reactor by triggering the shutdown system on signals from seismic sensors; (2) Analysis of the influence on the reactor

of explosive industrial facilities situated near the site has been completed. Corresponding measures are developed to ensure the reactor safety; (3) Work is currently performed for the control and protection system upgrading through the renovation of its components and optimization of transient algorithms; (4) A system to remove core residual heat in case of coolant level lowering below the reactor vessel outlet nozzles has been installed and tested. The electric drives of the system's sodium valves are powered from a reliable power source; (5) An independent feedwater system for the SGs has been put into operation; (6) An air-cooling system of the module SGs "Nadyozhnost" 1 and 2 has been put into operation. The electric drives of the systems valves are powered from a reliable power sources, and (7) An overpressure system for protection of the main and safety vessels has been installed.

The program for bringing BN-350 to compliance with the regulatory requirements is planned to be completed in 1997.

Lifetime extension implies performing analysis of the real state of the main systems and components, determination of their residual lifetime, finding items with expired lifetime and their replacement and validation of the BN-350 lifetime extension. Preliminary findings resulting from this activity proved that such components, as the reactor and safety vessels, reactor support system, in-vessel structures, valves still have a considerable lifetime reserve. The reactor operating data confirmed that its technical and economic performance during the full operation period is quite satisfactory. Taking into account all these facts a technical decision was adopted to prolong of another year the BN-350 operation period. A decision on possibility to continue BN-350 operation will be adopted on results of the ongoing work.

2.2. Reactor BN-600

The average load factor for the whole period of the reactor operation (since 1980, till January 1, 1995) equals ~73% and is up to the standards of serial domestic and foreign LWRs. The maximum load factor of 83,5% was reached in 1992. As to reliability indicators, BN-600 enters the ranking of the best Russian NPPs. The power diagram for BN-600 operation started from the data of its startup which is given in Fig. 4. Up to January 1, 1995, BN-600 total on-power operation time amounted to 102,139h. Cost of electricity being produced by the power unit is approx. 20% lower than that for fossil fuel plants operating in the same region. The main contributors to the nuclear island unavailability were: primary and secondary pumps and fuel pins and SGs faults. The core of the first type was in operation at rather high performance. However, as the initial operating experience showed, they turned out to be very strong. Already during the first fuel cycles loss-of-integrity events of the claddings started, increased swelling of FA's ducts and of some of absorber rods items were observed, as well as loss of ductility of the absorber rod guide sleeve material. Due to the fuel pins failures in 1983-1987 the reactor was shutdown six times for unplanned refuelling. The core design improvement became an important measure to provide operating reliability and safety of the reactor. The investigations of the failed fuel revealed stress-induced corrosion of the annealed austenitic steel claddings as one of the main cause of their early failures. The claddings were damaged mainly in the peripheral region of the core. It was due to the most unfavorable operating conditions for peripheral FAs. Because of their reshuffling and rotation in the course of operation, the furl pins linear heat rating and claddings temperature rose up to 54 kW/m and 710 °C respectively at the end of fuel cycle.

In the advanced core of M-design (the first core modification) the following decisions were realized to improve conditions for the fuel operation:

-       core height was increase from 75 to 100 cm, which decreased the fuel pins maximum linear heat rating down to 47.2kW/m;
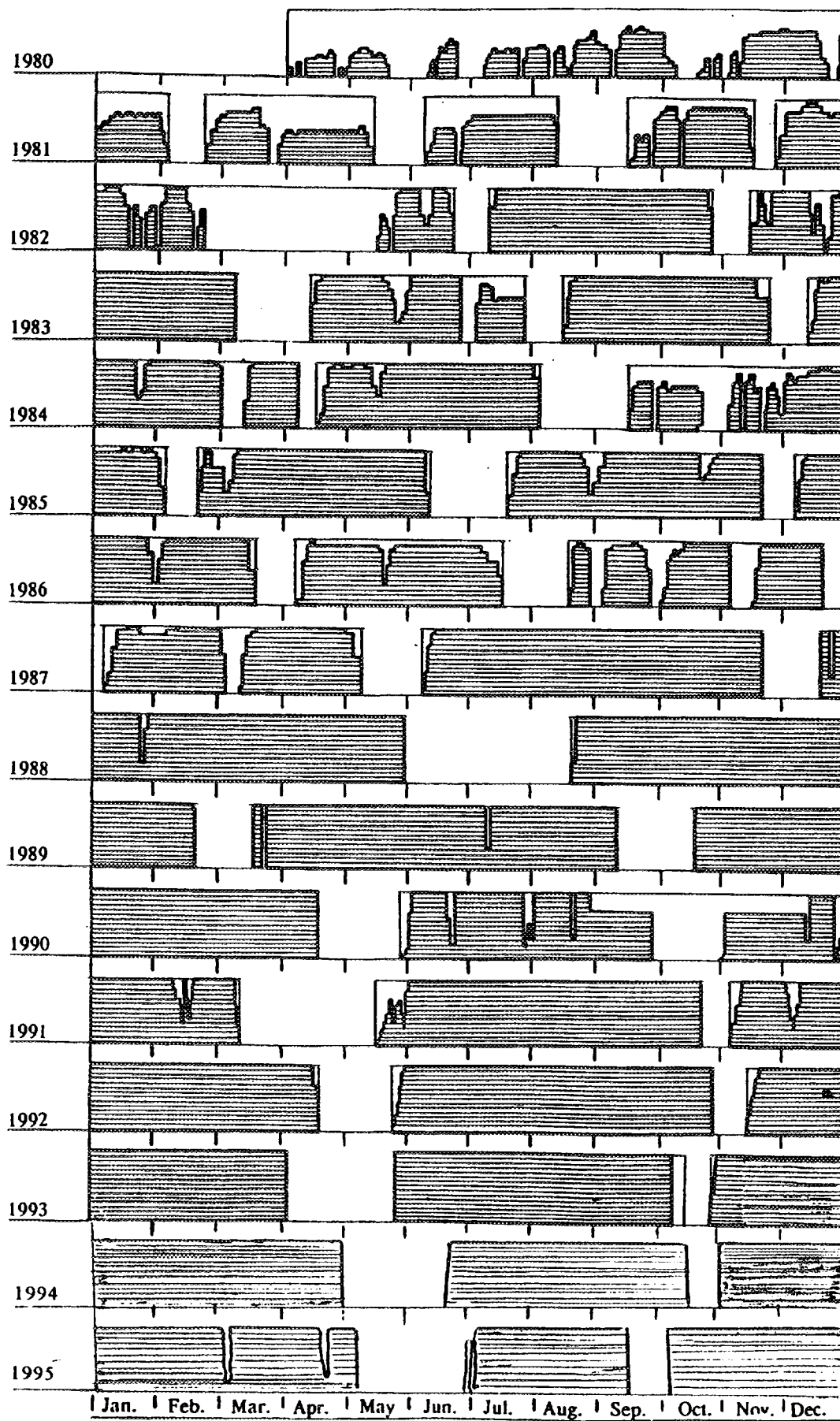
Fig. 4. BN-600 power unit operation power histogram

- reshuffling and rotation of the FAs were eliminated;
- swelling-proof cold-worked austenitic steel was used for the claddings and for duct of the FAs.

By the end of 1987 the reactor core was completely assembled by advanced FAs. Loss-of-cladding integrity events virtually terminated resulting in substantial reduction in fission products activity in the reactor gas plenum. Cesium nuclides concentration rise in the primary system terminated as well.

In 1990-1992, the reactor core has been changed over to the advanced M1-design (the second core modification). Ferritic steel was used in the new design for ducts and boron-system terminated as well.

In 1990-1992, the reactor core has been changed over to the advanced M1-design (the second core modification). Ferritic steel was used in the new design for ducts and boron-modified cold-worked austenitic steel for claddings. Fuel burn-up in the core has reached 10% ha the fuel cycle length 160 full-power days. Development of advanced radiation-resistant steel is the main problem now for the attainment of higher fuel burnup. Considering the present status of this problem, the fuel burnup of 12% ha (90 dpa) is believed to be quite realistic for the advanced reactor core. The core reactivity margin has to be increased through expansion of the medium fuel-enrichment zone at the expense of the adjacent FAs in the low enrichment zone. If the medium enrichment core zone would be expanded to the limits of one FAs row, the heat rating of fuel pins will remain at admissible level - approx. 48 kW/m. For the majority of the CRDMs their total operating period since the date of first criticality until January 1, 1995, amounts to 101,942 h, the time in sodium being 130,949 h. In the period of adjustment-trial tests a certain structural disadvantage in the CRDM design was revealed which caused their damage each time when a CRDM movable member reached one of its extreme position and it was not noticed by the operator. After minor modification the CRDMs have been operating reliable, without any disturbances. The design service life of the reactor coolant pumps was specified at 50,000 h. Problems with the pumps that resulted in unplanned energy losses occurred in 1981-1985 and were caused by unstable operation of the pumps speed control system. Impacts from the pump electric motor caused failures in the pump-electric motor coupling, increased vibration and fatigue cracks in the pump shafts. The use of advanced shaft and coupling, as well as pumps change over to a steady mode of operation after attaining the present level of reactor power, have allowed to exclude since 1985, any failures of the reactor coolant pumps. The modular SGs demonstrated high operating robustness. During the entire period of the SGs operation there were twelve water/steam-into-sodium leaks, half of which occurred in the first year of operation and were caused by propagation of latent manufacture defects. The leaks were mainly in the superheater modules (six events) and in the reheaters (five events), while in the evaporator there was only one leak (Table II).

The intermediate shell-and-tube sodium/sodium heat exchangers having tubes with expansion bends are in operation for 15 years without faults and troubles. Preparations are presently carrying out for one of the IHXs removal from the reactor and for its inspection to determine the residual life before their specified service life (20 years) expires.

Satisfactory results have been obtained on the reactor refuelling mechanisms operating experience. The adopted refuelling concept, relying on the utilization of rather simple mechanisms for servicing each individual part of the fuel transportation route, has been proved as quite reliable

TABLE II. BN-600 SG LOSS-OF-TUBES INTEGRITY EVENTS

Date:	24.06 1980	04.07 1980	24.08 1980	08.09 1980	20.10 1980	09.06 1981	19.01 1982	22.08 1983	06.11 1984	10.11 1984	24.02 1985	24.01 1991
Leak Place	RH	SH	RH	SH	SH	RH	SH	SH	E	RH	SH	RH
Leak Size	L	L	S	S	S	S	L	S	S	S	S	S

Abbreviations: RH - reheater; SH - superheater; E - evaporator; L - large leak; S - small leak

and safe. Total operating cycles in terms of double strokes for the in-reactor refueling mechanism amounts to 36,100 for the ramps - 9795 and for the ex-reactor fuel transfer mechanism - 29,200, that exceeds the respective design values.

During fifteen years of the BN-600 reactor operation there were twentyeight small and two large sodium leaks. The largest leakage of secondary sodium happened on May 6, 1994, in a drain pipe of an IHX during repair work on changing of an isolating valve when the reactor was shutdown. The sodium leak was accompanied by its burning and some equipment damage in the adjacent area. Approximately 1.2 - 1.3 m^3 of sodium were leaked. But only several tens of kilograms were burned. The remaining sodium was retained in the smothering catch pan system, covered with extinguishing powder.

3. CONCLUSION

The experience on LMFRs operation accumulated in Russia, proved their high level safety and reliability. During operation there were no substantial problems concerning reactor physics, operation of the equipment, sodium technology and materials which could cause diffciculties to further improvement. On basis of the experience it is practicable to create a next generation of LMFRs with broad power range and various functions.

BN-600 POWER UNIT 15-YEAR OPERATING EXPERIENCE

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XA9643086

Abstract

Comprehensive experience has been gained with the operating fast reactor BN-600 with a power out put of 600 MWe. This paper includes important performance results and gives also an overview of the experience gained from BN-600 NPP commercial operation during 15 years.

1. CORE AND BLANKET

The uranium fueled core was designed to consist of two 21% and 33% uranium-235 enriched zones, to obtain a 9.7% ha peak burnup. However, operating experience which has been accumulated by the time of the BN-600 commissioning led to the limited burnup of 7.3% ha and to 100 efd interval between the refuellings to be carried out three times a year. In spite of decreased burnup levels considerable number of fuel failures was still observed by the end of each operating cycle. In order to improve fuel performance the core was modified in 1987, as follows: a) three uranium enriched zones (17-21-26% enrichment, 8.3% peak burnup) were introduced; b) fuel subassemblies rotation and reshuffling were terminated, and c) core part height was increased from 75 cm to 100 cm, thereby reducing the linear rating from 51 kW/m to 47 kW/m. The reloads were two times a year after 165 efd refuelling interval. With this new core no fuel failures actually occurred and potential for higher burnup could be further realized. In 1993, the second modification of the core took place to allow the nominal burnup to be reached. The advanced structural materials[1] have been used. With these new materials a 10% ha peak burnup was obtained. This core was qualified as standard. Nowadays the possibilities are in hand to provide for the third modification of the core with more than 11% ha burnup. This would allow to increase intervals between refuelling which is important in view of the local conditions necessitating co-generation of heat for central heating in wintertime. Work on increasing burnup levels involved in-reactor testing of series of experimental subassemblies there were 5552 experimental subassemblies (more than 70,000 fuel pins). The uranium-plutonium vibracompacted fuel testing was finished successfully and standard MOX fuel testing is continued. A much longer lifetime of control rod guide tubes (from 200 to 640 efd), due to advanced structural materials, is another important result of a 15-year operation period.

2. REACTOR AND PRIMARY CIRCUIT SYSTEM

The BN-600 reactor is of the pool type, (Fig. 1.) i.e. besides the core and in-reactor components the reactor vessel encloses the entire primary heat transfer system comprising primary sodium pumps, sodium-sodium intermediate heat exchangers and pipe manifold. *The primary sodium pumps* are characterized by successful operation. At the beginning some failures of gear couplings took place. The failures were caused by coincidence of shaft

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[1] See paper F.M. Mitenkov "LMFRs design and operation experience"
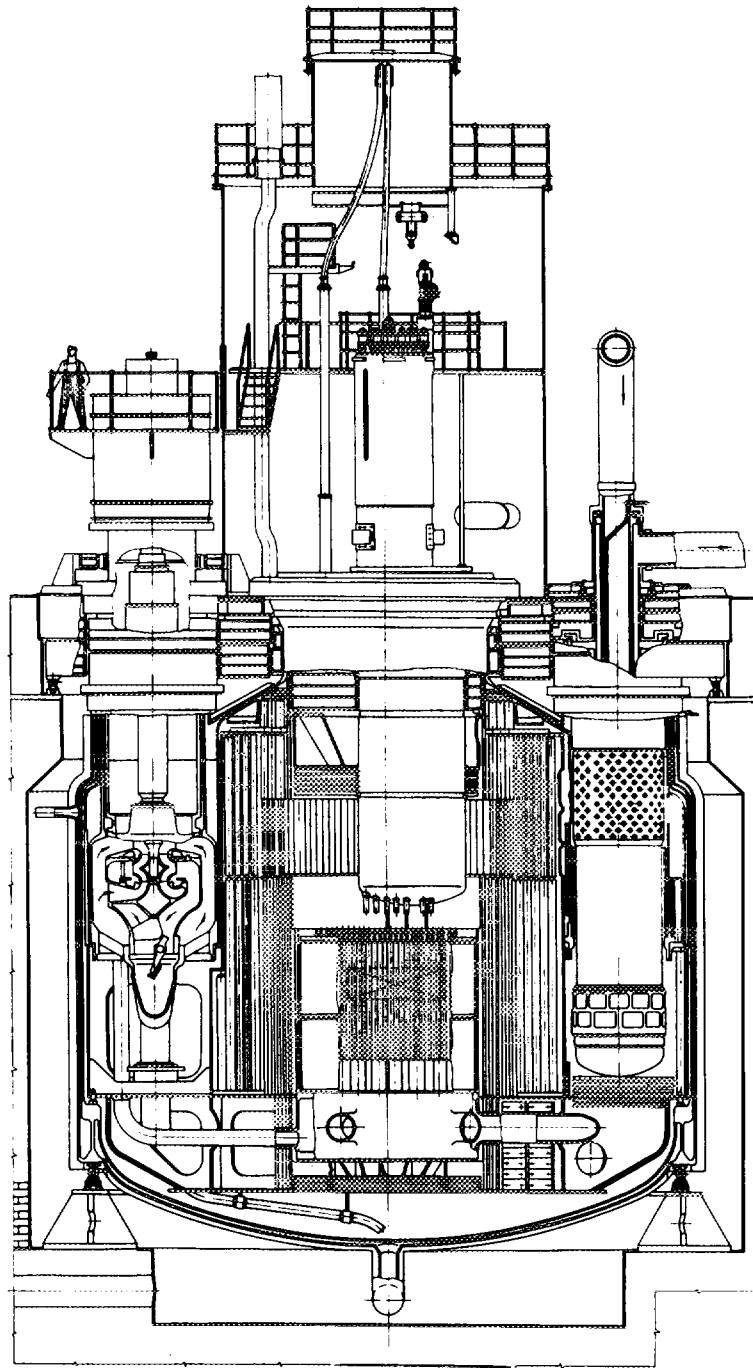
Fig. 1. BN-600 reactor

resonance frequencies and frequencies of torsional vibrations. After the cause had been identified and rotational frequency had been adjusted away from the resonance no failures occurred. The work on improving pump performance resulted in longer lifetime of the main pump components, the most important achievement being extension of impeller lifetime up to 50,000 running hours. Valuable experience was accumulated on replacement of removable pump components. There were two interventions on each primary pump to replace impellers after expiring their lifetime. The *intermediate heat exchangers* have been in trouble-free operation.

3. STEAM GENERATORS AND SECONDARY CIRCUIT SYSTEM

The secondary heat transfer system consists of three loops, and each of them consists of a steam generator, a secondary sodium pump and pipe manifold. *A steam generator* (Fig. 2) is a set of modules of three types, with evaporating, superheating and reheating modules (8 modules of each SG). Three modules: evaporator, superheater and reheater represent the SG section (in total a steam generator has 8 sections) which can be isolated using gate valves both on sodium and water-steam side. In the case of a water-into-sodium leak in one of the modules of any section, the latter to be isolated using gate valves thereby permitting to proceed the steam generator operation practically without reduction in NPP overall capacity. The operating experience had validated the flexibility of module type steam generator concept. Whereas 12 water-into-sodium leaks occurred, the electrical generation loss attributable to this cause was only 0.3%. Valuable work was done on evaporator lifetime extension from 50,000 to 105,000 running hours necessitating only one replacement of each evaporator module within the reactor lifetime instead of three replacements. The longer evaporator lifetime was based on the results from the extensive research programme and was contributed by improving water chemistry, decreased rate of the transient and emergency conditions against the design value, periodical re-agent cleaning, and water washing for friable deposit removal. Nowadays the planned replacement of the evaporators are under way. *The secondary sodium pumps* have been essentially in trouble-free operation. On the basis of the research programme their lifetime was extended up to 105,000 running hours.

4. TURBINES AND WATER-STEAM CIRCUIT SYSTEM

The turbines installed in the BN-600 power unit are of conventional design. They are characterized by successful operation. *The drain lines and valves* have been major contributors to the incident rate for latest years. Now just their performance is considered to be critical for the entire power unit performance.

5. ELECTRICAL EQUIPMENT

The power unit alternators are of conventional design. During the latest years the stator cooling system distillate leaks have been observed actually each operating cycle. Several times these led to unplanned heat transfer loop trips. The cause is an improper design of the valves seals.

6. INCIDENTS

None of the occurred incidents had impact on public and environment. All the incidents were beyond the OFF SITE IMPACT parameter on International Nuclear Event Scale, i.e. negligible for safety. The most serious incident (assessed as level 1 by the other parameters: *on site impact* and *in-depth protection*) was a sodium leak on the auxiliary primary sodium purification system pipeline 48 mm diameter occurred on 07.10.93 causing insignificant radioactive discharges to atmosphere which was equivalent to 0.001 buildup of natural radiation background on the boundary closest to the plant.

7. BN-600 PERFORMANCE ESTIMATION

The comparative estimation of NPP performance is a many-dimensional statistical problem, i.e. a cluster of indicators of various power units should be estimated against time distribution since the point estimation for one year or an individual plant is not representative

Fig. 2. BN-600 steam generator

due to discrepancy of annual individual indicator distribution estimates of the same plant and due to different operation schedules of various plants. It is also evident that the main indicator of good plant performance should be a total steady operation criterion rather than individual annual parameter achievements. Since statistical distribution of any indicator bears no symmetry, a sample median should be the main statistical estimate of an indicator. As regards an individual plant the median is a indicator value maintained throughout half period of the plant history. By analogy as regards a cluster of plants the median is an indicator value attributable to half of the plants. The comparison between the BN-600 NPP and world main NPPs performance indicators (Table 1.) has shown BN-600 to be in the top half of the world nuclear power units.

TABLE I. MAIN WORLDWIDE NPP PERFORMANCE INDICATORS

Indicator	World NPPs	BN-600
1. Load factor, %	73.0	73.2
2. Unplanned production losses, %	3.9	2.7
3. Reactor scrams per 7000 hours	1.1	0.0
4. Collective personnel dose, manSv	2.0	1.0
5. Low active solid waste, m^3	100.0	62.0

10. CONCLUSION

1.   The 15-year operation period have demonstrated reliability and safety of the pool sodium-cooled fast reactor power unit BN-600 with the module steam generators.

2.   The main performance indicators place the BN600 in the top half of the world power units that should be characterized as achievement of the commercial NPP performance level. These indicators are better than similar indicators of other fast reactors in the world.

3.   The BN-600 operating experience is essential for designing next generations of fast reactors, in the Russian Federation.

XA9643087

RECENT PROGRESS IN CONCEPTUAL DESIGN OF LMFR IN CHINA

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Abstract

The Chinese Experimental Fast Reactor. (CEFR-25) with the thermal power 65MW and electric power 25MW is the first step of the FBR development in China. The aims of this project are as following :
  • As a prototype to accumulate the experiencesof design, construction and operation of a fast reactor;
  • as an irradiation facility to develop fuels and materials with high breeding properties and burn up rate; and
  • as a test reactor core, envisaged, to test the fast reactor core with the fuel Ac-contained.
        After three years preparation for the CEFR design including the development, collection and reviewing of about 50 computer codes and the decision of the main technical selections and of design boundary conditions, from 1990 to July 1992, the conceptual design of the CEFR-25 has been completed. The confirmation and some optimization of the conceptualdesign have been carried out from October 1992 to the end of 1993.
        Based on theconceptual design and related optimization, the mainfeatures and charactiristics of the CEFR-25 has been given in this paper.
        Spent almost whole year 1994 for the input preparation the CEFR-25 technical design has been started in the early of the year 1995.
        The input for the technical design mainly includes:
  • design criteria
  • input parameters
  • technical selections   confirmed in conceptual design
  • design requirements
  • etc.
which are brifly described in this paper.
        For the design requirements to the CEFR-25, of the most important is that the reactor should have passive safety properties, in other words, duringany credible transient incident, for instance, ULOF, ULOHS and UTOP, etc. bythe negtive feedback of reactivity the reactor will enter into and be keepingat safety condition without meeds of any personal interference. The residual heat will be removed away by natual convection and natual circulation.
        Finally, the paper gives the general descriptions about the researchand development status for this reactor.

1. Introduction

China has become one of the countries with a fast growth of the national economy development. The electricity capacity increase and totalcapacity reached in the years 1986–1992, 1993 and 1994 could more or less reflectthis growth-up, which are shown in Table 1.

Tsble 1 Electricity capacity increase and
reached total capacity

Years	1986–1992 7yesrs	1993	1994
Electricity capacity increase(GWe)	~70	14. 38	15. 0
Reached total capacity(GWe)	~165	180	195

But China is a Country with a vast territory and large population. The electricity capacity per capita is now only 0.17KW, much lower still than that in the most countries of the world. For improving the living standards of the people, the energy production should be raised year by year. Along the basic policy of reforming the systems and "Opening the door" in China, it has been estimated that at the middle of next century, the total electricity capacity would be raised to 1200GWe, near 1KW per capita

The coal and hydropower will be still the predominant energy resources for the electricity generation atthatages. But they are not well-distributed inChina. About 60% of the coal resources are concentrated in morth and north -west China and about 70% of hydropower potential are scattered in south-west China, on the contrury, the large amount of population and industryare concentrated in the southeast and coastal of China where the energyresources are shortage. Even right mow for transportation of energy resources almost 48% of the railway capacity and 25% of the motor way capacity are needed which results in unreasonably high price of the energy/

From an environment protection and chemical industry resources preservation point of view the necessity of developing nuclear power should beconsidered. In Chinese nuclear power program, PWR has been seleeted as a main type of reactor for the first stage. Based on it the lack of uranium resources will be occuring if the total nuclear capacity in large scale is needed, So the FBRs will play a very important role for meeting the nuclear program inlarge scale to replace coal fired power plants as more as possible.

2. CEFR-25 project

2.1 Strategy study

The strategy study of the fast breeder reactor technology development in China has been conducted five years ago in the framwork of the NationalHigh Technology Program led by the National Science and Technology Commission. Based on this study, it was suggested that Chinese FBR long term development program would be divided into three steps:

a. experimental fast reactor (CEFR-25) with the thermal power of 65MW, matched by a turbine generator of 25MWe. It is intended to have its first criticality in the year 2000;

b. modulized fast breeder reactor plant (MFBR) with a modular size of 150- 300 MWt which will be optimized at design stage. The capacity of the MFBR plant could be 600, 900, 1200 MWe or more. The first MFBR plant would bein operetion in the year 2010-2015;

c. large fast breeder reactor plant (LFBR) with high breeding properties. The first LFBR would be in operation round the year 2050, and then to be deployed, based on its design and technology. improvement and plutonium accumulation by the first generation thermal reactors.

If 40-50GWe could be reached by thermal reactors in the year 2010 -2015, the target of 240 GWe nuclear electricity capacity will be realized only by FBRs in the middle of next eentury, which is about 20% contributionto the national total electricity generation.

2.2 Aims of CEFR-25

As the first step of Chinese FBR development, the aims of the CEFR-25 are as following:

a. As a prototype of fast reactors to accumulate the experiences of design, construction and operation of a fast reactor with a suitable electricity generation systems;

b. As an irradiation facility to develop fuels and materials with high breeding properties and burn-up rate; and

c. As a test reactor core, envisaged, to test the fast reactor corewith the fuel Ac-contained.

2.3 Technical selections and design boundary

After three years (1988-1990) preparation for the design of CEFR-25, including the development, collection and reviewing of about 50 computer codes and the decision of main technical selections and of designboundary conditions, from the year 1990 the conceptual desigm of CEFR-25 was started.

Having the clear aims of Chinese FBR development, the technical selections and some parameters should have its continuity in some extents for three and some parameters should have its continuity in some extents for three steps of the Chinese FBR development program mentioned above, with the purpose to facilitate the transmission of development step. It is possible now to take this strategy from experimental fast reactor up to commercial fast reactor, because much experiences have been provided by some predecessor countries of fast reactors. Followings ars the technical selections but mainly limited to what which would be adopted for future fast reactors:

a. The pool type of primary loop arrangement has been choised for Chinese FBRs. Even though pool type or loop type which represents two main selections according to the FBR experiences in the world has its own advantages and disadvantages, in pool type a accident loss of coolant is practically excluded and in case of loss of grid or pump failure the sodium with large quantity in the pool offers a big heat sink. Also almost all design of commercial FBR in the world are using pool type.

b. Plutonium Uranium mixed dioxide $(Pu, U)O_2$ is selected as the fuel of CEFR- 25, considering its better stability behavior under heat and irradiation conditions and its feasibility in China. 316 titanium modified stainless steel with 20% cold worked as cladding material and core structure material, which was relatively more studied by corrosion, irradiation and manufacturing in China. Some new fuels with high breeding properties, for instance, (Pu, U, Zr) Alloy fuel $(Pu, U)N$ fuel will be irradiated and developed in the CEFR-25.

c. As one of safety targets, the CEFR-25 and its followings should be designed with passive safety properties, it means that the reactor has strong negative feedback of reactivity with which the reactor could be shut down by itself and kept at safe conditon under any credible transient incidents, for example ULOF, ULOHS, UTOP etc. The residual heat will be removed by natural convection and natural circulation. It is selected that the Na-Na heat exchanger of the residual heat removal system will be put in the primary pool.

d. The primary storage of spent fuel subassembly in the periphery of the reflector is selected in stead of direct storage in the water pool. This selection will decrease the risk of fuel handling accident.

e. The sodium outlet temperature from core will be 530℃, maximum linear power of the fuel pin will be 430 W/cm.

2.4 Siting

The site of the CEFR-25 will be located in the China Institute of Atomic Energy about south west 40km far from Beijing City .

The primary exploration of the site has been completed including geological prospecting, hydrologic survey, local climate and population density and distribution investigation, for external events, including following investigetions: floods earthquakes, site surfacecollapse, surfacc faulting tornadoes, tropical cyclones, aircraft crashes, chemicalexplosions, etc. •••

After the possible earthquakes analysis, it is obtained that the maximam design operation and safety shut down earthquake dynamic parameters on the base rock are 0.107g and 0.214g respectively.

Due to no special site rule for an experimental fast reactor up to now, the site rule for the nuclear power reactors will be used for the CEFR-25 as the decision by the China National Nuclear Safety Administration.

2.5 Feasibility study

The preliminary feasibility study report which is needed for the CEFR-25 as a complex construction project according to the regulation promulgatedby the National Planning Commission, is composed of General report, Conceptual design of the Chinese Experimental Fast Reactor, Site evaluation, Safety and environment study and Economy study. This report has been approved by the Authorities in March 1994.

The feasibility study report including also 5 sub-reports, but more thorough than preliminary feasibility study has been completed, in whichthe conceptual design has been approved by the Authorities in January 1994.

3. Conceptual design

From the year 1990 to July 1992, the conceptual design of the CEFR-25 has been completed. The confirmation and some optimization of the conceptual design have been carried out from october 1992 to the year 1993.

CEFR-25 is a pool type sodium cooled experimental fast reactor withthe power 65MWt and matched with a 25MWe turbine generator. Plutonium and uranium mixed dioxide is selected for fuel, 316(Ti)ss as cladding material andcorestructure material. Two maim mechanical pumps withfourintermediate heatexchangers are designed for primary loop. Two circuits for secondary loop andone water-steam circuits with two evaporator and two superheater are composedof the third loop, but only one turbine generator are selected.

Two independent, same and passive residual heat removal systems are designed which capacity is about 0.5 MW for each. the each system is composed of one heat exchanger immersed in the bot pool and two Na-air coolerwith their high stacks. The air-doors are opened enough at opreation states in order to be sure to remove the residual heat necessary without any operation to the air-doors.

A guard vessel is used, when a leakage of sodium from main vessel has happened it will keep the core not only immersed in primary sodium butstill has enough natural convectisn capability.

In the preliminary conceptual design a top supported concept of doubl vessel was designed. As understand the bottom supported is favourable in the earthquake protection. Unfortunately the CEFR-25 site will be located in the 7thgrade region under 12 grades classification system. So the new selection could be changed to bottom supported.

Double rotating plugs with a straight moving handling machine have been selected for subassembly charge and discharge among the core, in-core storage and fixed position which is connected with the lowestend of theincline lift machine. A transfer machine could reach the highest end of theincline lift machine, So the new or spent fuel subassembly could be enterringin or leaving out of reactor vessel through the port right above the highest endof the incline lift machine.

After washing the spent fuel subassemblies will stored secondarily in water pool which could accept fuel subassemblies of 11.5 cores.

The purification system for primary sodium is arranged outside of reactor vessel and connected between them by piping with two quick -close valves and asiphonage destroy device is used to limit the quantity of the leakage if a leakage of sodium has happencd in the purification system.

A rather large storage tank of primary sodium has been designed in order to accept the sodium drained from main vessel corresponding to the sodium level decreased up to the top of the reactor core. This is an emergency measure for some unexpected necessities to look at the top of reactor core, i. e. the top of fuel or control rod subassemblies.

A steel made cover shade will be provided on the top of largerotating plug in order to protect the sensitive mechanisms on the rotating plugs and toplay a more barrier for the radioactive gas material leaked.
A reenforced containment would be equiped for this reactor, which will play the role of the last barrier for the radioactive materialsleaked, protect the reactor when the credible aircraft dropped down to thereactor building and give the habidants a safe sense.
The conceptual scheme drawihgs for the fuelsubassembly, reactor core arrangement, reactor block and water-steam loop are shown in Figs1-5, the main parameters of the reactor are given in Table 2.

1. Handle head
2. Hexagonal
3. Fuel Bundle
4. Grid
5. Foot
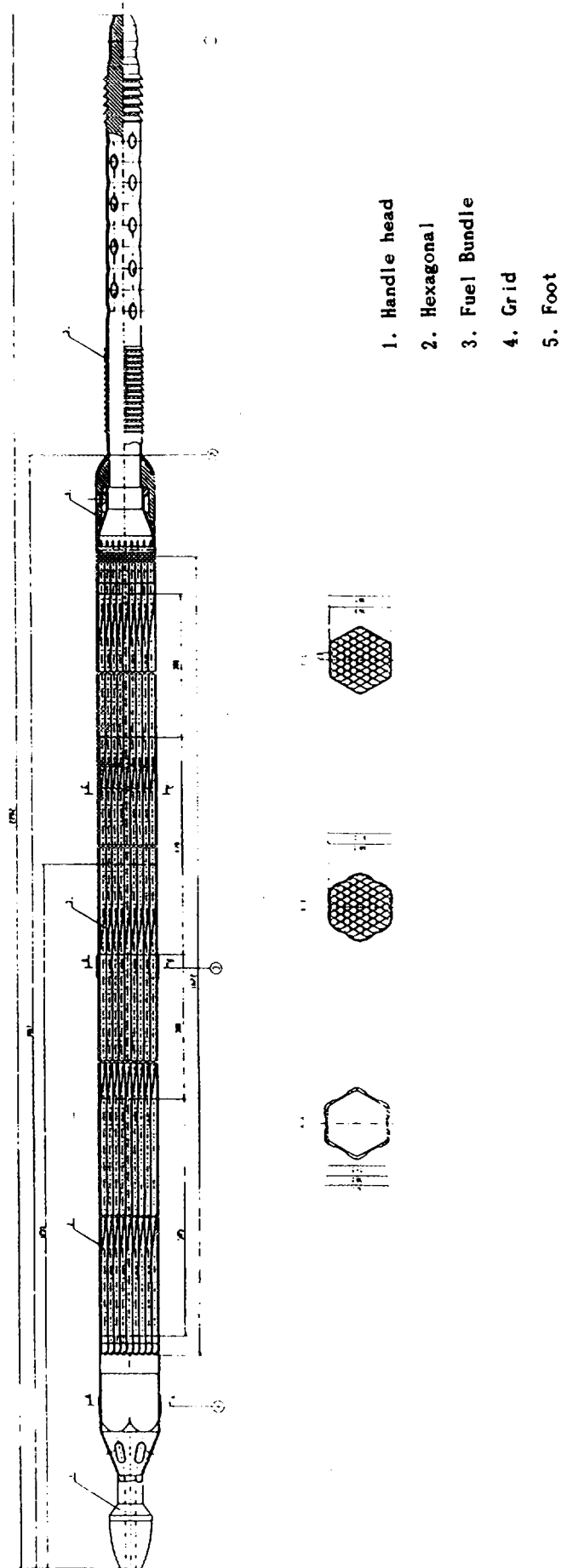
FIG. 1. Fuel Subassembly.
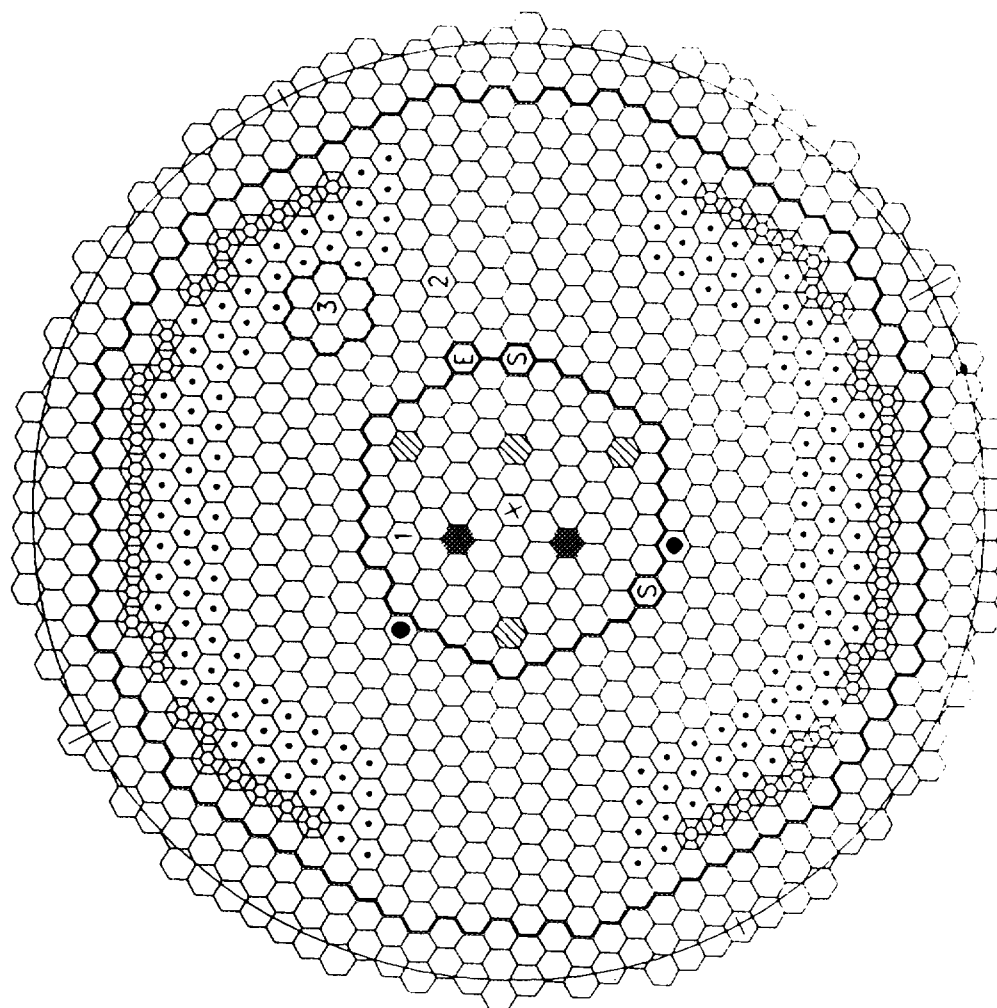
Fuel Subassembly	82
Reflector Subassembly	379
Sheilding Subassemply	152
Spent fuel Storage	54
Safety Rod	2
Compensation Rod	4
Regulation Rod	2
Neutron Source	2
Irradiation Facility	1
Irradiation Loop	7

FIG. 2. CEFR First core.
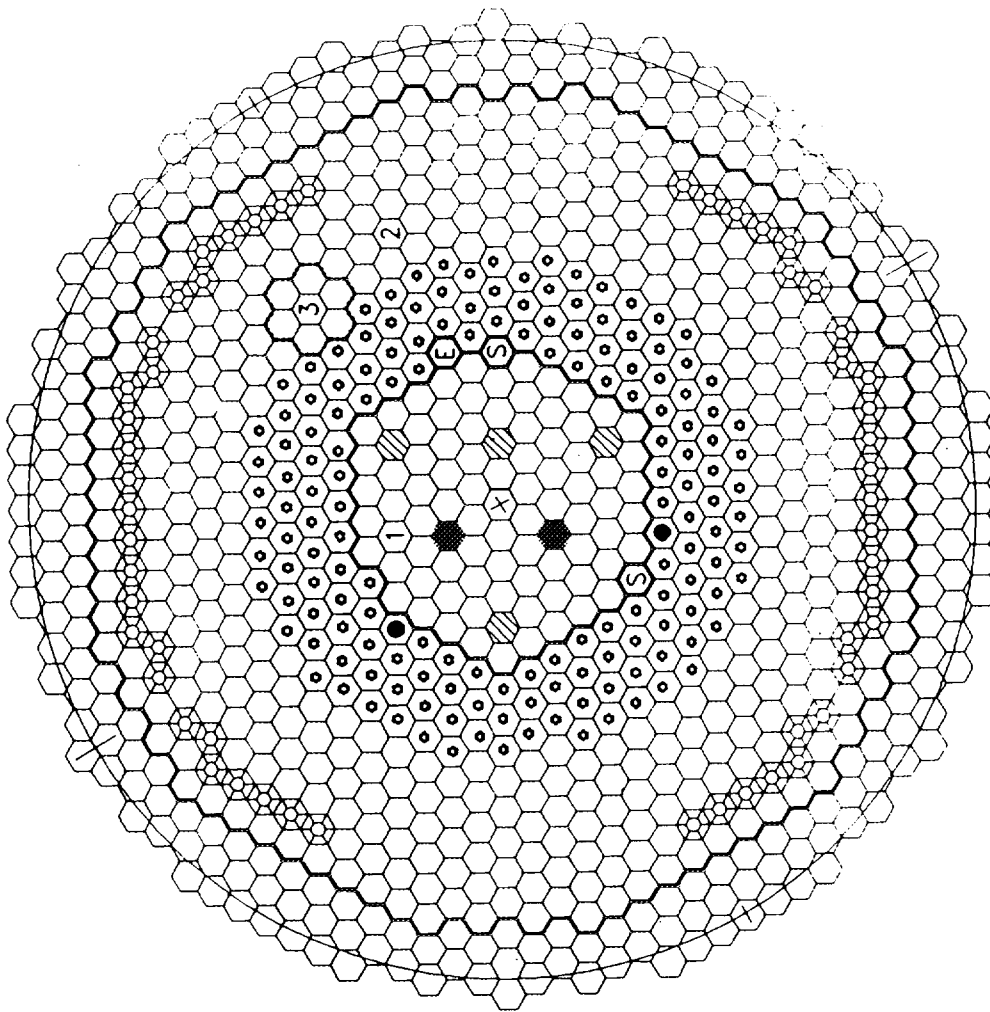
FIG. 3. CEFR Core.

①	Fuel Subassembly	82
⊙	Blanket Subassembly	158
②	Reflector Subassembly	378
⬡	Spent Fuel Storage	34
⬢	Safety Rod	2
⬢	Compensation Rod	4
⦿	Regulation Rod	2
Ⓢ	Neutron Source	2
Ⓔ	Irradiation Facility	1
③	Irradiation Loop	7

FIG. 4. Reactor block.

C	t/h
P	Mpa
T	°c

1 SG
2 Waste drain
3 Expansion vessel for waste drain
4 de-temperature and de-pressure
5 Turbine
6 Generator
7 condenser
8 water supply
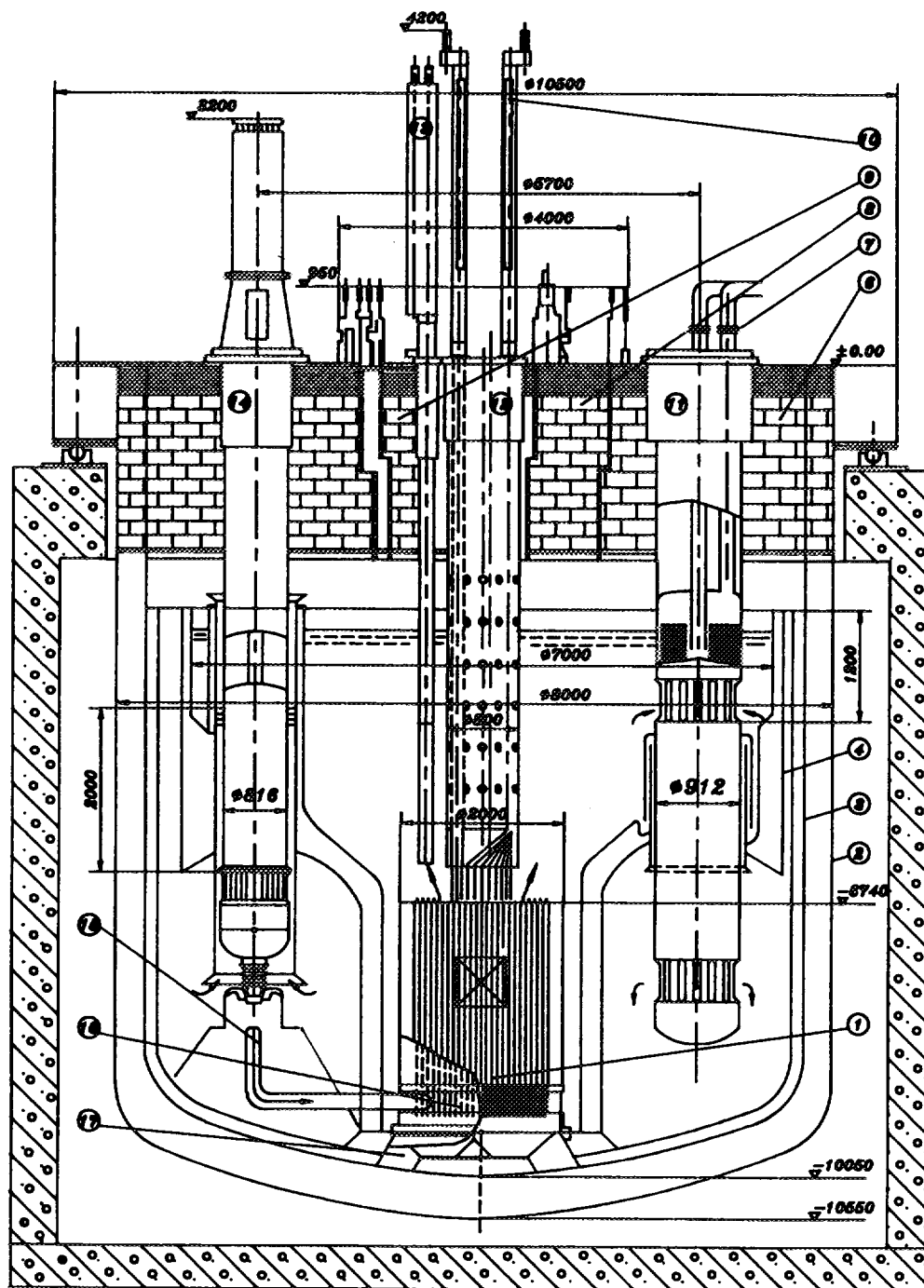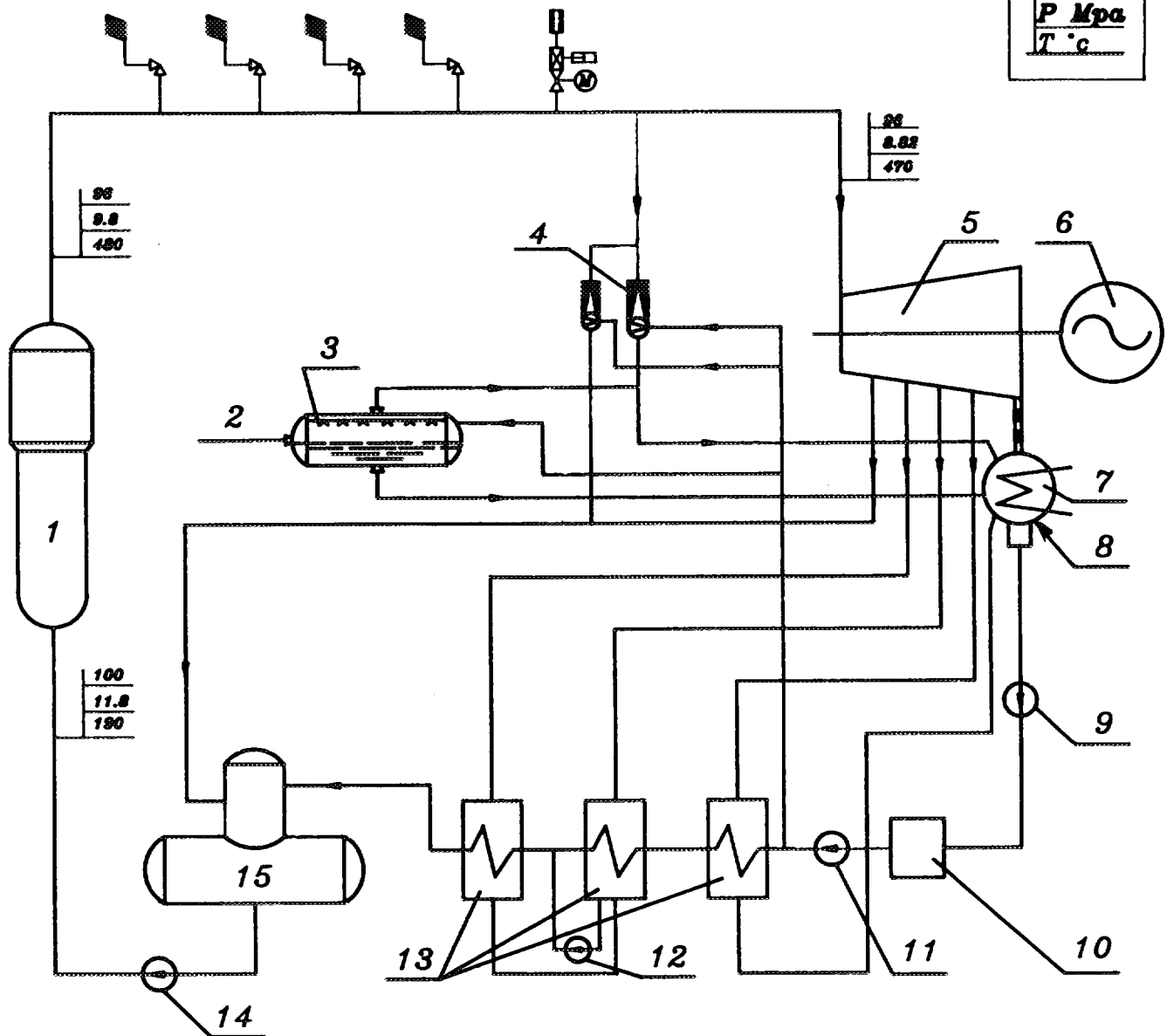
9 pump for condensed water
10 demineragation facility
11 raising pressure pump
12 Low pressure pump
13 Low pressure heater
14 water supply pump
15 de-oxygen

FIG. 5. Water-steam circuit of CEFR.

Table 2　Main Parameters of CEFR

Characteristics	Unit	General	first core
Thermal power	MW	65.5	
Electric power (turbo-generator)	MW	25	
Net electric power	MW	20	
Outlet/inlet average Temperature of the core	℃	530/360	
Fuel		$(Pu, U)O_2$	UO_2
Fuel loading Pu(total)	kg	121.6	
Pu-239	kg	93.2	
U-235	kg	97.6	257
U-235 enrichment	%	30	60.5
linear power max.	W/cm	430	
Burn-up max.	MWD/t	50000	
Neutron flux	n/cm².s	2.97×10^{15}	3.40×10^{15}
Isothermal coefficient	PCM/℃	-4.86	-4.14
Power coefficient	PCM/MW	-8.60	-8.51
Bubble coefficient max.	$	-9.73	
Average Doppler constant			
without Na	Td ρ/dT	-2.05×10^{-4}	-2.89×10^{-4}
with Na	Td ρ/dT	-2.47×10^{-4}	-3.69×10^{-4}
Neutron life	s	1.63×10^{-7}	2.02×10^{-7}
Delayed neutron fraction		4.48×10^{-3}	7.08×10^{-3}
Fuel subassembly			
No. of S. A		82	
Total length of S. A.	mm	2282	
Distance between center to center	mm	61.5	
Width across flats	mm	58.5	
No. of fuel pin per S. A		61	
Pitch of pins	mm	7	
Diameter/thickness of cladding	mm	$6 \pm 0.025/0.4 \pm 0.03$	
Diamter of wire	mm	0.95	
Diameter of fuel pellet	mm	5.1	

Cont

Characteristics	Unit	General	first core
Safety and control subassembly			
First shut-down system (control subassemblies)		6	
Second shut-down system (safety subassemblies)		2	
Absorber		B_4C	
Enrichment of B-10	%	91	
Primary loop			
Total sodium	t	260	
No. of circuits		2	
Flow rate of pump	t/h	2×713	
Pressure	mNa	35	
Rotating velocity	r/min	1000	
Inlet/outlet temperature Of IHX (primary Na)	℃	514/349	
Inlet/outlet temperature Of IHX (secondary Na)	℃	310/495	
Secondary loop			
Total Na	t	48.2	
Flow rate of pump	t/h	2×493.2	
Pressure	mNa	35	
Rotating velocity	r/min	1000	
Steam generator			
Type		straight tube	
Diameter/thickness of the shell	mm	EV:592/11	SH:540/11
Outlet/inlet temperature of Na	℃	310/463	463/495
Outlet/inlet temperature of H_2O	℃	340/190	480/340
Steam pressure	MPa	10.0	10.0
Steam rate	t/h	95.44	95.44
Third loop			
Turbine pressure	MPa	8.82	
Turbine temperature	℃	470	

4. Input of the CEFR-25 technical design

Spent almost whole year 1994 for the input preparation, the CEFR-25 technical design has been started in the early of the year 1995.

Based on PWR and FBR technology study experiences in China and FBR experiences in other coumtries the most design criteria for the CEFR-25have been established, including:

plant states classification;

safety classification of components and systems;

design criteria for components and systems

Concerning the input parameters for technical design, the reactor power, the sodium outlet temperature of the core, linear power of the fuel pin, steam parameters are the same as for CEFR-25 conceptual design. the burn-up rate in the first core still 50000MWd/t, the maximum target burn-up rate will be 100000MWd/t.

More clear requirements to radioactive material release from CEFR-25 have been proposed as an input for technical design.

a. Operational states (Normal Operation andAnticipated Operational Occurrences)

The public maximum annual effective dose equivalent as additional exposuredue to releases fromnuclear plantof gases andliquids containing radioactivity as a consequence of operational states,specified by Chinese National Authority is not more than 0.25 mSv (GB 6249-86). In the case of CIAE, all old facilities containing radioactivity have offered 0.07mSv as maximum exposure to the public. So 0.05 mSv is given as the limit ofpublic annual effective dose equivalent from the CEFR-25 and rest 0.13 mSv for future facilities.

Under operational states the limits of the annual releases from theCEFR of radioactive gases and liquids are given in Table3 in which calculation values are based on the CEFR conceptual design.

Tabel. 3   Limits of Radioactivity Releases

Gases	Calculation(Bq/a)	limits (Bq/a)
Inert gas	5.99×10^{12}	5×10^{14}
Iodine	2.61×10^{8}	1.5×10^{10}
Particles	2.4×10^{7}	4×10^{10}
Liquids		
Tritium	7.7×10^{7}	3×10^{13}
Other nuclides	2.5×10^{9}	1.5×10^{11}

b.   Accident States

(1)   Design Basis Accidents (DBA)

According to the guidance HAF 0703 issued by the State Nuclear Safety Administration, the intervention level is 5–50 mSv to whole body forhiding oneself and 50–500 mSv to whole body for evacuating when the emergencyof nuclear radiation accident has happened. Furthermore the CEFR site is only about 45km far away from Beijing city which has more than 10 million residents. So hope that no any emergency action for any accident happened.

Therefore, it is stipulated that after a Design Basis Accident the maximum public effective dose equivalent should be less than 0.5 mSv (for thyroid 5 mSv) in the CEFR case.

(2)   Beyond Design Basis Accidents (BDBA)

As the same reasons for DBA, after a BDBA the maximum public effective dose equivalent should be less than 5 mSv (for thyroid 50 mSv).

In the accident duration (30 days) the acceptable collection effective dose of residents in the region of 80km radius should be less than 2×10^4 man Sv, the same value also for collection thyroid effective dose.

c.   Intervention Requirements

The sufficient measures of reactor safety control and of prevention from radioative material release should be provided in the CEFR so that there is noany emergency intervention requirement for the residents beyond 4 0 0m from thereactor site.

5.   Research and Development for CEFR-25

The basic research for FBR technology in china was started in the end of 1960s during which the research emphasis were put on following fields:

reactor neutronics;

thermohydraulics;

materials;

sodium technology and

safety

up to the year 1987 more than 10 small facilitics and sodiam loop have bcen established.

Since the year 1987 th R&D for CEFR-25 were started, the great attention were played to CEFR-25 design study, sodium technology, fuels and materials and safety, up to now about 20 facilities and sodium loops insmall scale have been built which features of the reseach during this period are more closed to link the design request, some facilities are listed in Table 4.

Table4 Sodium Loop and Test Facilities

loops sand facilities	main parameters	complete time
Sodium purification loop	After purification O<10ppm H<10ppm Ca<10ppm C<30ppm	1991. 3
Automatic plugging meter	Temp. ranging(135~250°C) ±5°C	1990
Sodium cleaning tacility	PH=7~8. 3	1991. 3
Heat transfer sodium loop	Flow rate max. 30m³/h Pressure of pump 0.5MPa Transfer power 320KVA Total Power 500KVA Temp. 400~500°C	1990
Noise Signal collection system	for Temp. flow or pressure 0~100Hz for sonic 0~40Hz	1990
Sodium Boiling loop	Temp. max900°C Pressure 0.6MPa	1991. 3
Steam explosion facility	Reaction vessel Φ100×400mm with Heating system. High sgeed camara system. Signal sensor system	1990
Material Irradiation chamber fitted at Tandem	High lon:13MV Ni, C. P. Damage 100~150dpa Temp. 350°C~650°C	1990
High temperature sodium corrosion loop	Temp. 550°C~600°C Flw rate 20m³/h Velocity ranging 2~12m/s O_{min}~ppm	1991. 3

cont

High temperature sodium mass transfer loop	Temp. max; 650℃ O<20ppm Veicity 2~3m/s With high and low temp. test section	1991.3
High pressure bi-axial creep test facility	Pressure 200 MPa Furnace size Φ73×500mm	1990
Material fatique and creep sodium loop	Fatique test section 1 Creep test section 2 Temp. 500℃ Load precision ±1% System pressure<1MPa	1991.3
Fuel and cladding chemical interaction facility	Temp. max. 750℃	1990
High vacuum carbon hot pressing furnace	Vacuum5×10^{-5}Torr. Temp. 2500℃	1990
High freguency induction furnace and injection facility	Temp. 1973° K Vacuum 1.33×10^{-3}Pa(hot condition) Injection pressure 0.4MPa Volume. max. 200g	1990
Irradiation capsule	One pin irradiation in Swimming Pool Research Reactor	1990
H-meter sodium loop	Na volume: 100l Temp max. 550℃ Pressure 0.6MPa	1990
Sodium-H$_2$O reaction test facility	Na volume 40l Reaction vessel 10l Na flow tate 0-2m/s Na Temp. 300~500℃ H$_2$O inject velocity 0.01~1g/s H$_2$O Temp 350℃ H$_2$O pressure 16MPa	1990
Blast Menbrane and Blast discharge facility	Pressure threhord: ~0.82±0.005MPa	1990

Two sodium loops in large scale which are presented as a gift by ENEA, Italy are under reconstruction. Those loops and their auxiliary systemswill play an important role to test the endurance and thermal shockproperties of core subassemblies and to test some components of the CEFR-25.

6. Conclusion

The project CEFR-25 is the first step of Chinese FBR technolgy development program which is waiting for the approval of construction by the National Planning Commission. It's obviously that FBR development and deploymentare much important for the supply of electricity in next century and this technology should be developed step by step in China.

XA?6 43088

AN OVERVIEW OF REARA REFERENCE DESIGN

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Abstract

In a effort to keep fast reactors as an option for an eventual future utilization in Brazil, the Advanced Studies Institute (IEAv) is coordinating a research project (called REARA, from the Portuguese REAtores RApidos (Fast Reactors)) which aims to establish a basic know-how in important aspects of fast reactor technology. A reference design for the primary circuit of a 60 MWt experimental fast reactor has been completed. Fuel pin dimensions and other data were taken mostly from PRISM and EBR-II and were used for calculations which lead to a general core configuration. The reference design is presently being used for testing our calculational tools, for checking different methodologies for core calculations, etc.

The REARA reference design is presented, with emphasis in the main results and also in the contribution given by the project team in areas such as neutronics, thermohydraulics, etc.

1- Introduction

Brazil is a large country, with a population of more than 150 million distributed over its 8,5 million square kilometers. The "per-capita" electricity consumption is of the order of 1500 kWh, a value which is low when compared to most advanced countries. About 95% of the electricity come from hydrical resources, most of them located in the southeast region of the country. The estimated hydroelectric potential is 260 GW, from which about 60 GW is already being used. A recent study made by the Brazilian Federal Energy Board -ELETROBRAS-, which takes into consideration different scenarios for economic growth and the resources available today and also the possibility of exploring new resources, indicates that by the year 2015 the hydroelectric potential will be exhausted and, thereafter, the demand will have to be supplied by a growing number of thermoelectric power plants, either conventional or nuclear.

In spite of the planned participation of thermal reactors in the electricity grid- 1 PWR/Westinghouse (626 MW), already in operation and 2 PWR/KWU (1300 MW), awaiting decision for construction- it has been felt that efforts should be made in order to provide a sound background for a future transition to fast reactors. In 1992, after many discussions, authorities were convinced that a long term fast reactor program should be maintained, if Brazil wished to keep this reactor concept as an alternative for future use. A R&D project (called REARA, from the Portuguese expression REAtor RApido, which means Fast Reactor) was then established with the objective of having, within 25-30 years, an experimental reactor in which relevant experiments could be performed. Although the materialization of this reactor is distant in the future and also subject to many uncertainties, a decision has been taken to prepare its reference design to serve as our "experimental installation" and also to be used for testing calculational tools, checking different methodologies for core calculations and so on.

The main objective of this paper is to present an overview of the REARA reference design, emphasizing the main results and the difficulties encountered so far. It is important to stress that the reference design is part of a group of fast reactor research activities which intend to be a seed of a gradually growing program with the involvement of other Brazilian institutions.

2- REARA Reference Design
2.1- Objectives for the Experimental Reactor

In order to provide the "decision-makers" with information that would permit a better judgment on the future "value" of the fast reactor research project, we decided to fix the following objectives for the experimental reactor:

- to acquire experience in fast reactor design, in component fabrication and maintenance and in licensing, construction and operation;
- to verify the reactor inherent safety characteristics;
- to serve as an installation for R&D activities in fuel and other material testing;
- to verify the possibility of minor actinide burning and, eventually;
- to generate electricity.

In practice, this means that all efforts are to be made in this direction, which is not to be changed. Main research activities are aimed to emphasize safety and fuel cycle characteristics. Additional research has been performed in areas such as materials (HT-9), uranium recovery via electrorefine techniques, shielding, etc., within a managerial strategy of generating the maximum number of relevant results with a tight budget. We believe this will help convincing our "decision makers" that a consolidated long-term fast reactor R&D program will be useful not only for generating electricity, but also to provide a sound technological basis which will reduce the gap with respect to more advanced programs, facilitating international cooperation.

2.2- Experimental Reactor Main Characteristics

The main characteristics for the reference design were chosen in 1992, in such a way as to benefit from the recent international experience and also from the fact that previous Brazilian activities in fast reactors, although frustrated, have left a "seed" for future developments. This was the main reason for selecting sodium as coolant and U-10%Zr metallic alloys as the reactor fuel. The Instituto de Engenharia Nuclear- IEN (Nuclear Engineering Institute, in Rio de Janeiro) has a small sodium loop (100 kW) in operation and three others which are not mounted yet due to insufficient funding. The Instituto de Pesquisas Energéticas e Nucleares- IPEN (Nuclear Research Institute), which is located in Sao Paulo, had previous experience with metallic fuel for research reactors and became strongly motivated in engaging in a challenging research towards the development of U-Zr alloys.

The main reactor characteristics are indicated in Table 1.

2.3- Core Reference Design
2.3.1- Fuel Assemblies

The concept of the fuel assemblies for the REARA reference design was based on the international experience, with the dimensions taken mostly from PRISM /1/. The number of fuel pins in the assembly is 67, separated by helicoidal wire wraps and

Table 1: Main Design Characteristics

Thermal power (MW)	60
Electric power (expected) (MW)	20
Primary operating pressure (MPa)	0.11
Core inlet temperature ($^\circ$C)	370
Core outlet temperature ($^\circ$C)	470
Maximum burnup (MWd/T)	70000
Fuel type	U-10%Zr
Coolant	Sodium
Primary coolant circuit arrangement	pool type
Cladding material	HT-9

arranged in a triangular matrix, forming a hexagonal set. The fuel is a U-10%Zr metallic alloy with active length equal to 62.0 cm and having two (top and bottom) 30.0 cm nickel reflectors.

Table 2 shows the main data for both the fuel pins and fuel assemblies.

2.3.2- Control Assemblies

In the core there are 6 control assemblies (4 primary+2 secondary). The control material is boron carbide (B_4C) enriched in ^{10}B, with the pellets being clad in stainless steel cylindrical tubes. In the present reference design the control assembly inserting and withdrawing mechanisms have not been defined.

Table 3 shows the main data for both the primary and secondary control assemblies.

2.3.3- Special Assemblies

Consideration has been given to the utilization of some types of special assemblies distributed in the reactor core, in order to provide means of controlling neutron leakage, protecting against loss of flow accidents, controlling the excess of reactivity and so on. Among them we have the Gas Expansion Module (GEM), which has been tested in the Fast Flux Test Facility (FFTF) /2/ and is to be located (3 assemblies) in the active core periphery in order to protect against loss of flow.

Table 4 shows the main data for the gas expansion assemblies.

The core arrangement which has been used for calculations /3,4/ is indicated in Figure 1. The targeted average burnup is 70 MWd/kg, which will require about 1000 days of full power operation.

Table 2: Main Fuel Data

Fuel assembly	
Total length (mm)	3340.0
Number of pins/assembly	67
Duct wall-to-wall distance (mm)	71.0
Duct material	HT-9
Volume fractions (%)	
Fuel	41.0
Sodium	37.7
Structure	21.3
Fuel pin	
Fuel type	U-10%Zr
Active length (mm)	620.0
Pitch (mm)	8.4
Pitch/Diameter	1.17
Fuel diameter (mm)	5.37
Pin diameter (mm)	7.2
Cladding thickness (mm)	0.5
Cladding material	HT-9
Wire diameter (mm)	1.2
Total fuel pin length (mm)	2090.0

Table 3: Main control assembly data

Primary control/Secondary control (#)	4/2
Control material	B_4C
Enrichment (% ^{10}B)	90.0
Control follower	Na
Volume fractions (%)	
Sodium	40.0/94.9 [a]
Structure	22.6/5.1 [a]
Control	37.4/- [a]

[a] Value for control inserted/Value for control withdrawn

Table 4: Main gas expansion assembly data

Number (#)	3
Duct material	HT-9
Inert gas	He or Air
Volume fractions (%)	
Sodium	94.6
Structure	5.4

2.4- Neutronics and Termohydraulics Calculations

Using cross section data from the Japanese library JFS-2 (JAERI-Fast Reactor Group Constant Set - Version II, 70 energy groups) and the diffusion theory codes EXPANDA (generates weighting spectrum in 70 groups and uses it to collapse cross sections do 6 groups) and CITATION (RZ geometry, 6 energy groups), evaluation of some reactor parameters have been performed and are indicated in Table 5. Table 6 shows results for core inventory as a function of burnup.

Table 5: Main core parameters

Maximum flux (10^{15} n/cm^2.s)	1.97
Average flux (10^{15} n/cm^2.s)	1.07
Doppler coefficient (-TΔk/ΔT)	7.07×10^{-4}
Sodium void reactivity (-Δk)	3.27×10^{-2}

Table 6: Core inventory (kg) as function of burnup

	0 days	100 days	200 days	500 days	800 days	1000 days
^{235}U	391.0	384.0	376.0	354.0	333.0	318.0
^{236}U	-	1.229	2.451	6.074	9.627	11.950
^{238}U	420.0	419.0	417.0	414.0	410.0	408.0
^{239}Pu	-	0.733	1.461	3.619	6.428	7.118
^{240}Pu	-	-	0.003	0.021	0.055	0.088

REMOVABLE
SHIELDING

INTERCHANGEABLE
REACTIVITY
COMPENSATION ZONE

ø 151.76

ø 16.00

SPENT / FRESH
FUEL CANISTERS

ø 231.26

CONTROL ASSEMBLY

FUEL ASSEMBLY

GAS EXPANTION MODULE
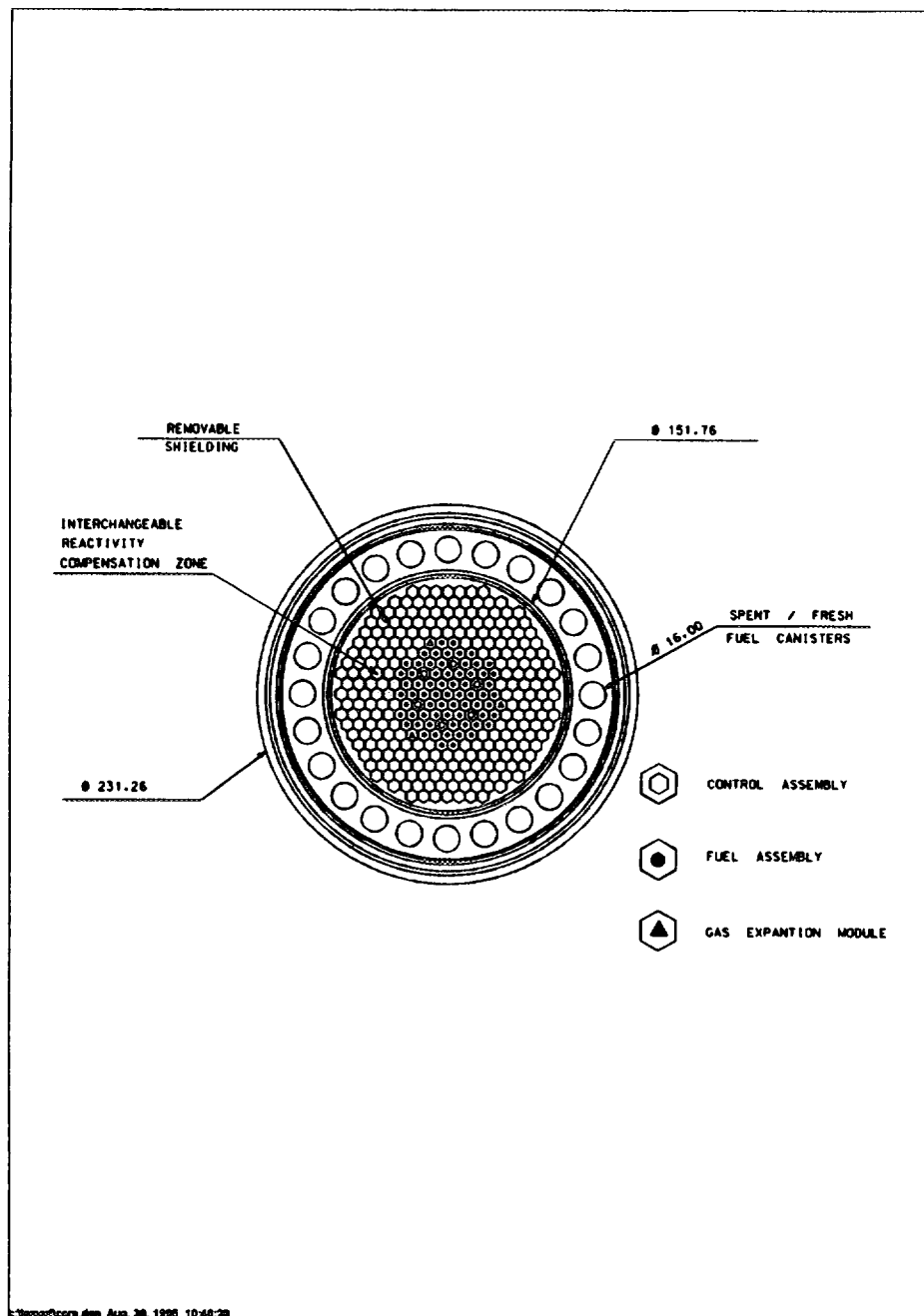
Figure 1: Core arrangement at BOL

It is important to stress that the results which have been obtained must be considered within the perspective of an initial reference design, useful for checking calculational tools and methodologies. For this purpose, an independent core calculation is being performed at the Instituto de Engenharia Nuclear - IEN (Nuclear Engineering Institute), with no conclusive results so far, but already giving some indications of discrepancies due to nuclear data, which are presently being investigated.

Based on data selected from the literature on EBR-II and PRISM, a set of reference values for thermohydraulics calculations have been fixed and used for the reference design of the primary sodium pumps and for the Intermediate Heat Exchanger (IHX)/5/. Main characteristics of the IHX (shown in Table 7) were determined with the computer program TCIPRO /6/, which is written in C-language and runs on a PC386.

56

Table 7: IHX main characteristics

Primary sodium flow rate (kg/s)	226.0
Secondary sodium flow rate (kg/s)	152,0
External radius (m)	1.699
Metal tube number (#)	1397
Central pipe OD (m)	0.311
Total heat transfer area (m2)	233.0
Total height (m)	3.859

We are now in the process of integrating important items such as core, IHX, pumps, etc. into a simple software simulator which will be of help in performing stability analyses and so on.

3- Concluding Remarks

Rather than investing a great deal of effort towards developing an elaborate conceptual design for the experimental reactor, we decided to establish a simple reference design which could be used for checking calculational tools and for testing different methodologies. For some time in the future, the reference design will be our "experimental installation". The transition to a conceptual design of our own will definitely come, of course conditioned to the consolidation of the program, which will be facilitated by the celebration of formal cooperation agreements with other institutions in Brazil and maybe abroad.

Our past activities have put too much emphasis on the reactor itself, with a few conceptual designs being proposed, some of them very interesting. but with no receptivity by the Brazilian technical community. Our "decision makers", with their attention turned to short term problems, were not impressed either. We felt it was necessary to set up a managerial strategy which could create a compatibility between the long term objectives and the need for short term support. For motivating the participation of other research institutions, we have decided to emphasize the several technical challenges in areas such as reactor safety, fuel, materials and so on, and to avoid (at least for the time being) discussing merits of fast reactors with a community biased towards thermal reactors. For motivating the "decision makers" we concluded that each phase of the project must generate the maximum number of results which are not only relevant, but also which are easy to be evaluated. In practice, this means that every result should be transformed in an asset for larger funding. The lesson we have learned from our past experience is that we should qualify for "big money" after demonstrating that good results can be produced with "small money".

In order to avoid repeating the mistakes made in the past, we are now planning for a gradual growth. The project partial objectives (and with good potential to be attained!), including the reference design, are presently being defined as to rapidly provide a common basis for discussions with other institutions in Brazil. The idea is to show that a long term fast reactor program will not only result in the materialization of the reactor, but can also generate, along its way, relevant "know-how" which can also be useful for other sectors. This strategy is showing itself to be efficient: in 3 years the number of

people involved jumped from 13 to about 40, which is still an insignificant participation when looked from the viewpoint of more advanced programs. For us this signifies that an important seed has been sown, as judged also by the fact that now there are some interesting research activities going on in different institutions and funding is increasing, although still limited by the present economic problems the country is facing.

Difficulties in accessing calculational codes or nuclear data, many of them of restricted distribution, is making us either use the existing ones (with their limitations) or develop our own (with difficulties in qualification). The international effort towards consolidating fast reactors as the only proven technology capable of efficient utilization of depleted uranium and transformation of long-life transuranic waste must take this aspect into consideration. Fast reactor activities in some developing countries, with good potential to give small but important intellectual contributions to the fast reactor field, eventually end up having an early termination, due to difficulties in accessing more updated calculational tools. This may not have a simple solution, but it needs to be discussed. The future of fast reactors in Brazil will certainly depend on our own efforts and on our "decision makers". But, in our viewpoint, it will also depend on establishing the grounds for an international cooperation which promotes the maximization of the contribution that developing countries can give to the international efforts in the fast reactor field.

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THE DEVELOPMENT OF DEMONSTRATION FAST BREEDER REACTOR (DFBR)

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Abstract

The demonstration fast breeder reactor(DFBR) has been under development as the essential step toward the commercialization of the Fast Breeder Reactor(FBR) around 2030. THE JAPAN ATOMIC POWER COMPANY(JAPC) finished the conceptual design study of DFBR in 1993 under a contract awarded by nine electric power companies and Electric Power Development Co. Ltd.. Based on its result, these utilities have decided that the DFBR will be a top entry loop-type reactor with electricity output of 660MWe. The design targets of the DFBR have been set from the viewpoints of safety, plant construction cost, power generation cost, maintainability and plant repairability. This paper outlines the conceptual design of the DFBR.

1 INTRODUCTION

In the course of developing a Japanese FBR, the experimental FBR "Joyo" is being operated satisfactorily and the prototype FBR "Monju" has achieved criticality as of April,1994. This leads to the prospect of the construction of the demonstration FBR "DFBR" and the commercialization of the FBR at around the year 2030. The design study and related R&D for the DFBR is being conducted by the Japan Atomic Power Co.(JAPC), which is the principal organization responsible for the construction as entrusted by nine electric power companies and Electric Power Development Co. Ltd.. Research and development on the DFBR is being conducted by the Power Reactor and Nuclear Fuel Development Corporation(PNC), the Japan Atomic Energy Research Institute(JAERI), the Central Research Institute of the Electric Power Industry (CRIEPI) and JAPC. These organization established the Steering Committee for Coordinating R&D on the FBR.
In designing the DFBR, it is important for commercialization of the FBR to reduce the plant construction cost, which is rather greater than that of the light water reactor(LWR). From this

viewpoint, two types of FBR were compared. These are the pool type reactor, developed in Europe, and the top-entry loop-type reactor, which has the most compact arrangement of the primary system components. The top-entry loop-type was selected for the DFBR because of following considerations:

1) Major primary components such as the intermediate heat exchangers(IHX) and the pumps are outside of the reactor vessel, and this facilitates maintenance and repair.

2) The system has flexibility to introduce such innovative technologies as the electro-magnetic pump integrated component, which is needed for commercialization of the FBR.

3) Experience gained at the prototype "Monju" must be fully utilized. Considering that the top entry system is quite a new concept, the conceptual design study, the evaluation study of commercialization prospects and the water hydraulic tests using models of thermal-hydraulic properties peculiar to the top-entry system were conducted.

Through these studies, technical feasibility and the possibility of the FBR commercialization was confirmed, and the plant concept was also established.

Also based on the results of these studies, the presidential committee of the Federation of Electric Power Companies determined the basic specifications for the No.1 DFBR in January 1994, and decided to promote its development aiming for start of construction in early 2000.

The design of the top entry loop type DFBR is summarized below.

2 WHOLE SYSTEM

The design targets of the DFBR are as follows:
1) Safety level equivalent to that of LWRs
2) Economic feasibility at about 1.5 times the cost of the LWR on a 1,000MWe basis
3) High burnup and long operating cycle to reduce cost of electricity generated
4) Reactor outlet temperature of 550 °C to achieve high thermal efficiency
5) Easier maintenance and repair, taking advantage of the distributed equipment layout.

Table 1 lists the major specifications of the plant.
Figure 1 shows the appearance of the top-entry loop type DFBR. The thermal output of the reactor is 1,600MW with three main cooling loops, allowing for the capacity of the primary components to be enlarged gradually toward a large FBR in the commercialization phase. The main steam conditions are 495 C and 169atg, in order to reduce generation costs based on the structural feasibility of the once-through type steam generator (SG).

The primary cooling system consists of a reactor vessel, three IHX vessels, three pump vessels and reverse-U shaped pipings to connect these vessels. The reverse-U shaped pipings reduce the piping length and the required installation space, and eliminate vessel nozzles. These vessels have free coolant surfaces and the argon cover gas is maintained at equal pressure by mutually connecting pipings. The cover gas pressure is about 0.9atg in order to maintain positive pressure at the highest point of the primary piping. The coolant surface level in the IHX vessel

Table 1   Plant Reference Specifications

| Item	Specifications
Basic specifications	
1 Reactor type	Top entry loop type reactor
2 Thermal output	1,600MWt
	(Electric output: about 660MWe)
Other major specifications	
3 Number of loops	3 loops (about 530MWt/loop)
4 Reactor outlet temperature	550°C
5 Main steam temperature and pressure	495°C/169atg
6 Core and fuel	
(1) Core	Homogeneous core
(2) Fuel	Pu/U mixed oxide fuel pellet with center hole
(3) Burnup	About 90GWd/t (Initial phase)
	About 150GWd/t (High burnup phase)
(4) Breeding ratio	With blanket : About 1.2
	Without blanket : About 1.05
7 Steam generator	Once-through type
8 Reactor shutdown system	2 independent systems
9 Decay heat removal system	4 loops of DRACS
10 Reactor containment	Rectangular reinforced concrete containment with liner
11 Fuel handling system	Rotating plug, manipulator type
12 Reactor building	Horizontal seismic isolation building
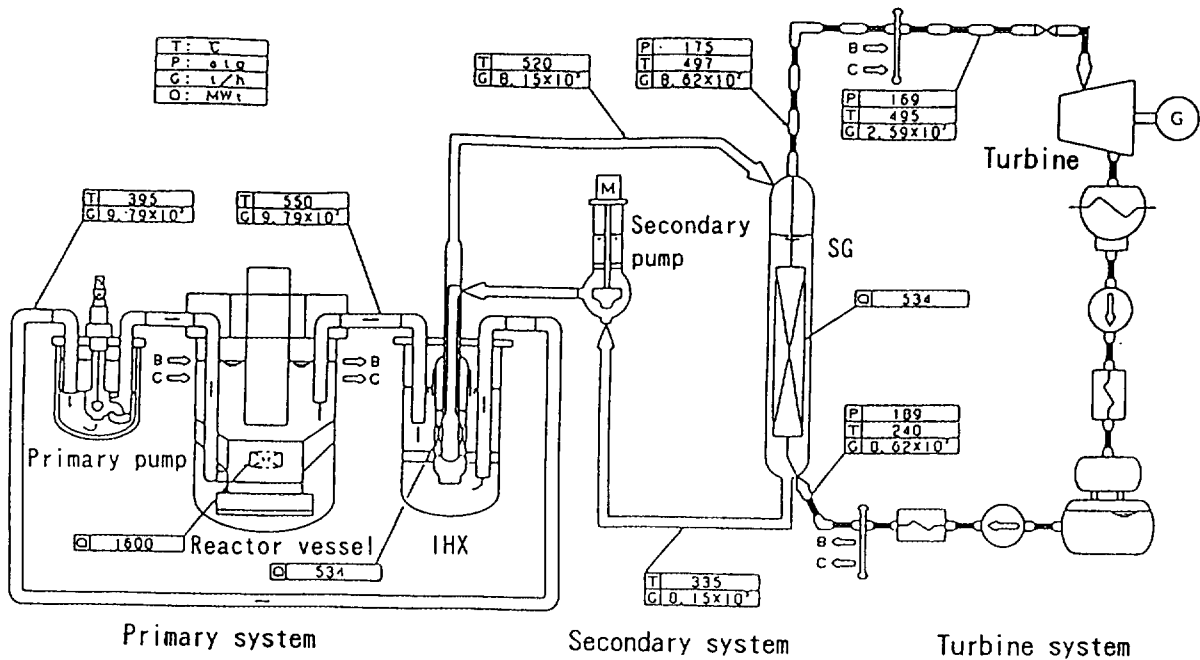
Figure 1 Cooling System & Heat Balance

during operation is lower than that of the reactor vessel due to the pressure loss in the hot leg piping.

The decay heat is removed by the turbine bypass line if the water-steam system is available, or by the decay heat removal system as a safety system if the water-steam system is unavailable.

3 CORE AND FUELS

The FBR has the potential to raise burnup, extend the operating cycle and reduce generation costs by using long-life fuel cladding, because the FBR generates new fuel during fuel burning. The swelling due to neutron irradiation and high temperature creep influence the life of the cladding. For this reason, the initial phase core based on the current technology is designed to produce an average burnup of 90,000MWd/t with an operation period of the order of 15 months. We are also aiming at improving the design to give better burnup and longer operating cycle, with the expectation that cladding materials with excellent resistance to swelling will probably be developed during the life of the plant, while still maintaining plant condition. The design of the breeding ratio is variable between 1.2 to 1.05 with or without a radial blanket, which provides flexibility for the balance of supply and demand of plutonium.

A fuel pin of 8.5mm diameter is used to achieve the burnup and operating cycle listed in Table 1.

Increasing the margin for fuel integrity by using fuel pellets with a central hole, the core has two homogeneous regions, and the height and equivalent diameter of the core are 100cm and 299cm respectively, based on the pressure loss and impact on the reactor vessel. Advanced austenite stainless steel (PNC1520) is used as the cladding material of the initial phase core fuel because of its high resistance to swelling deformation and creep damage caused by irradiation with fast neutrons. This material was developed by PNC, and same irradiation data have been accumulated at present.

The required reactivity of the control rods can be attained by 24 control rods in the initial phase core, but 30 control rods are provided in consideration of the high burnup phase core.

For further improvements of core safety, feasibility of a measure to insert negative reactivity by increasing neutron leak passively as the core flow reduces ; namely gas expansion module (GEM), has been studied to be evaluated that GEM alone has a capability to prevent core damage during ULOF for large size core of DFBR class and considered highly effective.

4 REACTOR STRUCTURE

The reactor structure consists of the reactor vessel, roof deck, rotating plug, upper internal structure(UIS), reactor vessel thermal protection structure, core support structure and guard vessel. Figure 2 shows the reactor structure. The material of the reactor vessel and main primary component is the modified SUS316 stainless steel for fast reactors(316FR) with high creep strength. This material is used for the purpose of maintaining structural integrity under the condition of 550°C of reactor outlet temperature.

The diameter(10.4m) of the reactor vessel was minimized so as to adjust the range of travel of the fuel handling machine and the
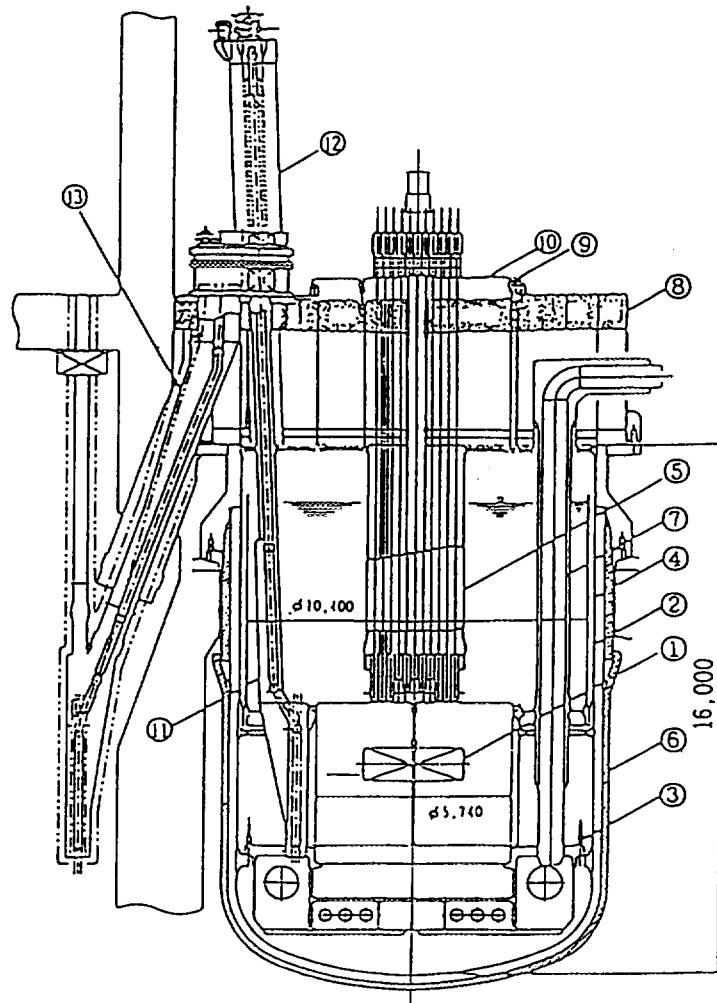
No.	Name	Material	Q'ty
1	Core		1
2	Reactor vessel	316FR	1
3	Core support structure	SUS304	1
4	Reactor wall protection structure	316FR	1
5	Upper internal structure	316FR	1
6	Guard vessel	SUS304	1
7	Inlet pipe	SUS304	3
8	Roof deck	CS	1
9	Large rotating plug	CS	1
10	Small rotating plug	CS	1
11	In vessel fuel chute	SUS304	1
12	Fuel transfer machine	CS	1
13	Ex-vessel fuel chute	SUS304	1

Figure 2   Concept of Reactor Structure

layout of components on the reactor vessel head. The height(16.0m) of the reactor vessel was set so that the fuel assemblies are submerged in coolant during refueling, and in consideraion of the vertical rigidity of the core support structure and traveling properties of in-vessel inspection instruments in the reactor lower plenum. The thickness of the vessel wall was set as 50mm to withstand seismic forces.

The coolant(liquid sodium) discharged from the pump flows into the core support structure through inlet piping, and the main flow heated at the core passes through the upper plenum and the outlet piping, then flows into the IHX. A part of the coolant branches off from the inlet plenum of the core support structure, passes through the intermediate plenum and flows along and cools the reactor vessel wall and exits to the upper plenum, where it joins the main stream from the core.

The guard vessel is installed in order to limit the leakage and cool the core in case of hypothetical leakage from the reactor vessel, and is filled with nitrogen gas.

The roof deck supports the reactor vessel and also supports the rotating plug, UIS, fuel handling machine, and direct reactor heat exchanger(DHX) of the decay heat removal system. The roof deck is a box structure(carbon steel) with ribs of 13.0m diameter, 6.2m high with a 60mm thick plate, and is structurally integrated with the reactor vessel via a different material joint.

Radiation shielding concrete is placed at the upper part of the roof deck. The roof deck, irradiated by the primary piping and the reactor vessel radiation heat, is cooled to ensure that the electrical components installed on the upper surface and the concrete function properly.

The UIS is a cylindrical structure located above the core and supports the control rod guide tubes and instrument wells. The cylindrical drum is made of 316FR with no vertical welding seam, to cope with the thermal stress near the liquid surface. The surface of the UIS near the core outlet is covered with Alloy 718 as a measure against thermal striping.

The low-temperature sodium circulation system keeps the reactor vessel wall temperature at a low temperature so that the creep effect is negligible, because extremely high reliability is required. There are three cylindrical layers of the liner inside the reactor vessel wall, and the cold sodium branched from the inlet plenum to the core support structure flows upward through the clearance between the reactor vessel wall and the outer layer of the liner, flows over the top of outer layer,descends through the clearance between the outer layer and middle layer and finally discharges into the upper plenum. The flow rate for cooling the reactor vessel wall is about 3% of the primary coolant rated flow. Each liner is made of 316FR and thick enough to prevent buckling caused by earthquakes or thermal stress near the liquid surface. An inner layer is provided for the thermal protection of the wall cooling part against 550 in the upper plenum. As for the thin cooling liner, because flow-induced self-oscillation has been generated at the Super-Phenix in France, the mechanism, and the condition of oscillation generation and so forth have been investigated by model tests, the stiffness of liner and vessel wall cooling flow rate were determined in order to prevent the oscillation generation. The discharging part of the low-temperature sodium circulation system is coated with Alloy 718 to prevent thermal striping as the return stream of coolant mixes with the hot sodium in the upper plenum.

The core support structure is a box structure with ribs in order to keep sufficient rigidity to withstand vertical seismic motion while avoiding excessive thickness. It is attached to the reactor vessel with a single plate flange, and the backup structure prevents the core from falling in the event of damage at the single plate.

5 COOLING SYSTEM COMPONENTS

316FR is used for those components of the cooling system operating at high temperature in order to assure the structural integrity of the system.
The structural concept of the IHX is shown in Figure 3.
The design of the IHX is intended to be compact, so that the primary coolant flows inside the heat transfer tube, using a convention straight tube in actual use.
The structural concept of the primary pump is shown in Figure 4.
The primary pump is a single stage, single suction type which has a simple structure and proven experience in previous reactors. The coolant leaked from the pump axial seal part is returned to the pump lower plenum through the leak flow control valve.
A loose-part trap structure is provided on the discharge side of the pump to prevent loose parts, if any, from flowing into the core.
The primary main piping is shown in Figure 5. This piping has a reverse-U shape and the coolant flows in/out the vessel at the top. The piping inserted into the vessel is supported by the roof deck with a support pipe via a Y-piece structure.
The horizontal thermal expansion of the piping is absorbed by the bending deformation of a vertical pipe, and the vertical thermal expansion is absorbed by the sliding joint. The ex-vessel part of the piping has a duplicated construction, and an outer pipe is provided to facilitate rapid detection of any sodium leakage and to help retain the sodium.
The steam generator(SG) is an integrated once-through type with helical-coil heat transfer tubes which was proven in the development of the prototype reactor "Monju". The SG material is modified 9Cr-1Mo steel because of its superior strength at high temperature and its resistance against stress corrosion cracking (SCC).

6 SAFETY FEATURES

6.1 Reactor Shutdown System

In order to assure a level of safety equal to the LWR, the reactor has two shutdown systems, a primary reactor shutdown system and a backup reactor shutdown system, each of which can rapidly stop the reactor independent of each other, considering the core characteristics of the FBR. In order to reduce the risk of common-cause failure, the primary reactor shutdown system employs proven solid control rods of the gas-accelerated insertion type, while the backup reactor shutdown system employs articulated control rods of the gravity-dropping type.
The backup reactor shutdown system is provided with a Self-Actuated Shutdown System(SASS), which is automatically delatched due to the Curie point effect by abnormal rising of coolant temperature.
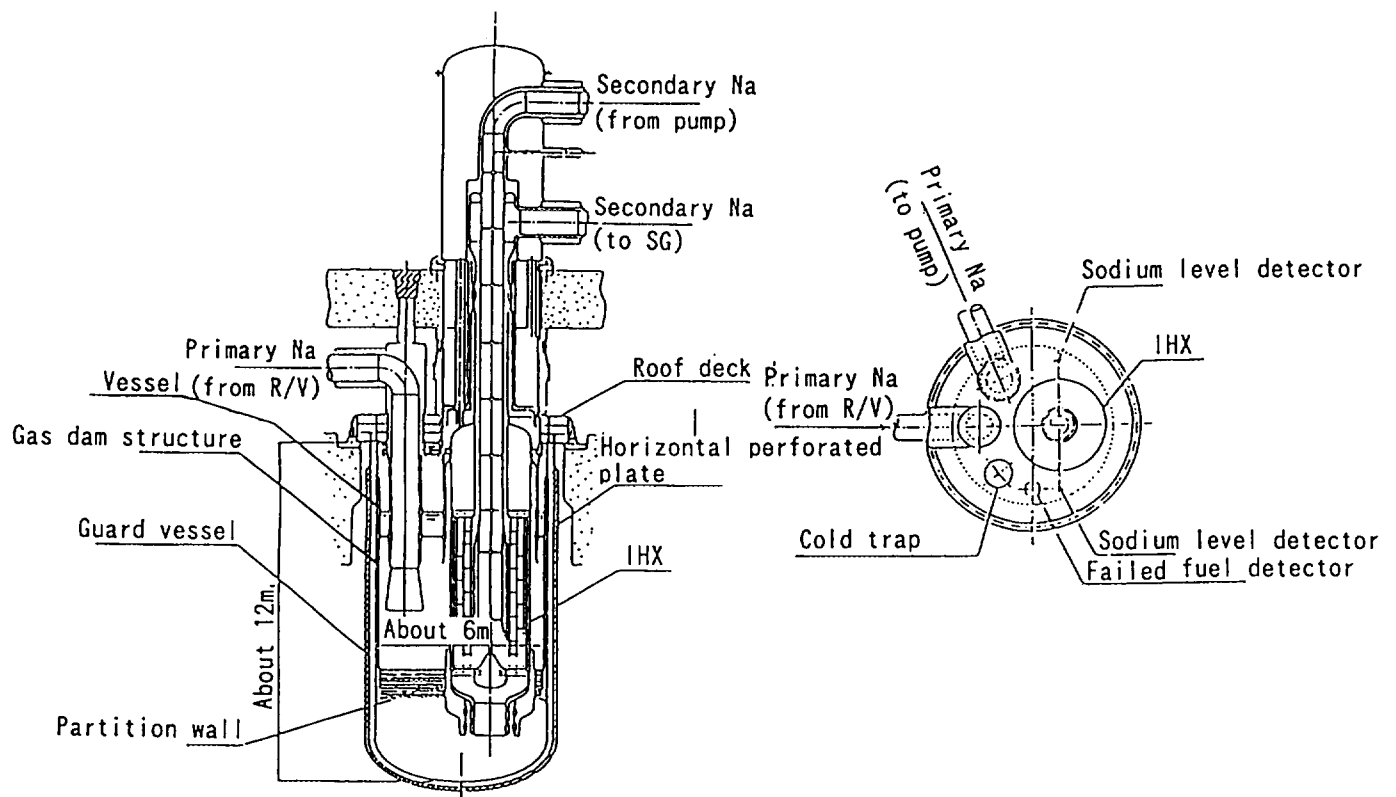
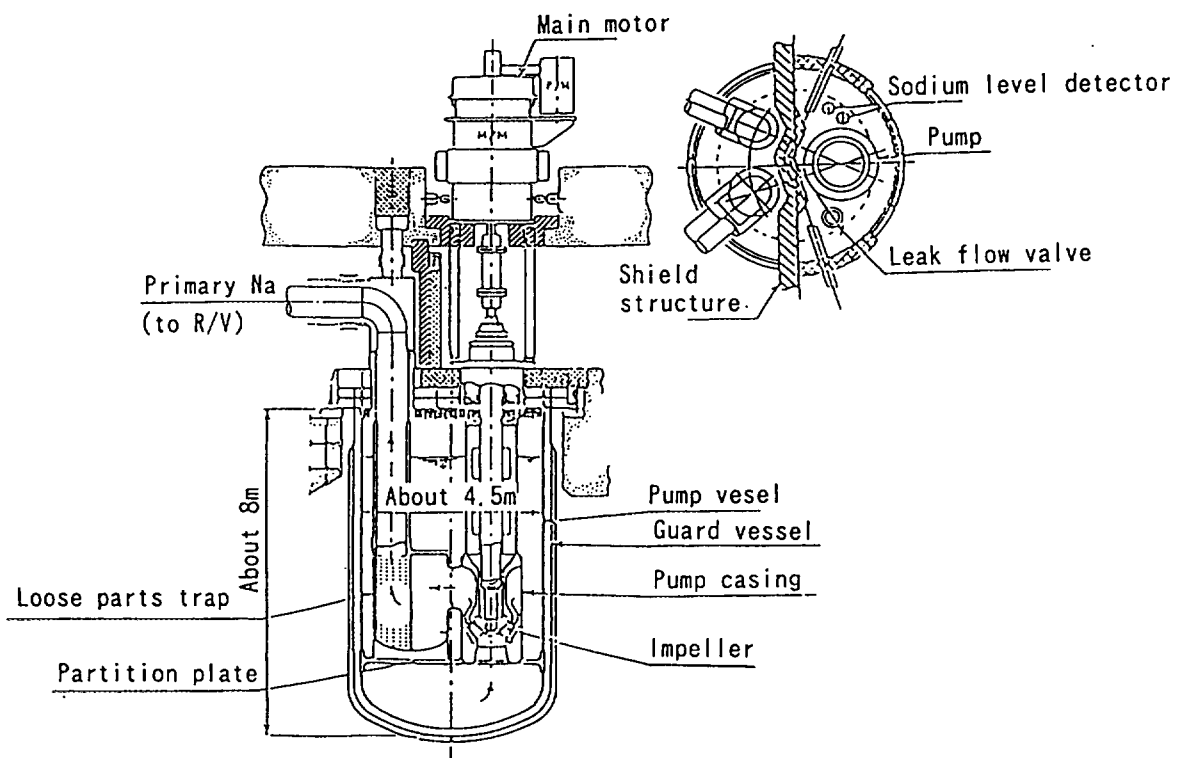Figure 3 Concept of Intermediate Heat Exchanger
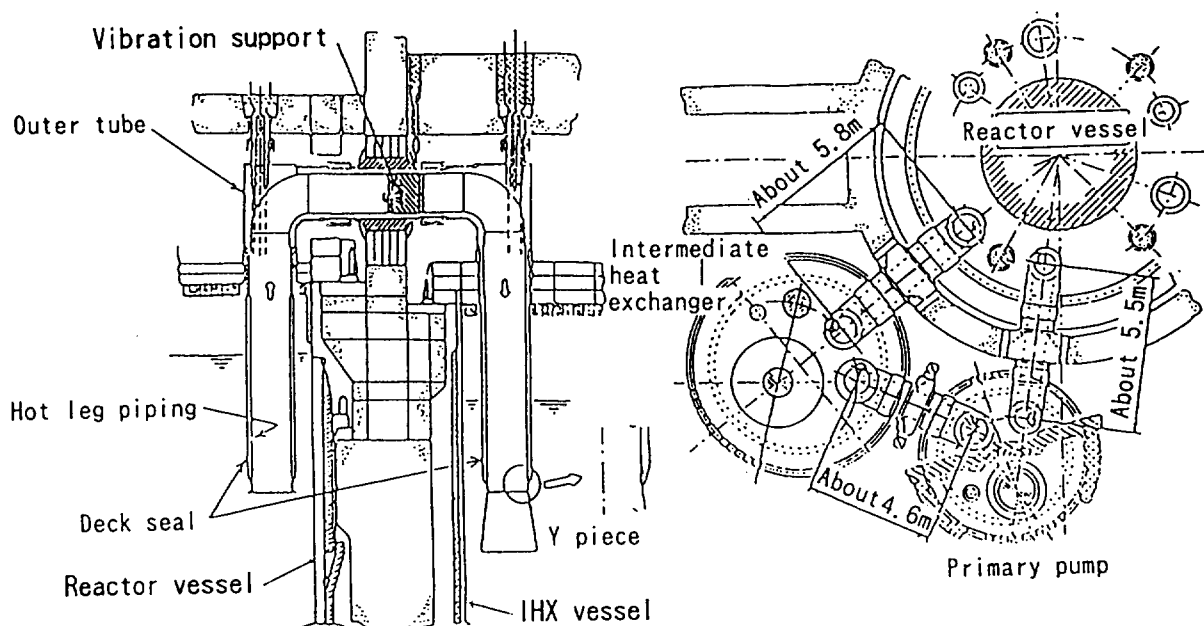

Figure 4 Concept of Primary Pump

Figure 5   Concept of Primary Piping

6.2 Decay Heat Removal System

The concept of the decay heat removal system is shown in Figure 6.
A four-loop Direct Reactor Auxiliary Cooling System(DRACS) is
applied with heat exchangers immersed in the reactor vessel,
which removes the decay heat directly from the vessel.
This design allows decay heat to be removed by the circulation
within the reactor vessel alone. This system employs both forced
and natural circulation utilizing the features of the FBR,
sodium coolant and high-temperature operation. Four independent
systems are installed to provide redundancy. In order to reduce
the risk of common-cause failure, this system has a diversity of
the damper control devices for the air cooler.

6.3 Containment facility

The design conditions of the containment facility were determined
at a design temperature of 150 °C and a design pressure of
0.5kg/cm ,assuming the event sequence that the sodium ejects
into the containment facility at the accident, using both
temperature and pressure to rise. In order to utilize the space
of the facility, a rectangular concrete structure with an inner
lining, integrated with the building, is employed under these
conditions. A secondary containment facility is provided for
those areas with penetrating piping and cables like the LWR.
This is maintained at negative pressure and is provided with an
emergency gas system to absorb radioactive substances.

6.4 Emergency Power Supply

An emergency power supply system with three independent emergency
diesel generators is provided in order to shutdown the plant
safely in the event of loss of external power.
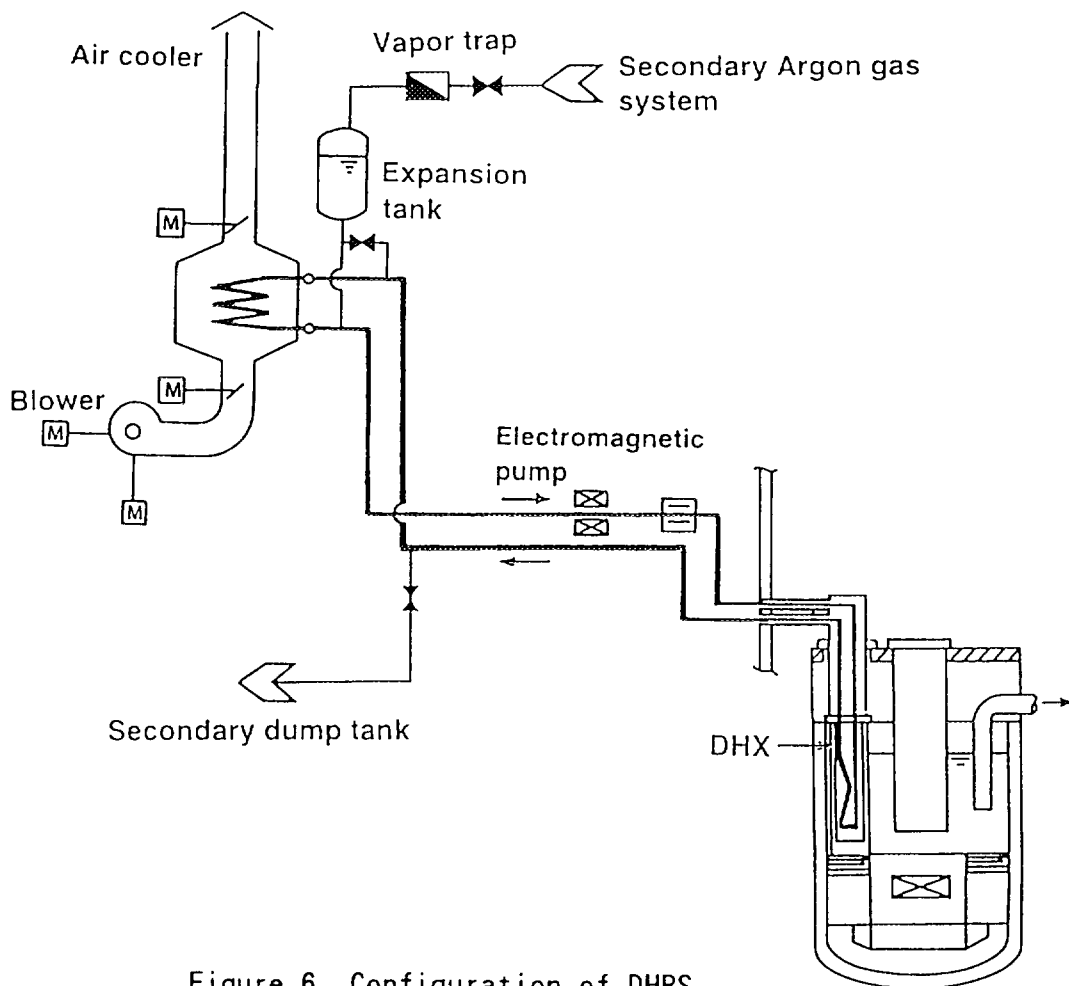
Figure 6   Configuration of DHRS

6.5 Reactor Protection System

The reactor protection system has a "2 out of 4" logic design to improve reliability. Different designs are used for the operation of the primary reactor shutdown system and the backup reactor shutdownsystem, including the detectors and logic circuits, to reduce the probability of common-cause failure.

7 REACTOR BUILDING

Reactor building of the No.1 DFBR may employ a horizontal seismic isolation design to rationalize the facilities by greatly reducing seismic load. Seismic isolation is already employed in about 80 common buildings in Japan, but has never been applied in a nuclear plant. Seismic design technical standards have been drafted for nuclear plants and discussed by the ad-hoc seismic committee in the Japanese Electricity Association. For the DFBR, many seismic isolation elements made of laminated rubber shown in Figure 7 are set between the upper base mat and the lower base mat, and steel dampers are installed to quickly attenuate small displacements with low frequency.
The concept of the reactor building applying seismic isolation is shown in Figure 8. Seismic isolation elements greatly reduce horizontal seismic loads, but slightly amplify vertical seismic

Figure 7   Concept of Seismic Isolation System
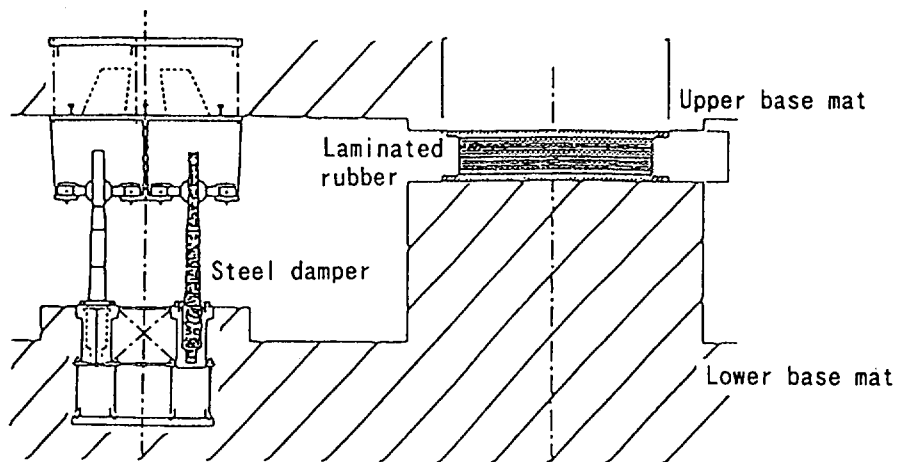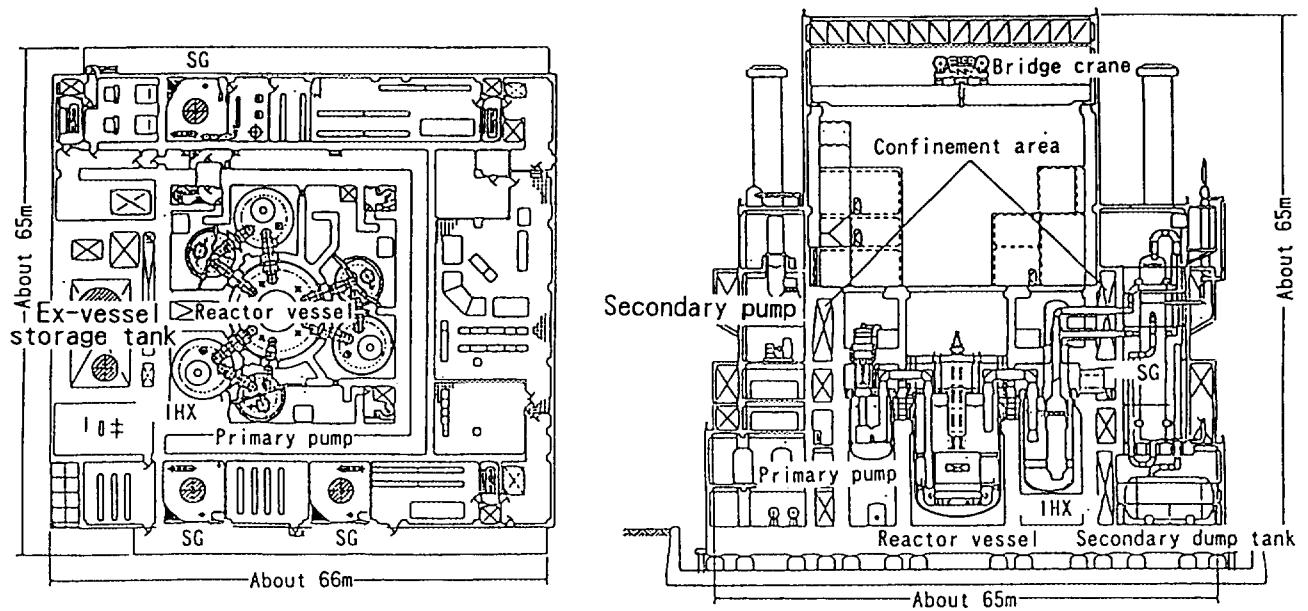

Figure 8   Concept of Reactor Building

vibration. The center of gravity of the building should be so
low as not to add disturbing moment to seismic isolation elements.
The design of piping crossing between the seismic isolation
building and aseismic buildings should accommodate large relative
displacements between the buildings during an earthquake.
The layout of all components has been analyzed in detail to
ensure a route for effective access and maintenance of components
and to reduce the size of the building.

8 CONCLUSIONS

The conceptual design of the DFBR outlined in this paper is based
on the results of the DFBR design study of JAPC. Currently a new
phase of design study is in progress for the three years from
FY1994, based on those results and the basic specification of the
DFBR determined in January 1994. The main issues in this study
are enhancement of core safety, increase of the pressure margin
of containment, feasibility of technical issues and licensability
of seismic isolation plant for introduction to the DFBR. Great
importance has been attached to these issues in the discussion
of the basic specification. Through these studies, the goal to
make the design of the whole DFBR plant harmonious with both
safety and economic viability and to provide preparation for the
basic design.

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PROTOTYPE FAST BREEDER REACTOR MAIN OPTIONS

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Abstract

Fast reactor programme gets importance in the Indian energy market because of continuous growing demand of electricity and resources limited to only coal and FBR. India started its fast reactor programme with the construction of 40 MWt Fast Breeder Test Reactor (FBTR). The reactor attained its first criticality in October 1985. The reactor power will be raised to 40 MWt in near future. As a logical follow-up of FBTR, it was decided to build a prototype fast breeder reactor, PFBR. Considering significant effects of capital cost and construction period on economy, systematic efforts are made to reduce the same. The number of primary & secondary sodium loops & components have been reduced.

Sodium coolant, pool type concept, oxide fuel, 20% CW D9, SS 316 LN and modified 9Cr-1Mo steel (T91) materials have been selected for PFBR. Based on the operating experience, the integrity of the high temperature components including fuel and cost optimisation aspects, the plant temperatures are recommended. Steam temperature of 763 K at 16.6 MPa and a single TG of 500 MWe gross output have been decided. PFBR will be located at Kalpakkam site on the coast of Bay of Bengal. The plant life is designed for 30 y & 75% load factor. In this paper the justifications for the main options chosen are given in brief.

1.0 BACKGROUND TO THE DESIGN OF PFBR

1.1 Energy Demand

The utilities in India generated 351 TWh of electricity during the year 1994-95. The cumulative installed capacity is 81.2 GWe (Coal 52.2, Hydro 28.3, Gas 5.6, Nuclear 2.3 and others 0.3 GWe). The per capita electricity generation works out to 380 KWh/a which is only 1/7th of the world average. There is an urgent need to increase the energy generation for rapid industrialisation to improve the living standard of the large population. The growth of electricity generation during the last 30 years has been on an average 8% and it is likely to continue to grow in the coming decades. See Fig 1. However, there is a gap of about 15% between demand and supply of electricity in the country. The total installed capacity is expected to be about 120 GWe by the year 2000. The efforts in increasing the installed capacity should be complemented by efforts in reducing the transmission and distribution losses (23%), improving the capacity factors of the power plants (current average 50%). India is spending Rs 190 billion/a (6 billion US $) for import of oil and this is the single largest burden on the foreign exchange reserves. This is the price being paid for lack of a clearcut policy on energy independence. The growth of the installed capacity is limited by the available financial resources. So far, generation, transmission and

distribution of electricity was a Government activity. With the economic liberalisation policy started in 1991 it is being privatised - domestic and foreign. Foreign investment is increasing on a big scale in the power industry.
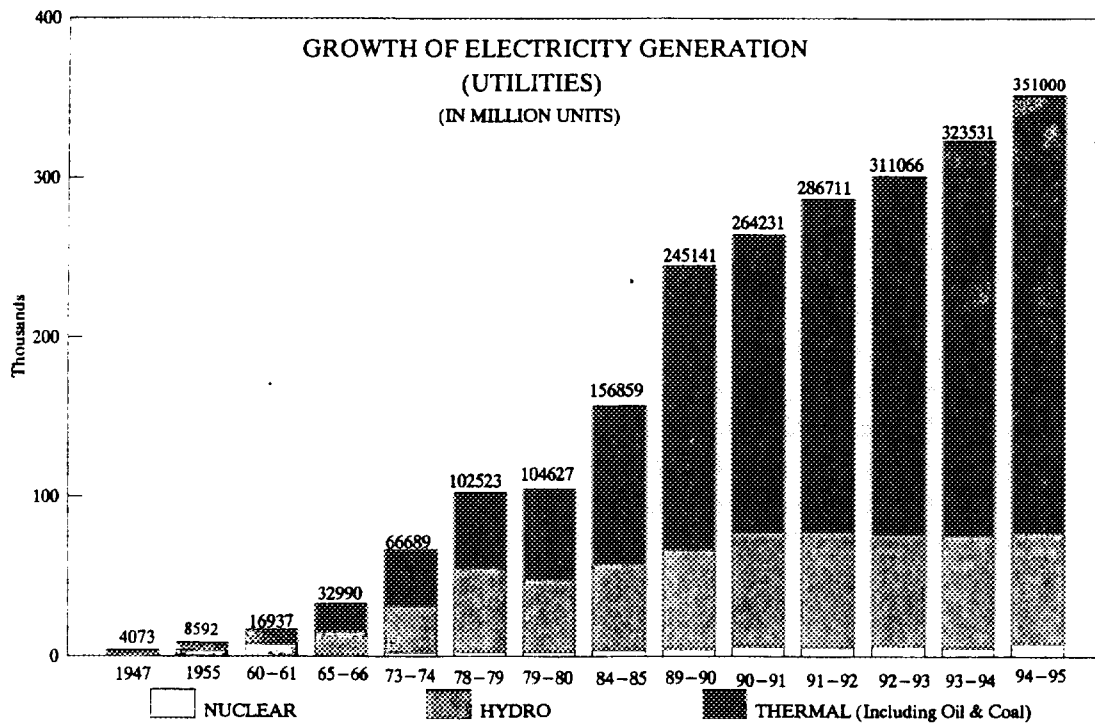
Fig 1

1.2 Energy Resources

Large energy resources available in the country are coal and nuclear energy through FBR. See Fig 2. Coal would not last for more than 50 years for a targeted electricity generation capacity of 500 GWe. Therefore, it is imperative that FBR should be introduced on a commercial scale at the earliest so that by the year 2050, dependence on coal will be very much reduced. Indigenous Uranium reserves are estimated to be about 50,000 t which can support 10 GWe PHWR and about 300 GWe FBR.

Transportation of poor quality coal (~ 40% ash) over long distances (~ 1000 Km) is going to be difficult when the generation capacity increases. Environmental concern - greenhouse effect and acid rain - is important for decreasing the production of electricity from coal. The cost of electricity produced from coal will increase at a rapid rate due to deep mining, transportation over long distances and stringent environmental requirements on the discharges. Gas and Oil cannot make important contributions in the generation of electricity, although gas is making rapid progress in the present condition because of the short gestation periods. The potential hydroelectric power is about 80 GWe at 40 % load factor while 30 GWe has been realised so far. There may be difficulties in further realisation because of the prospective sites being located in

72

difficult terrains and related environmental concerns. Apart from being costly, the renewable energy resources like Solar, Wind, Ocean and Biomass can only make a small contribution in the total generation of electricity in the country.
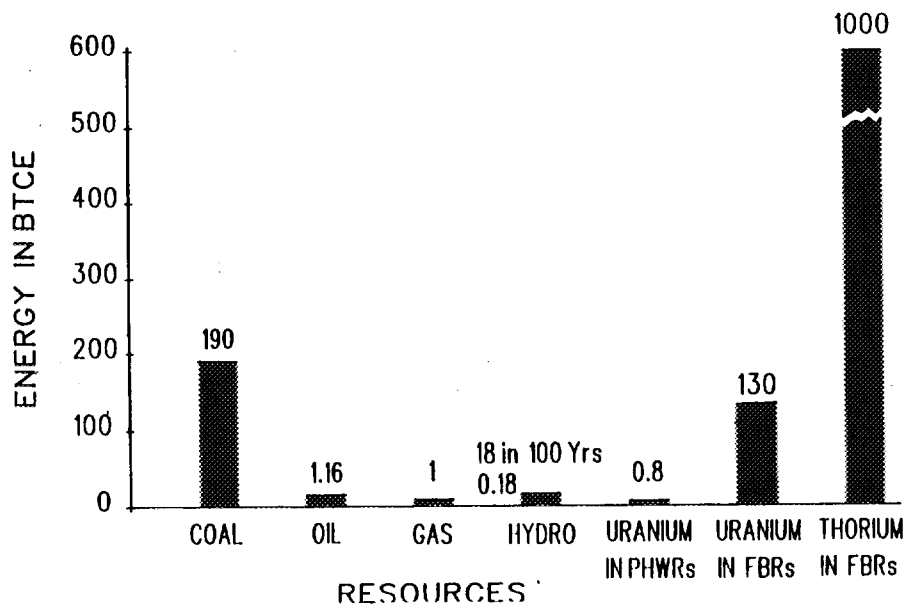
Fig 2

1.3 Thermal Reactor Programme

India is operating 10 power reactors of 200/220 MWe capacity each which have logged 110 reactor-years of operating experience (2 BWR and 8 PHWR). 4 x 220 MWe and 2 x 500 MWe PHWR units are under construction. The indigenous content of the recent reactors has been very high at 85 to 90% and the Indian industries have risen to the occasion in taking up manufacture of high quality components for the nuclear industry. In order to be economically competitive with the coal fired power stations, it is essential to reduce construction time and increase the capacity factor of nuclear power stations.

1.4 Fast Reactor Programme

India started its fast reactor programme by construction of 40 MWt Fast Breeder Test Reactor (FBTR) . Most of the components of the FBTR have been manufactured by the Indian industries. The reactor attained its first criticality in October 1985 and its rated power of 10.5 MWt in December 1993 with the Mark I small core. Though the performances of the various systems have been generally satisfactory with respect to the ratings, the reliability of operation has to be increased considerably. Design and construction deficiencies were observed and the same have been removed by systematic analyses and followup actions.   Generation of electricity at about 2.5 MWe is expected from the Mark I core in the coming months. 600 mm of Hg vacuum has been achieved in the condenser during commissioning test. With the change to Mark II full

size core in 1996, the reactor power will be raised to 40 MWt. FBTR will be used as an irradiation facility for fuel development and to gain operating experience.

1.5   FBR Cost Competitiveness

The development of FBR was takenup very vigorously in USA, Europe and erstwhile USSR in the 60's and 70's because of the following two reasons,

- FBR is a large energy resource.
- FBR produce electricity more economically than thermal reactors.

Construction experiences of BN 600, SUPERPHENIX and MONJU have indicated that FBR are costly by a factor 2 to 3 compared to PWR and considerable cost reductions are essential for their commercialisation.

Any large power plant must fulfill the following in order to be competitive with the alternative energy resources.

- Low capital cost
- Short construction period
- High capacity factor

The components of UEC from FBR can be approximated as shown in Table I. Reduction in the Fuel Cycle Cost reduces UEC only marginally.

Table I

Capital	O & M	Fuel Cycle
75 %	15 %	10 %

Since the capital cost forms the major component in the UEC, efforts must be made to decrease the same. Large capital cost reductions have been demonstrated in the designs of AP 600, CANDU 3, EFR, BN 600 M by design simplification. The low capital cost not only reduces UEC but also decides the quantum of capacity that can be added from a given amount of finance allotted. Lower the capital cost, higher will be the growth rate.

As in any developing country, India is finding scarcity of capital particularly for power industry where the investment is large and returns are spread over a long period. The present interest rates are 12% on Government borrowing and about 18% on the market borrowing compared to about 8% in the developed countries. The high interest rate becomes one of the

demanding factors for reducing the construction period. Otherwise, the Interest During Construction becomes a major investment cost and reduces the competitiveness of the project. The long gestation period in the construction of the Nuclear Power Plants is one of the eroding factors in cost competitiveness. The effect of capital cost, construction time & interest rates on UEC can be seen from the following.

$$= DCC + ICC + EDC + IDC + OMC + FCC$$

Components $= DCC + K_{ICC}.DCC + K_{EDC.}.DCC + K_{IDC}.DCC + 0.03(1+e)^n (1+K_{ICC})DCC + FCC$

of $= DCC + K_{ICC}DCC + K_{EDC}DCC + K_{IDC}DCC + 0.04(1+e)^n DCC + FCC$

UEC $= DCC.(1 + K_{ICC} + K_{EDC} + K_{IDC} + K_{OMC}) + FCC$

$$= DCC. K_{TOTAL} + FCC$$

where, DCC - Direct Capital Cost, ICC - Indirect Capital Cost, EDC - Escalation During Construction, IDC - Interest During Construction, OMC - O&M Cost, FCC - Fuel Cycle Cost, n - number of years and e - escalation rate.

The reduction in DCC would result in reduction of ICC, EDC, IDC and OMC.

2:1 Debt/Equity ratio is assumed in the calculations invloved in the following table which shows clearly the effect of construction time and interest rates on UEC.

Table II :

No of years	Interest rate = 8 % & Esc. rate = 4 %					Interest rate = 16 % & Esc. rate = 8 %				
	K_{ICC}	K_{EDC}	K_{IDC}	K_{OMC}	K_{TOTAL}	K_{ICC}	K_{EDC}	K_{IDC}	K_{OMC}	K_{TOTAL}
6	0.35	0.13	0.13	0.05	1.66	0.35	0.27	0.26	0.06	1.95
8	0.35	0.17	0.17	0.05	1.75	0.35	0.38	0.37	0.07	2.17
10	0.35	0.22	0.22	0.06	1.86	0.35	0.50	0.49	0.09	2.42
12	0.35	0.27	0.27.	0.06	1.96	0.35	0.62	0.62	0.10	2.70

Construction of a NPP takes 8 to 10 years and when compounded with the high interest rates leads to high IDC. 50% of the investment cost of the Kaiga Project (2 x 220 MWe PHWR) is due to IDC. Apart from Unit Energy Cost (UEC) consideration, the shorter construction periods make a rapid growth of installed capacity and thereby the national economy. Therefore, it is highly desirable to plan to minimise the construction period, even at the conceptual design stage. This can be done by improving the infrastructure and reducing the number of systems and components and their sizes.

By the time FBR is commercialised in India, further construction of PHWR would not be viable due to limited availability of U and the only competition for the FBR would be from coal fired power stations.

We are proposing to the government to fully finance the PFBR.

2.0 Design Objectives of PFBR

- The purpose of constructing PFBR is to demonstrate on an industrial scale techno-economic viability of an FBR Power Plant in India.

- The concepts selected for PFBR should be based on the operating experiences of FBR. Any innovative design concept is to be incorporated only after thorough research and prototype test. Atleast 4 reactors of 500 MWe capacity are likely to be built after PFBR. Therefore, the designs must be optimised and standardised so that the follow on plants will have minimum modifications.

- High breeding ratio is not an essential requirement of PFBR Fuel cycle. The growth of FBR initially will depend on availability of technology, availability of finance, public acceptance and then on Plutonium. PFBR will be a breeder reactor but without emphasis on high breeding. Economics of fuel cycle is an important factor at this stage of FBR development.

- Experience in design, construction and operation of all the FBRs, particularly all the incidents and accidents, shall be considered systematically in PFBR design.

- The reactor shall meet the PFBR safety criteria issued by Atomic Energy Regulatory Board (AERB).

- Design simplification i.e. reduction in number of systems and components without compromise on safety and reliability is to be carried out. Experience of thermal reactors indicates that complex plants take longer times to construct and are difficult to operate.

In this paper the main options selected for PFBR are discussed. Options of core, main heat transport systems, core component handling systems and safety are presented in the companion papers of this meeting.

3.0 MAIN OPTIONS

3.1 Sodium Coolant

Sodium has been the unanimous choice of all fast reactors due to the following.

- Nonmoderating properties essential for FBR

- High boiling point permits high temperature without pressurisation. This results in high thermodynamic efficiency and thin walled components.

- High thermal conductivity of sodium results in high heat transfer coefficients, even at low velocities. Heat transfer areas are low. Passive decay heat removal is possible due to high heat transfer, large ΔT and low viscosity.

- No corrosion of structural materials at high temperature over long periods of operation.

- Large margin between operating temperature ($\sim$ 823 K) and high boiling point ($\sim$1155 K) gives sufficient safety margins for heat removal in emergency conditions.

- Owing to extensive R&D and reactor operating experience over 4 decades, high level of technological maturity has been reached.

The disadvantages of sodium are the following:

- High radioactivity of primary sodium, deposition of sodium vapour in cold region of covergas, cleaning of sodium from components can be taken care of by proper design.

- Opacity of sodium, its chemical reaction with air and relatively high melting point (371 K) make inservice inspection and repair difficult. Very careful considerations have to be given in the conceptual design and development of special devices needed for this purpose.

- Large leaks of sodium can result in fires and Na-concrete reactions. Provision for diverse sodium leak detection systems, sodium collection trays, selection of suitable concrete, steel lining, etc. can mitigate the problem.

- Large sodium water reaction in SG: High quality of manufacture of SG, sensitive leak detection system, isolation of SG on detection of small leaks can avoid this problem. Also by proper design of secondary sodium circuit, this can be taken care of.

3.2 Pool type Concept

The loop and pool concepts have been discussed qualitatively in several forums. Both types are in practice. We have decided pool concept for PFBR.

The main advantages of pool type are:

- Simple shape of reactor vessel without any nozzles and low neutron radiation dose, results in high reliability and its easy inservice inspection and repair.

- Large thermal inertia of the pool attenuates thermal shocks, results in slow temperature rise during decay heat removal and load throw conditions, and gives long times for operator actions.

- capability to withstand higher work potential under core disruptive accident.

- containment of radioactive components and fluids is easy.

- Compact primary sodium circuit layout.

The disadvantages of pool type are:

- Thermal hydraulics of hot and cool pools is complex. Specially developed computer codes and several scaled down water models can solve these problems successfully. Operation of EBRII, PHENIX, PFR, BN600 and SUPER PHENIX have demonstrated satisfactory performance.

- Large size of vessel needs site assembly. The vessels are assembled in site workshop by welding only. Hence same quality as in shops can be achieved without any additional time for manufacture.

- There is concern about seismic design of thin shell vessels with large mass of sodium. Preliminary analysis indicates that the structural integrity can be assured for moderate seismicity.

- There is more interdependence in component designs . The designs of primary sodium pump, IHX, inner vessel, roof slab get interconnected very much. 2-3 iterations can give good judgement for design decisions.

- Difficulty of maintenance on top of pile: Due to closeness of many components on reactor assembly resulting in space constraints can prolong maintenance works, thus affecting capacity factors. This can be taken care of by reducing the number of components and making them compact. Thorough check on drawing boards for lack of interference and adequate space for component handling and full scale sector mock up can eliminate such problems.

- Measurement of flux in startup range and primary sodium flow require special instrumentations.

3.3   Reactor Power

Reactor power is the most important decision in the design of India's prototype fast reactor. 200-300 MWe capacity, as has been the case for all the FBR countries would appear an obvious choice to avoid large extrapolation from FBTR (15 MWe). However, the following considerations lead to select 500 MWe capacity .

- Large size pool type reactors have operated in other countries upto 1250 MWe and basically there are no technological problems. Such a confidence was not there in the late 60's.

- Specific capital cost is lower for 500 MWe than lower power, say 250 MWe (~30 %).

78

- Medium size power is very much desirable for constructing more number of follow on plants, before a large commercial size plant is built.

- Coal fired power plants and PHWR of 500 MWe capacity have been designed. The coal fired plants are in operation and PHWR are under construction. The conventional power equipment of this size, particularly TG set, are readily available.

- Design and development efforts for 250 MWe and 500 MWe plants are comparable.

- Constructability of 500 MWe size components was assessed based on the experience of 220 and 500 MWe PHWR and FBTR. The Indian industries are equipped with necessary machines. High quality SS welding can be done if attention is paid to training of welders and taking care of weld shrinkages. Transportation of components except reactor vessels and roof slab is possible. The vessels and roof slab will be assembled in site workshop. Development of manufacturing technology is in progress to gain industrial experience. Delivery schedules need considerable improvements.

- Financial risk is of course more for a 500 MWe plant than a smaller plant. This would be reduced by systematic design and development.

3.4 Fuel

PuC-UC has been used as fuel for FBTR due to nonavailability of enriched uranium for mixed oxide option. For PFBR enriched uranium is not required. Though carbide gives high breeding ratio, it raises safety problems in fabrication because of its pyrophoricity. Fabrication cost is also high. Fuel burnup is lower compared to oxide because of its high swelling rate. Reprocessing on prototypic scale has not yet been done anywhere and this cost is also expected to be more. Being a large power plant, proven fuel cycle is essential. High breeding is not the objective for PFBR. As the entire plant must be designed around the fuel, a firm and early decision is essential. Most of the large size FBR use MOX fuel. This choice was natural since the technology of mixed oxide fuels is very similar to that of UO_2, which is used in thermal reactors. MOX fuel has shown excellent performance in all FBR with respect to high burnup (up to 2,00,000 MWd/t on full size subassemblies) and has proven reprocessing technology. A large amount of safety experimental results is available. Extensive experience is also available in India from thermal reactors. After thorough debating on choice of fuel, MOX has been decided.

3.5 Main Structural Materials

20% CW D9 material is selected for cladding and hexcan because of its improved resistance against swelling due to neutron irradiation, high strength at operating temperature and

good corrosion resistance against Na and fuel. AISI 316M has been used in FBTR for sodium components except SG. It has given very satisfactory results. We like to continue its use with some improvements. SS 316 LN gives improved corrosion resistance while maintaining high temperature strength. Hence it is selected for out of core sodium components . Use of SS 304 LN for cold leg sodium components is being discussed. Modified 9Cr-1Mo steel has been selected for SG because of its adequate high mechanical strength, freedom from the risk of stress corrosion cracking (problem with stainless steels) and also decarburization (problem with 2.25Cr-1Mo).

3.6 Operating Temperatures

High reactor outlet temperature is always preferred for achieving high thermodynamic efficiency. However, this is limited by the fuel burnup and component structural integrity considerations. In order to satisfy the allowable clad hotspot temperature of 973 K, the reactor outlet temperature is to be limited to 833 K for core ΔT of 150 K. As regards structural integrity of high temperature components, with the recent advancements in high temperature design codes and structural analyses methodology, it is possible to select as high as 825 K for the reactor outlet. Detailed inelastic and viscoplastic analyses have been performed for the control plug, inner vessel and IHX using ORNL and Chaboche viscoplastic models. While the permissible reactor outlet temperature is about 770 K in order to satisfy the design rules of RCC-MR through 'elastic' route, the viscoplastic analysis indicates that the temperature can be 825 K. Modified. 9Cr-1Mo can permit upto 775 K steam temperature. The turbines used in the conventional thermal power stations allow steam temperature of 811 K. The reactor inlet temperature and hot & cold temperatures of the sodium in the secondary sodium circuit are arrived at from overall cost optimisation studies.

Based on all the above considerations and discussions with the supplier of turbine for steam reheat cycle, the following plant temperatures have been selected.

- Core inlet/outlet : 670/820 K
- Primary sodium inlet/outlet to IHX : 817/667 K
- Secondary sodium inlet/outlet to IHX : 628/798 K
- Feed water inlet/steam outlet : 508/766 K
- Steam conditions : 763 K at 16.6 MPa

3.7 Number of TG Set

Turbine is the most important component, on the conventional side of a NPP. It is also the most costly equipment. It operates at high pressure and high temperature and rotates at high

speed. It has large number of parts and auxiliaries. Reliable TG is essential for a high capacity factor. In India, the TG sets are manufactured by M/s Bharat Heavy Electricals Ltd. As in most of the countries, the present day large size turbines are modular in design and can give the desired power output with the specified steam conditions by combination of proven modules having high reliability. Reliable turbines are available for sodium to steam reheat as well as steam to steam reheat cycles. Presently, there are 15 TG sets of 500 MWe capacity operating in India, for the coal fired power plants. Their reliability is very high (~ 99% availability excluding planned maintenance times).

Use of a single TG set in place of two saves about 30% of total TG cost and reduces plant maintenance. Therefore, single turbine of 500 MWe gross output is selected for PFBR after detailed discussions with the supplier.

3.8   Plant Life

Longer life with higher plant capacity factor is desirable for economic production of electricity. Plant life is mainly limited due to radiation and creep-fatigue damage. Considering the selected materials, analyses capability and maturity of design codes, a 30 y life and 75 % capacity factor is used for the design.. Some of the components, such as SG, control rod drive mechanism, cold traps, may have to be replaced. However, for the calculation of UEC, 25 y plant life with 62.8 % capacity factor is used based on the costing procedure .

4.0   Site Selection

PFBR will be located at Kalpakkam site on the coast of Bay of Bengal for the following reasons:

- Closeness to the design office, R&D laboratories and FBTR
- Site of low seismicity
- Infrastructure availability
- Kalpakkam is away from coal fields
- Electricity demand of this region.

Kalpakkam will be a part of the Southern Region Grid, the current capacity of which is about 15 GWe and expected to reach 35-40 GWe by the year 2005.

5.0   SUMMARY

Demand for energy will continue to grow for decades. Large energy resources available in India are coal and nuclear energy through FBR. Apart from reducing the capital cost, it is

essential to plan to minimise the the construction time even at the conceptual design stage for commercialisation of FBR. Sodium coolant, pool type, 500 MWe, oxide fuel, 20 % CW D9, SS 316 LN & Modified 9Cr-1Mo materials have been selected for PFBR. Steam temperature of 763 K at 16.6 MPa and a single TG of 500 MWe gross output have been decided.

CONCEPTUAL DESIGN OF PFBR CORE

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Abstract

The design options selected for the core of the 500 MWe Prototype Fast Breeder Reactor are presented. PFBR has a conventional mixed oxide fuel core of homogeneous type with two enrichment zones for power flattening and with radial and axial blankets to make the reactor self-sustaining in fissile material. Pin diameter has been selected for minimisation of fissile inventory. Considerations for the choice of number of pins per subassembly, integrated versus separate axial blankets, and other pin and subassembly parameters are discussed. As the core size is moderate, no special schemes for reducing the maximum positive sodium voiding coefficient is envisaged. Two independent, diverse, fast acting shutdown systems working in fail-safe mode are selected. The number of absorber rods has been minimised by choosing a layout for maximum antishadow effect. Nine control and safety rods are distributed in two rows for power flattening by differential insertion. Three Diverse Safety Rods, are also provided which are normally fully withdrawn. The optimisation of layout of radial and axial shielding and adequacy of flux at detector location are also discussed.

1.0 CORE DESCRIPTION

The basis for Prototype Fast Breeder Reactor (PFBR) core design is safe operation, economic power generation and self sufficient fissile material production. With these requirements, the core design is arrived at by numerous

iterations between thermal, mechanical and neutronic analyses. Under Indian conditions, the initial fuel material is derived from plutonium and depleted uranium obtained from spent Pressurised Heavy Water Reactor (PHWR) fuel.

Mixed oxide fuel has been selected for PFBR on account of extensive worldwide experience available with this fuel, proven capability for safe operation to high burnup, ease of fabrication and handling, and proven economic reprocessing. The low breeding ratio and long doubling times for this fuel is not considered a disadvantage in the initial stage on account of the large amount of initial plutonium becoming availiable from the PHWR programme.

A conventional homogeneous type of layout for the core and blankets has been selected. Radial heterogeneous configuration, inspite of advantages of higher breeding ratio and reduced sodium void coefficient, was not considered due to increase in Pu enrichment, higher fissile inventory, larger overall core size, reduction in Doppler coefficient, increased thermal striping protection requirements for above core structures, and possible difficulty in achieving optimum neutronic coupling between core zones without extensive experimental studies. However, the use of axial heterogeneity is retained as an option for the future for some improvement in breeding ratio, reduction in sodium void coefficient and lower displacement damage fluxes.

The active core consists of 181 fuel subassemblies of which, 85 are in the inner enrichment zone with 21% PuO_2 content and 96 are in the outer enrichment zone with 28% PuO_2 content. There are three rows of radial blanket subassemblies followed by one row of steel reflector and one row of intermediate shielding

84

boron carbide subassemblies outside of which are the internal fuel storage locations and then the radial bulk shielding subassemblies of steel and boron carbide. There are twelve absorber rods arranged in two rings of which nine constitute the primary Control and Safety Rod (CSR) system and three constitute the secondary Diverse Safety Rod (DSR) system. Fig.1 gives the cross section of the core layout and Table 1 summarises the main parameters of the core.

Each fuel subassembly consists of 217 helium bonded fuel pins of 6.6 mm outer diameter using 20% cold worked D9 alloy cladding separated by helically wound spacer wires giving a pitch ratio of 1.25. Each pin has a 100 cm column of mixed oxide fuel, 30 cm each of upper and lower depleted UO_2 blanket columns and upper and lower fission gas plena. The axial blankets are integrated within the fuel pin. Fig.2 gives the section of the fuel pin and subassembly.

The core restraint against thermal and swelling induced outward movement of subassemblies is provided by spacer pads on the fuel, blanket and reflector subassembly sheaths, at axial position at the middle of the top blanket. This concept has the advantage of moderate interactive forces between subassemblies and permits "flowering" of the fuel subassemblies to some extent to contribute to the negative power coefficient.

2.0 FUEL PIN DIAMETER AND MAXIMUM LINEAR PIN POWER

The choice of fuel pin diameter significantly affects the fissile inventory, fuel cycle cost and the breeding gain. Parameteric studies made for PFBR indicate that the optimum pin diameter from fuel inventory doubling time as well as fuel cycle
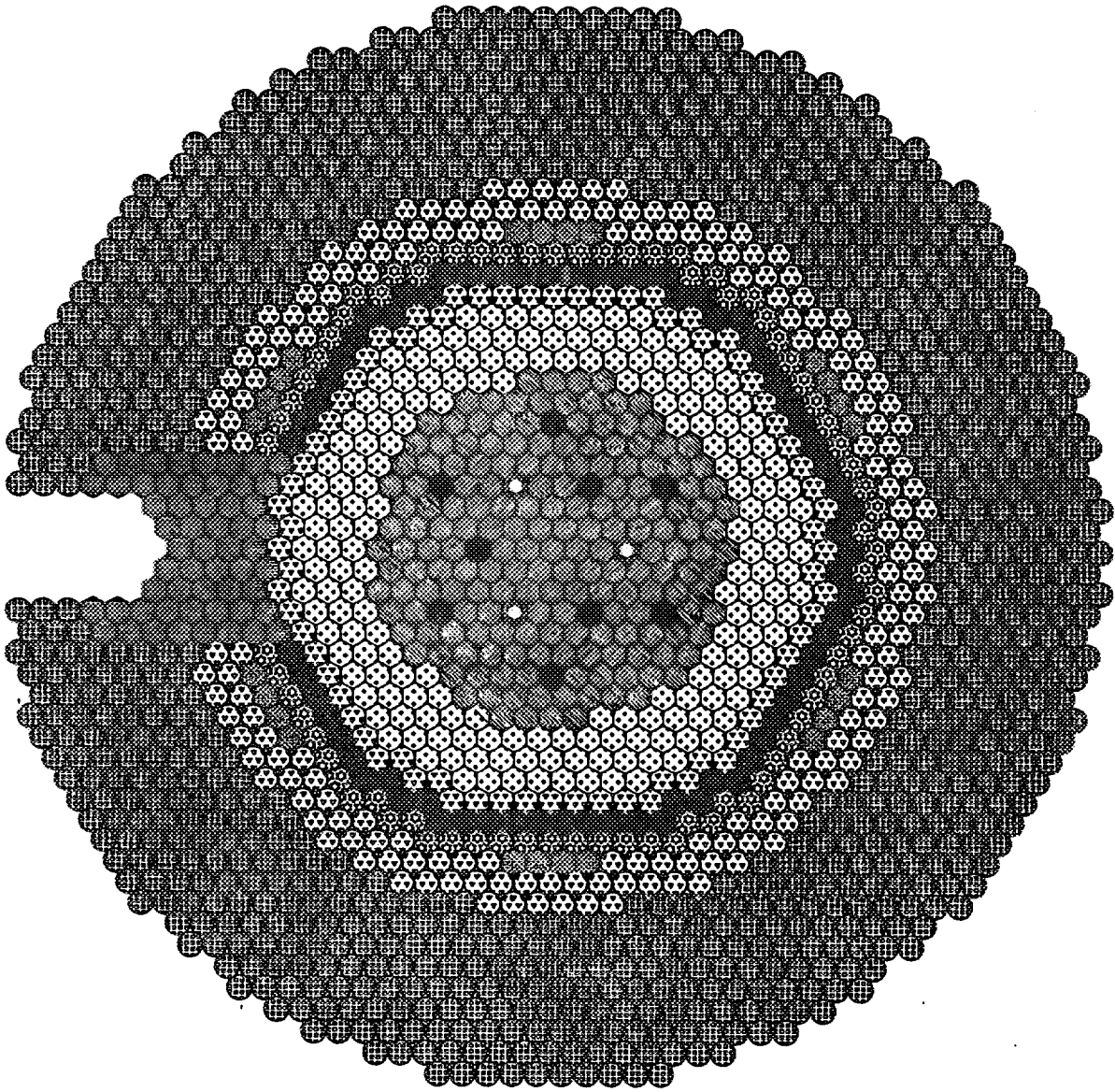
SYMBOL	TYPE OF SUBASSEMBLY	No.	MASS PER SUBASSY. IN Kg
⬡	FUEL (INNER)	85	245
⬡	FUEL (OUTER)	96	245
⬢	PRIMARY CONTROL ROD	9	200
⬡	SECONDARY CONTROL ROD	3	200
⬡	BLANKET	186	320
⬡	STEEL REFLECTOR (INNER)	72	355
⬢	B$_4$C SHIELDING (INNER)	69	185
⬡	STORAGE LOCATION	75	245
⬡	RESERVE STORAGE LOCATION	24	355
⬡	ENRICHED BORON SHIELDING	56	185
⬡	STEEL SHIELDING (OUTER)	180	330
⬡	B$_4$C SHIELDING (OUTER)	903	265

FIG. 1. PFBR core configuration.

SECTION—AA

217 FUEL PINS

INNER HEX A/F 124·9 ±0·3

A/F MAX 131·6

AT BUTTONS A/F 134·7 ±0·05

TRIANGULAR PITCH 8·25 (NOMINAL)

SECTION—BB

SPACER WIRE ∅1·65 WOUND HELICALLY AT PITCH 150.

CLAD O.D. 6·6 I.D. 5·7 THK 0·45

PELLET ∅ 5·55

SECTION—XX

HEAD

ADAPTOR

TOP SHIELD

FUEL PIN BUNDLE.

TOP BLANKET

FUEL

BOTTOM BLANKET.

∅100 CIRCULAR LINE CONTACT ON GRID PLATE SLEEVE

FOOT

COOLANT ENTRY SLOTS

2 HOLES ∅3

FUEL PIN ASSEMBLY

FIG. 2. PFBR subassembly.

TABLE 1. CORE DESIGN PARAMETERS

Core thermal power	1250 MWt
Maximum linear pin power	450 W/cm
Fuel pin outer diameter	6.6 mm
Active core height	100 cm
Equivalent core diameter	199 cm
No. of pins per fuel subassembly	217
Fuel smeared density	82.5% TD
Maximum core ΔT	180°C
Maximum core ΔP	55 m Na
Subassembly pitch	135 mm
Sheath thickness/Subassembly size	3.2/131.3 mm
Fuel pins clad thickness	0.45 mm
Axial blanket thickness (each)	30 cm
Core volumetric fractions	
Fuel/Gap/Sodium/Steel	33/2/41/24 %
Maximum neutron flux	8×10^{15} n/cm^2-s
Breeding ratio	1.07
Effective delayed neutron fraction	340 pcm

cost considerations for mixed oxide fuel is 8 to 9 mm depending on the out-of-pile cycle time. However, based on initial plutonium availability, it is decided to restrict the inpile plutonium inventory to two tonnes and consequently the pin diameter is fixed as 6.6 mm, which also has the advantage of smaller core size and lower sodium void coefficient. However, the choice of smaller pin diameter and consequent lower fuel volume fraction has the disadvantages of reduced breeding ratio, reduced core conversion ratio (and hence increased CSR worth for

burnup compensation and increased power swing with burnup), reduced Doppler coefficient, reduced fuel residence time and increased fuel cycle cost.

Irradiation testing and basic fuel properties measurements for PFBR fuel composition have not yet been made, but with available data and appropriate uncertainty margins, a maximum linear pin power of 450 W/cm is conservatively fixed.

3.0 SUBASSEMBLY SIZE

Selection of the number of pins per fuel subassembly depends on several factors. A safety consideration is that an individual fuel subassembly must remain sufficiently subcritical under immersion in water. Further, high mechanical loading due to weight and high thermal loading due to decay heat becomes increasingly difficult to be taken into account in the design of the fuel subassembly handling and transport mechanisms. Having fewer number of pins per subassembly favours increased core average coolant outlet temperature and increased average discharge burnup. However, these advantages must be balanced against the disadvantages of having to load/unload larger number of subassemblies at each shutdown, increased number of locations in the grid plate and increased number of core monitoring positions (for outlet temperature and failed fuel). Taking into account the above considerations 217 fuel pins per subassembly has been selected leading to a subassembly size of 131.3 mm (outside distance across flats). For this choice the fuel handling equipment are designed for a maximum extraction load of 15000 N and decay power of 5 kW per subassembly.

4.0 ENRICHMENT ZONING

Power flattening by enrichment zoning permits reduction in fissile inventory for given power output. For cores of PFBR size, two enrichment zones are adequate to achieve the desirable power flattening. Parameteric study made varying zone volume ratio V_1/V_2 from 40/60 to 80/20 showed that for the fresh core the power form factor varied little for V_1/V_2 from 50/50 to 70/30 (less than 2%) with the optimum at 60/40. However, it was found that the stability of the form factor with burnup was better for values of V_1/V_2 less than 50/50. The present selection of V_1/V_2 is 47/53.

5.0 FLOW ZONING

Subassembly outlet temperature flattening by flow zoning reduces thermal striping and increases the mixed mean outlet temperature which improves efficiency. Based on parametric studies the number of flow zones recommended are seven for the fuel, four for the blanket and one each for the absorber, reflector, inner shielding and storage subassemblies.

6.0 ACTIVE CORE HEIGHT

The advantages of small core heights are increased fuel volume fraction for same coolant pressure drop, reduced subassembly length, easier fuel fabrication and reduced sodium void coefficient, but the disadvantages are increased fissile inventory, larger number of pins and larger core radius. The penalty in Pu inventory was calculated as a function of H/D ratio. A limit of 10% was placed on fissile inventory penalty compared to optimum H/D resulting in a lower limit of 0.4 for H/D ratio. Present selection of 100 cm core height has a H/D ratio of 0.5.

Variation of core height from 80 cm to 120 cm was studied by appropriate thermal-hydraulic-neutronic computations. For fixed pin diameter, core pressure drop and core temperature rise, increase of core height decreases fuel volume fraction affecting neutron economy, but at the same time improves H/D ratio leading to reduced neutron leakage and better neutron economy. These two opposing consequences result in optimum core height from fissile inventory and breeding gain considerations to lie between 90 and 105 cm. The study of fuel cycle cost as a function of core height was difficult to make as fabrication cost data as a function of pin length is uncertain but it appeared that the fuel cycle cost was relatively insensitive to the core height around 100 cm.

Further study was made after selection of subassembly size keeping coolant pressure drop and subassembly pitch constant. In this study, other parameters such as pin diameter, clad thickness, pitch ratio, inter-subassembly gap and so on were allowed to vary. On account of increased fuel volume fraction the 80 cm core compared to the 100 cm showed marked improvement in breeding ratio and doubling time at the cost of 6% increase in inpile Pu inventory and nearly 25% increase in the number of fuel subassemblies. The 120 cm core showed 15% reduction in number of subassemblies and 7% reduction in inpile Pu inventory, but marked deterioration in breeding ratio and doubling time. The sodium void coefficient varied in a complex manner with core height on account of simultaneous change of core dimensions and sodium volume fraction. However, the overall change in sodium void coefficient was of the order of only 2%.

7.0 LOCATION OF FISSION GAS PLENUM AND AXIAL BLANKETS

Location of fission gas plenum at the top of the pin is considered desirable in order to prevent the possibility of fission gas passing through the core when there is a fuel pin failure. However, due to higher temperatures a top plenum becomes much longer than a bottom plenum for the same stored gas pressure and mass. Since the relative safety is not much different, for PFBR, the major fission gas storage is in a bottom plenum (71 cm) with a small amount of storage in a top plenum with spring (20 cm).

Based on fuel residence time, buildup of plutonium, coolant pressure drop and fuel fabrication considerations, an axial blanket thickness of 30 cm on each side has been selected. the effect of having axial blanket pins separate from the fuel pins was studied as it leads to lower fuel pin fabrication costs due to shorter lengths and fewer number of blanket pellets. It was found that on account of choice of mainly bottom plenum storage, the separation of the top axial blanket does not lead to much breeding penalty, whereas separation of the bottom axial blanket does lead to considerable reduction in breeding gain. Since overall subassembly length is smaller and subassembly fabrication is simplified, the choice of having both blankets integrated in the fuel pin has been made.

8.0 RADIAL BLANKET THICKNESS

Study of the variation of blanket breeding ratio with radial blanket thickness indicated 90% saturation at a thickness of 40 cm. Consequently, the number of radial blanket rows is fixed at three. A study for reduction of the number of blanket rows to two showed that in this case there is 0.04 reduction in

breeding gain and the reactor may not be a net producer of plutonium in case cycle losses exceed 1%.

9.0 REACTIVITY COEFFICIENTS

The inlet temperature coefficient and the power coefficient are overall negative as for other fast reactors of this type with the dominant negative component coming from the Doppler effect in U-238. The positive contribution from the coolant expansion coefficient is smaller and delayed compared to the negative contribution from the Doppler effect and fuel expansion.

No special design provisions have been made to increase the Doppler coefficient or reduce the sodium void coefficient which is about 3.5 dollars for the reference core. As compared to the case of the 8 mm or greater pin diameter, the 6.6 mm pin diameter reference core has marginally smaller Doppler coefficient but also substantially smaller positive sodium void coefficient (lower by over a dollar). However, the effect of voiding leads to a greater reduction in Doppler coefficient for the 6.6 mm pin diameter case as compared to the larger pin diameter cases.

As already stated, decreasing the core height to 80 cm from 100 cm caused little reduction in sodium void coefficient but considerably increased the core size and inpile fissile inventory and was hence not considered.

10.0 REFUELLING INTERVAL

This is an important parameter which affects the core characteristics in several ways. Shorter refuelling interval improves average discharge burnup, reduces excess reactivity and

control requirements, improves breeding ratio, reduces number of internal fuel storage locations, reduces the out of pile inventory and improves the doubling time. On the other hand, longer refuelling interval improves the reactor availability and reduces the thermal cycling of the reactor (due to shutdown). There is also an upper limit on refuelling interval which comes from the design peak fuel burnup. It is necessary to choose the (fixed) refuelling interval as a simple fraction (1/2, 1/3 etc.) of the residence time of the peak rated subassemblies to prevent gross reduction in the average discharge burnup.

Studies showed that for a peak discharge burnup of 100,000 MWd/t the suitable refuelling interval is 180 full power days. As initially the fuel burnup will be limited to 50,000 MWd/t the refuelling interval is correspondingly reduced to 90 full power days.

Based on the above choice of refuelling interval the excess reactivity and control requirements, the fuel management scheme and the number of invessel fuel storage locations have been arrived at.

11.0 REACTIVITY CONTROL SYSTEM

The core is protected by two independent shutdown systems each capable of bringing the reactor to cold shutdown. The number of rods in each shutdown system is sufficient to ensure cold shutdown by that system even with its most reactive absorber rod unavailable. There are nine CSRs in the first shutdown system and three DSRs in the second shutdown system. The twelve rods of the reactor are distributed in two rows to maximise the antishadow effect and to minimise the required number of rods.

In order to avoid common mode failure and to achieve diversity, the design of both the systems is in such a way that parts related to safety action are entirely different. There is a mechanical coupling between the mobile assembly of the CSR Drive Mechanism and CSR, whereas there is an electromagnetic coupling between the DSR Drive Mechanism and DSR. Similarly, the rods of each shutdown system are of different design. Two independent plant protection logic systems provide trip signals to the corresponding shutdown system and independent and diverse control system for each shutdown system is provided. The drop time of the rods under emergency conditions is less than a second.

For both CSRs and DSRs, enriched boron carbide is chosen as the absorber material on account of high nuclear cross section, ease of fabrication, availability and good experience. Study of the use of Eu_2O_3 and EuB_6 (natural) was made to avoid the use of B-10 enrichment. However, Eu_2O_3 was only as good as natural boron and EuB_6 (natural) was as good as 30% enriched boron. Since, over 50% enriched boron is required the choice of boron carbide was retained.

The nine CSRs are in both the rings to enable power flattening adjustment by differential insertion. All the CSRs have the same B-10 enrichment and their individual worths depend on their position. Speed of movement is sufficiently low (2 mm/s i.e. maximum 1 cent/s) to enable desired control and hence, separate regulating rods of low worth are not provided. The minimum total worth of CSRs is 8000 pcm. Enough additional worth is provided for calculational uncertainty and burnup.

Three DSRs of equal reactivity worth are in the inner ring. The total minimum reactivity worth of these rods is 3000

pcm. Enough additional worth is provided for calculational uncertainty and burnup. DSRs are not used for reactor control and are in fully raised position during reactor operation.

There is 6000 pcm of core excess reactivity (1700 pcm power and temperature defect, 3100 pcm BOEC to EOEC reactivity swing, 600 pcm operating margin and 600 pcm uncertainty margin). The emergency shutdown reactivity available for scram is over ten dollars and can handle postulated incidents like loss of flow, transient over power, loss of regulation, fuel melting and slumping in a few subassemblies and sodium voiding in a few subassemblies. The shutdown margin in the fuel handling state is 5000 pcm which can cater to hypothetical postulated errors like removal of any two absorber rods in shutdown state or replacement of absorber rod by fuel subassembly.

The possibility of use of central absorber rod was studied as it decreased the total number of absorber rods to ten and also had the incidental advantage of lowering central sodium voiding reactivity gain. However, in a prototype reactor the central location is an important position for experimental access purposes and it was not considered advisable to block it with an absorber rod.

3800 pcm of reactivity is required to compensate the reactivity loss on going from BOL to BOEC in three cycles. This is not provided by the absorber rods but by diluent subassemblies in the initial fresh core which are progressively unloaded during the approach to equilibrium.

12.0 IN-VESSEL SHIELDING

The in-vessel shielding consists of mainly the radial shielding and the upper axial shielding. The shield design is

96

essentially governed by the dose rate specified for personnel working close to the secondary sodium pipelines in the steam generator building area. The dose rate criterion presently specified by the Atomic Energy Regulatory Board of India is 20 mSv/y for occupational exposure. The corresponding criterion for shield design has been specified as 1 micro-Sv/h. The allowable secondary sodium activity has been computed for the above dose limits at a distance of 20 cm from the surface of a secondary sodium pipeline and comes to be 3 Bq/cc.

The fuel storage locations were initially after two rows of stainless steel reflector. To reduce fission rate in the storage locations the outer row of stainless steel reflector was replaced by a row of boron carbide shielding subassemblies. Low fission rates in storage locations were required for reducing shielding and ease of cooling during handling of irradiated subaseemblies. This change also helped in reducing the radial shield thickness.

The in-vessel radial shielding is located after the fuel storage locations. The intermediate heat exchangers centre line is located in the sodium pool at a radial distance of 4 m from the core centre. Radial shield optimisation studies were carried out such the secondary sodium activity does not exceed 3 Bq/cc. Initially, optimisation studies were carried out with shield materials like carbon steel, graphite, 2% borated graphite and stainless steel. This led to excessively thick shields and, in turn, to larger vessel diameter. As reducing the vessel diameter leads to significant reduction in costs, in subsequent studies, only stainless steel and boron carbide have been considered. Using boron carbide it has been possible to reduce the radial shield thickness from 15 rows to 9 rows (each

row corresponds to a thickness of 13.3 cm of the material). The proposed configuration for the radial shield consists of two rows of stainless steel followed by seven rows of boron carbide. It was also found that the last three rows of boron carbide could be of vibro-compacted boron carbide powder instead of boron carbide pellets.

The proposed design limits for neutron irradiation dose is 1 dpa and for He production is 0.1 appm for components such as core cover plate, control plug and the grid plate. These limits have been used for the design of the upper axial shields which are integrated in the subassembly itself. Design of upper axial shields were initially carried out with stainless steel and borated graphite. The upper axial shields need to satisfy two criteria viz. the neutron fluence seen by the core cover plate and the secondary sodium activity. It was found that the neutron fluence on the core cover plate was low and did not influence axial shield deisgn which is hence based on secondary sodium activity. The proposed upper axial shield now consists of 15 cm of stainless steel followed by 63 cm of boron carbide. No shield has been provided in the lower axial direction. Calculations show that the helium production rates are much smaller than 0.1 appm and the dpa at grid plate is 0.25 dpa.

In vessel transfer post (IVTP) for transfer of irradiated subassemblies has been located in the radial shield. Hence it was necessary to design locally a better shield to take care of the loss of shielding due to the presence of the IVTP. It is proposed to place 5 rows of enriched boron carbide around the IVTP (Fig.1).

13.0 DETECTOR LOCATION FLUX

The flux at the detector location below the reactor vessel during start-up was estimated using the presence of inherent neutrons in fresh and irradiated fuel. It was found that the flux at detector location was too low for use of fission counters or boron detectors. Feasibility of using a neutron guide to enhance flux at detector location was studied. It was found that provision of neutron guide in blanket first row enhances the thermal equivalent flux by a factor of ten only. It was considered feasible to use high sensitivity He-3 counters without the use of neutron guide for the neutron monitoring during routine startups. For the initial core loading and approach to criticality, use of in pile detector and auxiliary neutron source is planned.

14.0 CONCLUSION

The conceptual design of PFBR core has evolved taking into account constraints of initial plutonium inventory, limited possibility of extensive advanced fuel irradiation testing and limited possibility of experimental investigations on core configurations radically different from the conventional homogeneous mixed oxide fuelled type. In accordance with the constraints, a core design has evolved which places high importance on successful past experience, minimisation of plutonium inventory, small core size, low initial capital cost, economic power generation on closed fuel cycle, and just self-sustaining from plutonium breeding considerations.

CONCEPTUAL DESIGN OF REACTOR ASSEMBLY OF PROTOTYPE FAST BREEDER REACTOR

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Abstract

The conceptual design of Reactor Assembly of 500 MWe Prototype Fast Breeder Reactor (as selected in 1985) was reviewed with the aim of 'simplification of design', 'Compactness of the reactor assembly' and 'ease in construction'.

The reduction in size has been possible by incorporating concentric core arrangement, adoption of elastomer seals for Rotatable plugs, fuel handling with one transfer arm type mechanism, incorporation of mechanical sealing arrangement for IHX at the penetration in Inner vessel redan and reduction in number of components. The erection of the components has been made easier by adopting 'hanging' support for roof slab with associated changes in the safety vessel design. This paper presents the conceptual design of the reactor assembly components.

1.0 INTRODUCTION

After successful construction, commissioning and operation of Fast Breeder Test Reactor (40 MWt, 13 MWe), design of a 500 MWe Prototype Fast Breeder Reactor (PFBR) is in progress in India. Pool type layout is chosen for the Primary sodium circuit as it accommodates most of the active components in one single vessel, eliminates the high pressure active sodium piping, provides better containment for radioactivity and reduces the exposure of operating personnel to radioactivity. The concepts chosen for the various components of Reactor Assembly (fig. 1) in order to have simplification of design, compactness and ease of construction are presented in this paper. The main characteristics of PFBR are given in Table I and the major dimensions of Reactor Assembly components are listed in Table II.

2.0 CONCEPTS CHOSEN FOR MAJOR COMPONENTS

2.1 Main vessel

Main vessel contains the primary sodium and supports the core, grid plate, core support structure and inner vessel. It also serves as a boundary against release of radioactivity under operating & accident conditions and absorbs the energy released during Core Disruptive Accident (CDA). It is a cylindrical vessel with no penetrations (fig.2) and hence provides maximum reliability against sodium leaks. Bottom cover is of special shape arrived from buckling considerations. The
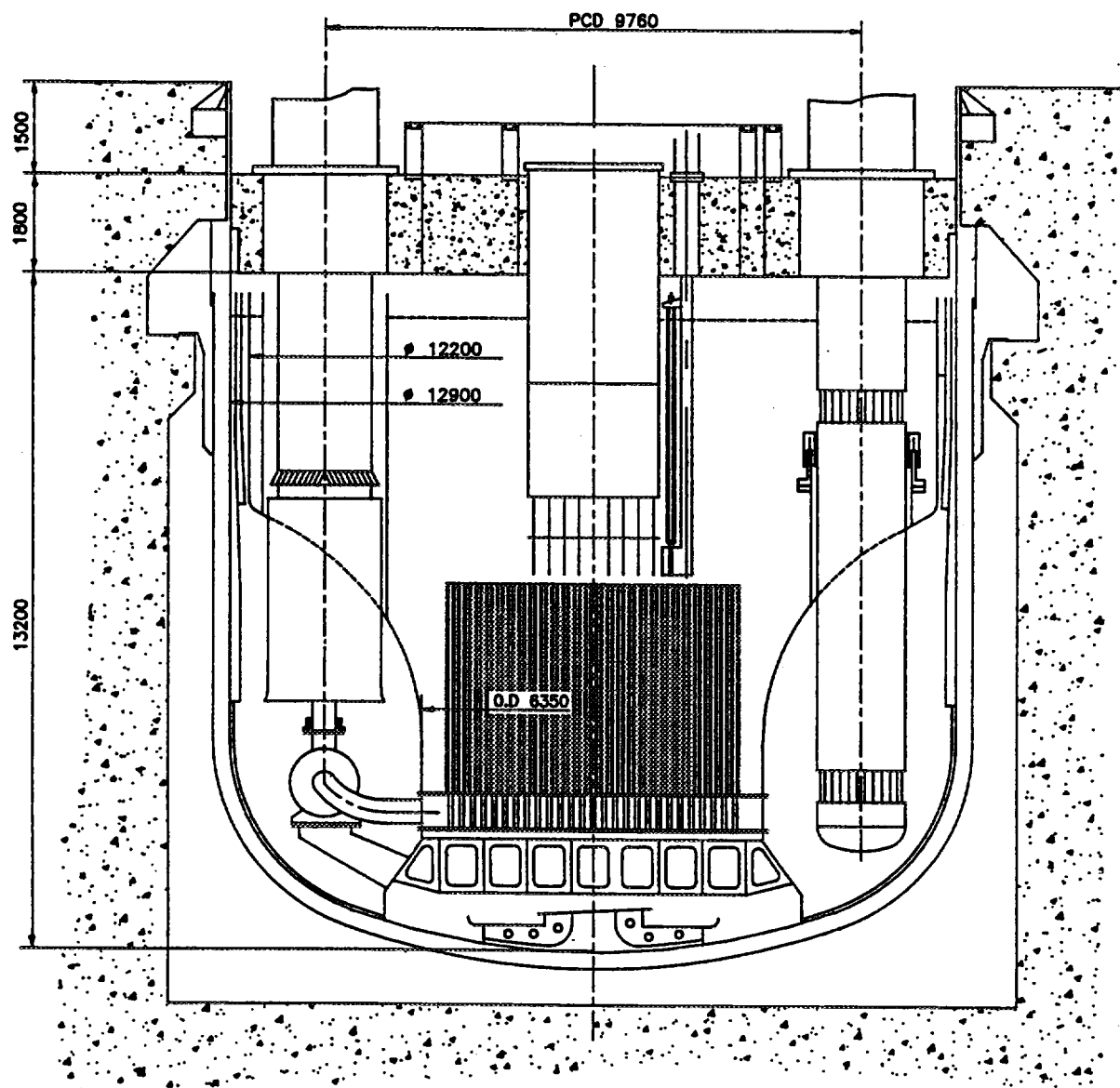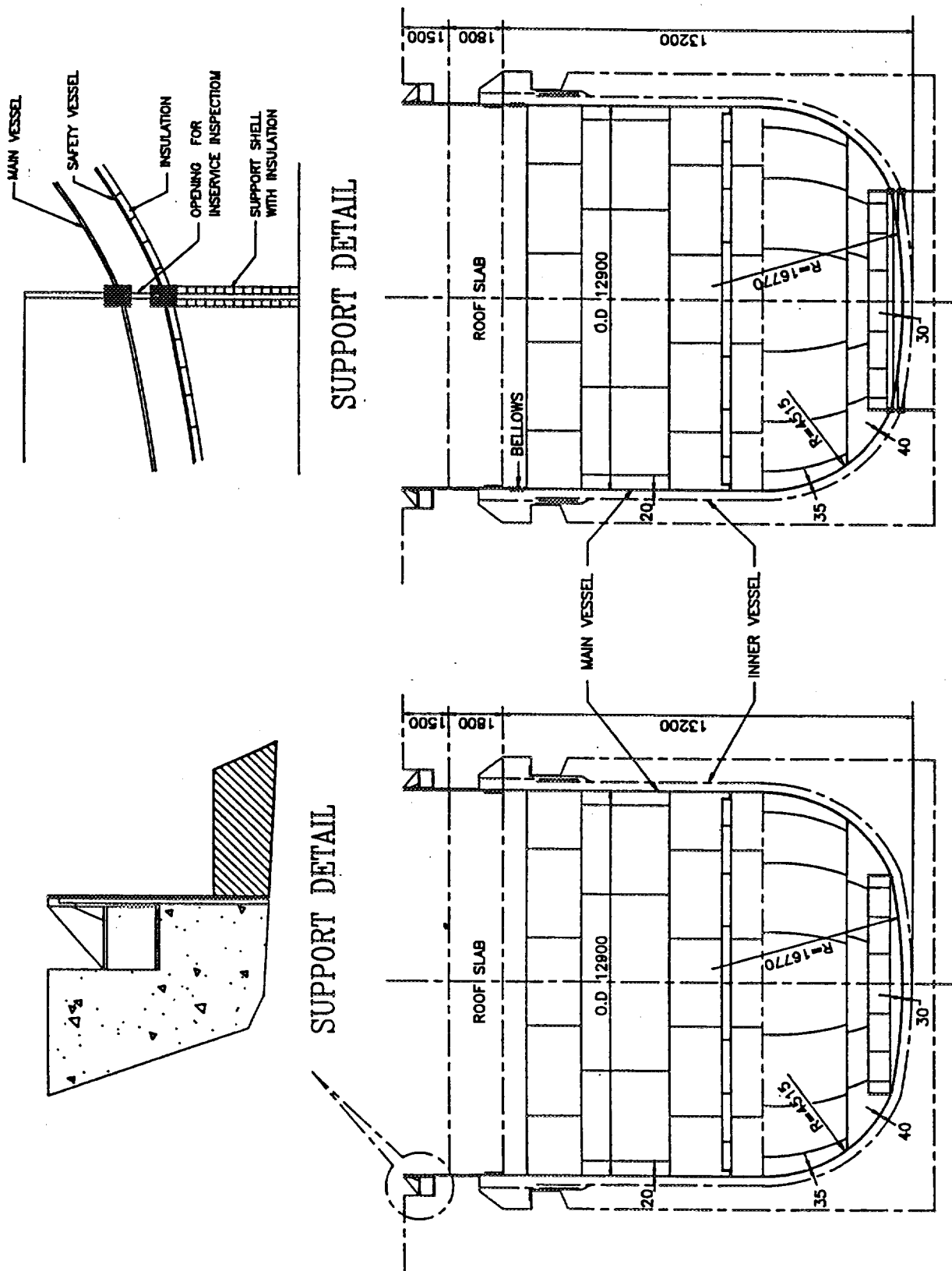
FIG. 1. Reactor assembly.

TABLE : I   PFBR — MAIN CHARACTERISTICS

REACTOR POWER	1250 MWt / 500 MWe
PRIMARY CIRCUIT	POOL TYPE
ACTIVE CORE DIA / Ht	2 m / 1 m
FUEL	UO_2 — PuO_2
PRIMARY INLET TEMP	670 K (397°C)
OUTLET TEMP	820 K (547°C)
SECONDARY INLET TEMP	628 K (355°C)
OUTLET TEMP	798 K (525°C)
PRIMARY Na FLOW	8.5 m^3/s
NO. OF PRIMARY PUMPS	2
NO. OF IHX	4
NO. OF SECONDARY LOOPS	2

TABLE: II   SIZE OF REACTOR ASSEMBLY COMPONENTS

COMPONENT	SIZE (m)	
	DIA	Ht
MAIN VESSEL	12.9	13.2
SAFETY VESSEL	13.53	13.52
INNER VESSEL	12.2	9.1
ROOF SLAB	12.9	1.8
LARGE ROTATABLE PLUG	6.93	1.8
SMALL ROTATABLE PLUG	4.68	1.8
INTERMEDIATE HEAT EXCHANGER	2.52	15.63
SODIUM PUMP	2.2	14.3

SUPPORT DETAIL

SUPPORT DETAIL

MAIN VESSEL

INNER VESSEL

MAIN VESSEL WITH TOP SUPPORT

MAIN VESSEL WITH BOTTOM SUPPORT

FIG. 2

MAIN VESSEL
SAFETY VESSEL
INSULATION
OPENING FOR INSERVICE INSPECTION
SUPPORT SHELL WITH INSULATION

ROOF SLAB
O.D 12900
R=16770
R=4515
BELLOWS
1500 1800 13200
20
35
30
40

104

main vessel is suspended by a cylindrical shell supported on the reactor vault. This arrangement simplifies the erection of reactor assembly components, eliminates the risk of buckling of the support shell under seismic conditions and permits free dilation of main vessel in radial and axial directions, thus minimising thermal stresses.

Supporting the main vessel at the top also eliminates the need for large diameter bellows, which has to be designed for higher pressure due to CDA conditions, if the vessel was supported at the bottom (fig.2). In order to enhance the structural integrity and to minimise the effects of age hardening, the temperature of the upper portions are maintained below 723 K (450 ^{0}C) by cooling with cold sodium. This also reduces the axial temperature gradient in the vessel and the damage to the shell because of thermal cycling due to changes of sodium level in the cold pool.

2.2 Cooling circuit for Main Vessel

The cooling circuit consists of cooling pipes and thermal baffles (fig.3). The cooling pipes are spaced suitably to provide uniform flow distribution along the circumference of the vessel. The cooling pipes are bound together by a circular ring to avoid flow induced vibrations (fig.4). This arrangement eliminates the need for support brackets on the main vessel which are undesirable from the thermal inertia consideration. The thermal baffle has a weir arrangement at the free end to reduce the energy of free fall of sodium so that level oscillations due to fluid structure interaction are minimised and gas entrainment risk in the sodium pool is reduced.

2.3 Core Support Structure

The core support structure (CSS) is a box structure with orthogonal stiffeners. This arrangement results in effective use of structural material for supporting the core loads. It is supported on the main vessel through a cylindrical shell of sufficient length, to avoid transmission of rotations to main vessel due to its deflection. Supporting CSS from the cylindrical portion of main vessel is not considered as it results in increased main vessel height.

2.4 Grid Plate

A single grid plate is used to support and locate the core and shielding subassemblies. This simplifies the supporting of grid plate on the CSS, keeps the grid plate nozzle to pipe connection on the cold pool side which can be of welded construction and avoids penetration of these pipes through the CSS shell which requires bellows for sealing. A fully bolted construction is adopted to simplify manufacture of this large diameter component. The bolted construction offers ease of erection of components and reduces the time involved. The grid plate has four inlet pipes, with a pair of nozzles connected to each of the two primary sodium pumps through a spherical header.

INNER VESSEL

INNER BAFFLE

OUTER BAFFLE

MAIN VESSEL

20

20

OD 12200

25

25

3000

6380

OD 12900

20

2000

25

THERMAL BAFFLE

R2

R1

R3

σ 2

σ 1

WEIR SHAPE

FIG. 3.

106

CIRCULAR RING

COOLING PIPE

OD 12900

20

R=4515

35

R=16770

40
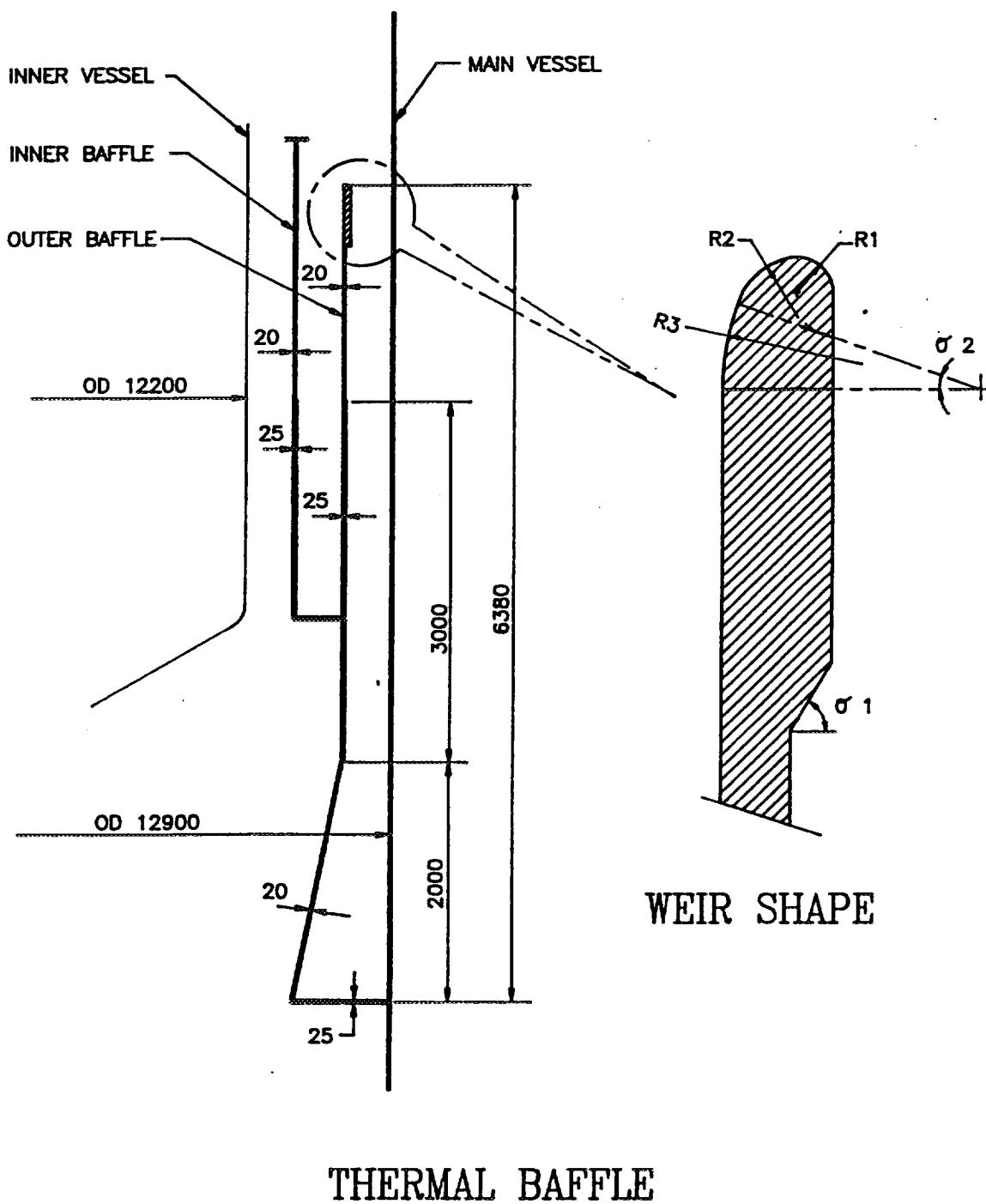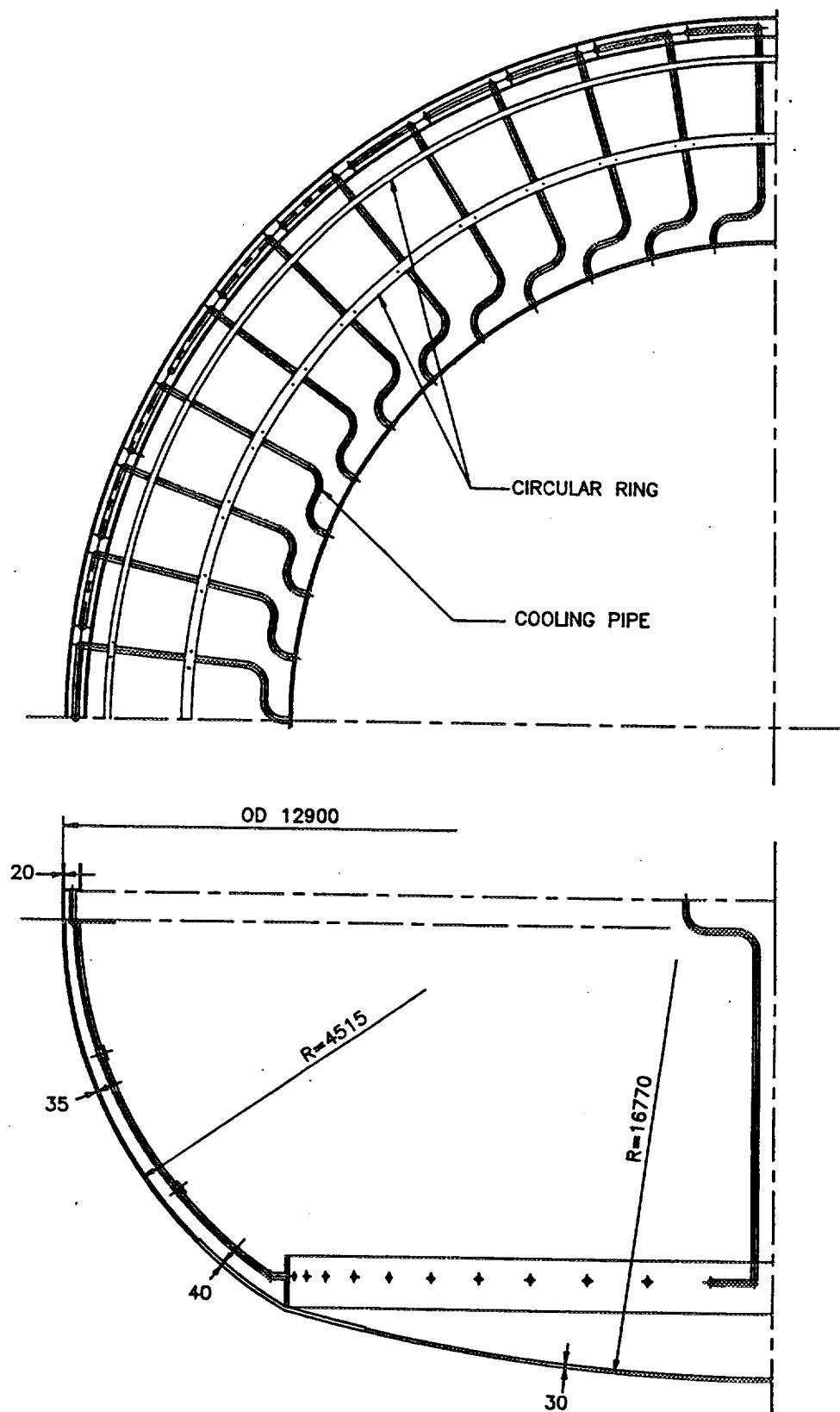
30

FIG. 4. Main vessel cooling pipes.

The sleeves provided for shielding subassemblies at the periphery of grid plate have no provision for radial entry of coolant and they assist in uniform distribution of coolant flow towards central regions of grid plate. However, provision for axial entry of sodium at the bottom of shielding subassemblies is made. The leakage flow from the bottom of grid plate is utilised for the purpose of cooling shielding subassemblies and main vessel.

2.5 Core Catcher

Though the probability of occurance of fuel melt down is very remote, it is foreseen that local melting of fuel pins in subassembly can occur due to flow blockage and the effect will propagate to the surrounding six subassemblies causing fuel pins to melt in them.

Hence, in accordance with defence in depth philosophy and to take care of the residual risk of fuel pin melt down in seven subassemblies, a provision of core catcher below CSS is envisaged. The details of core catcher and core subassemblies are presented in another paper.

2.6 In-Vessel Transfer Position

The invessel transfer position (IVTP) is located as close to the core centre as possible to reduce the diameter of Large Rotatable Plug (LRP) and hence the diameter of the main vessel . In order to achieve this, shielding assemblies containing boron carbide enriched in Boron-10 are used in the vicinity of IVTP. The tilter for handling of subassemblies in IVTP is guided by a special sleeve arrangement in grid plate and is supported on the CSS.

2.7 Inner Vessel

The inner vessel which seperates the sodium into hot pool and cold pool, is fixed on the grid plate by a flange at its lower end. The lower shell surrounds the core, which is concentrically located and the upper shell surrounds the IHX and primary sodium pumps, the stand pipes of which penetrate redan portion of vessel. The location and shape of redan from upper shell to the lower shell is arrived at based on the thermal hydraulics and structural considerations. A single vessel with torroidal shaped redan has been selected.

To minimise the leakage of sodium from hot pool to cold pool at the penetration of IHX in the vessel, a mechanical seal with 'piston rings' is used. This seal is very compact and results in minimum diameter of the upper shell. The alternative type of sealing, argon pocket seal, results in larger diameter of the upper shell, has the risk of introduction of argon gas along with sodium into the core inadvertantly and hence has been avoided.

2.8 Roof Slab

Roof slab along with rotatable plugs and control plug provides thermal and biological shielding in the upper axial direction of the reactor. It also supports to the components such as primary sodium pumps, intermediate heat exchangers and rotatable plugs and provides a leak tight enclosure for the cover gas. Since thick plates (600 - 800 mm) are not available, a box structure made of 30 mm thick carbon steel plates (to avoid PWHT) is used as roof slab. Concrete of density 3.5 g/cc is used as the shielding material as it is cost effective. For cooling , air is chosen as the medium as it does away with the need to have a large supply of nitrogen and associated safety problems. The bottom plate of roof slab is cooled by air jets from an inlet plenum, thus making it possible to maintain uniform temperature. The ensuing hot air is then used to heat the top plate to reduce the temperature difference between the plates (fig. 5). This reduces the bowing of roof slab due to differential temperature. The temperature of the top plate is maintained between 373 K (100° C) to 383K (110°C) to avoid deposition of sodium in the annulus between the shells at various component locations. The vertical shells in roof slab at Pump and IHX penetrations are also cooled to control the circumferential temperature difference in them within acceptable limits.

The ex-vessel transfer of core subassemblies is done through a primary ramp for which a penetration is provided in roof slab. The penetration is cooled by forced circulation of air in the jacket surrounding it, so that in case the fuel transfer pot gets stuck in the vicinity of the penetration, cooling can be provided to avoid undue rise in temperature. The heat removal capacity of the ramp cooling circuit is 5 KW.

2.9 Rotatable Plugs

The rotatable plugs are used to position the invessel transfer machine over the core subassembly to be handled inside the main vessel. The diameter of the main vessel is controlled by diameter of LRP and hence it is desirable to keep it minimum. The combination of two rotatable plugs and one transfer arm machine which results in reasonable diameter of LRP has been chosen (fig. 6). The structural, shielding and cooling aspects of rotatable plugs are same as those of roof slab. During fuel handling, when the plugs are being rotated, they are cooled by air from reactor hall using blowers located on them temporarily.

2.10 Support Arrangement for Rotatable Plugs

To avoid cold spots on the support structure where sodium aerosols could form deposits, the support flanges for LRP and Small Rotatable Plug (SRP) are located at roof slab top elevation (fig.7). During reactor operation, sealing is provided by 'O' rings which are disengaged prior to rotation of the plugs. In order to reduce the flange width of the support arrangement,
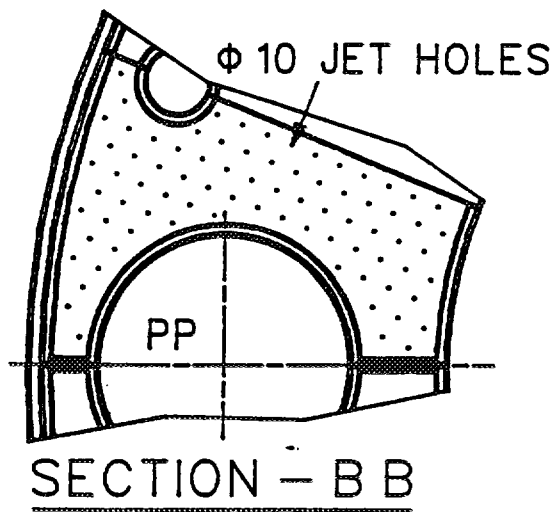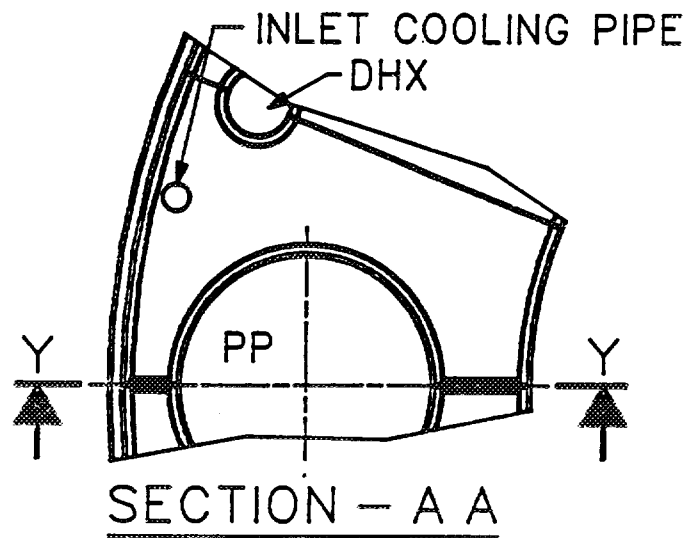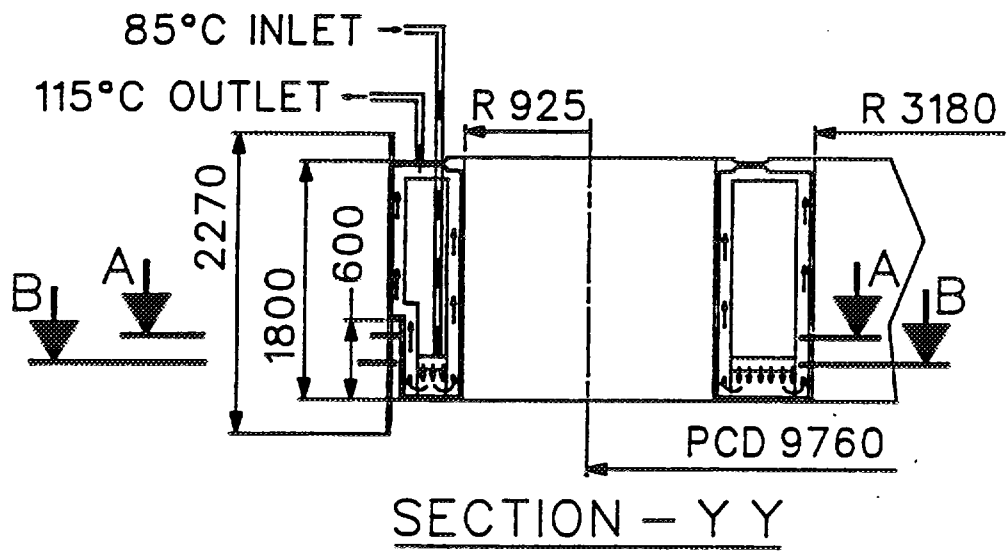
FIG. 5. Roof slab cooling system.

R 3110

R 3465

SRPF = ø 4685

LRPF = ø 6930

R275

R160

Ø2520

R325

R200

572.5

CP

SRP

R 2342.5

LRP

R 2027.5

667.5   Ҁ SRP

Ҁ CP,LRP,MV   1907.5   Ҁ IVTM

667.5   Ҁ SRP   572.5

Ҁ CP,LRP,MV   1907.5   Ҁ IVTM

FUEL HANDLING SCHEME
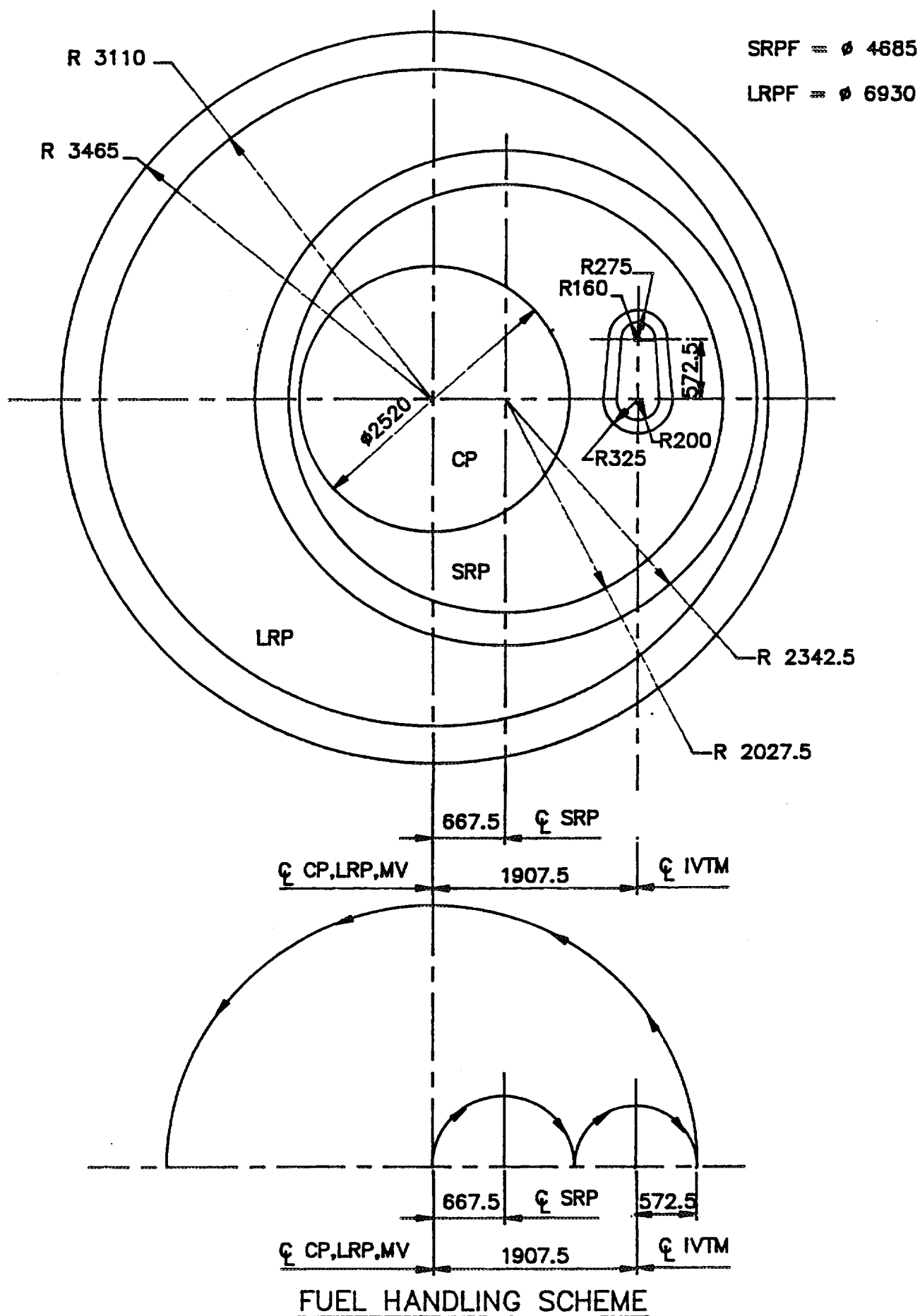
FIG. 6. Size of rotatable plugs for handling scheme with 1 transfer arm.

355/315*

HYDRAULIC LIFTING DEVICE

350

260/175*

STRIKER PLATE

131

V-RING SEAL

1775/1690*

BEARING SUPPORT SHELL

SEAL HOLDER

RADIAL 'O' RING SEALS

LIP SEALS
ENERGISED BY
GAS PRESSURE

LRP / SRP
SUPPORT RING

100

100

SUPPORT FLANGE FOR LRP / SRP

* DIMENSIONS REFERS TO SRP

FIG. 7. Sealing arrangement for rotatable plugs
(under normal operation).

112

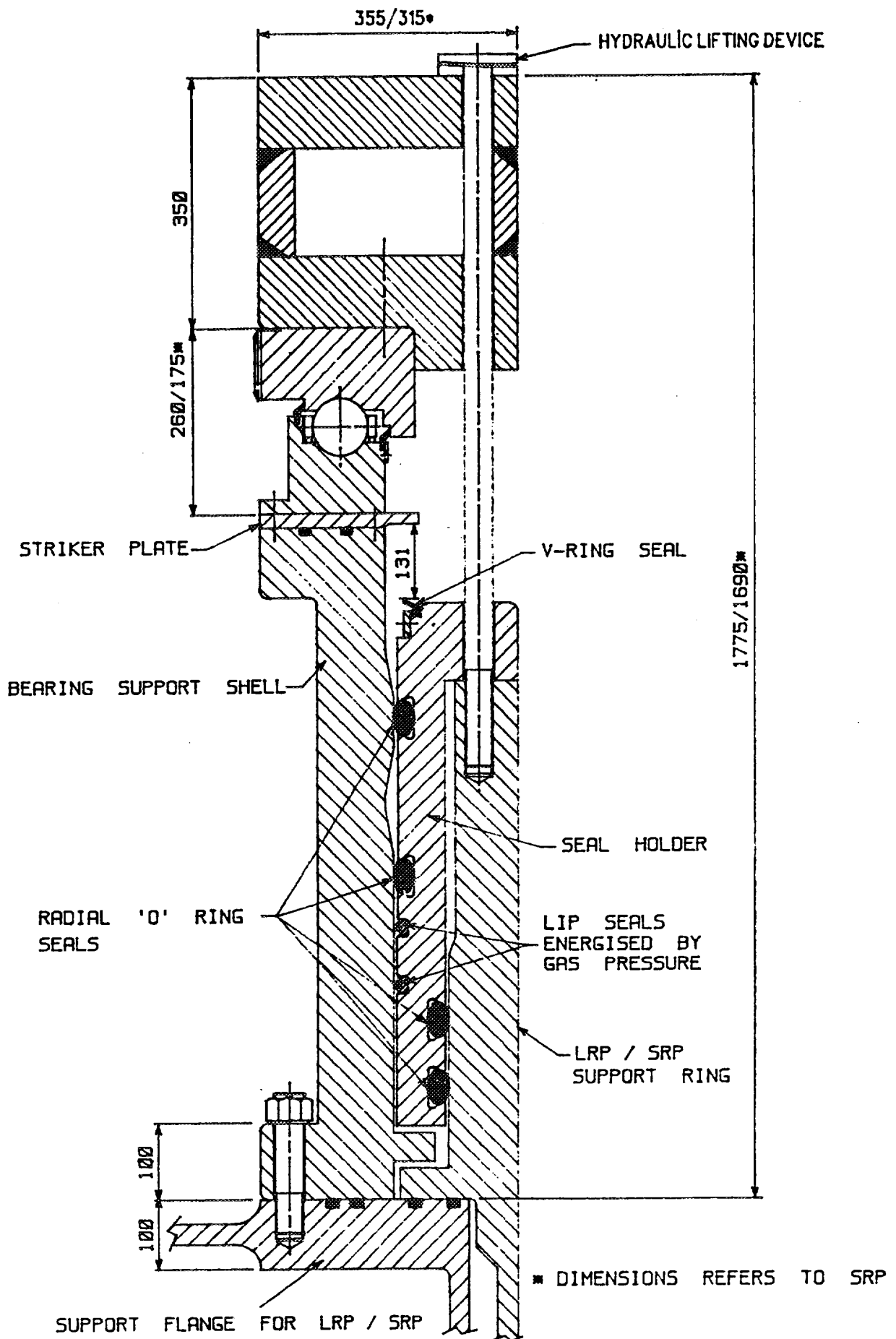elastomer seals (V ring seal and lip seals) are chosen for sealing during fuel handling conditions. A seal holder is provided which can be lifted separately to enable easy replacement of the seals. Liquid metal seal has not been used as it requires costly machining, increases the construction time, results in larger flange width and also requires auxiliaries.

2.11 Control Plug

Control plug provides support for the control rod drive mechanisms, thermocouple tubes housing the thermocouples which monitor the outlet temperature of fuel subassemblies and failed fuel identification modules. A separate control plug is provided on SRP to facilitate easy replacement if required. Though integration of control plug with SRP results in smaller diameter of LRP, a separate control plug is selected in order to have the above advantage.

2.12 Intermediate Heat Exchangers

In order to reduce the capital cost and construction time of the reactor, 2 loop concept with 2 IHX per loop has been chosen. The IHX tube size and the diameter of IHX have been optimised to have savings in overall cost of the component.

2.13 Primary Sodium Pumps

There are 2 primary sodium pumps and they are of top suction, single impeller type. The removable part diameter of the pump is kept minimum by integrating the pump intake skirt with the stand pipe of the inner vessel. The outlet from the pump is connected to a spherical header which distributes the flow through two pipes to the grid plate.

2.14 Safety Vessel

A safety vessel surrounds the main vesel, so that in the unlikely event of sodium leaking from main vessel, the sodium level in hot pool does not fall below IHX inlet windows and a path for natural convection cooling is available. The safety vessel is supported directly on reactor vault. This arrangement also helps to reduce the outer diameter of roof slab. The nominal gap between the vessels is 300 mm from considerations of in-service inspection. An anti- convection barrier is fixed in the gap between the vessels at an elevation close to bottom of roof slab in order to minimise convection currents .

Thermal insulation in the form of panels of thickness 100 mm are fixed on the outer surface of safety vessel so that the area of thermal insulation is minimum, the vault remains cool and intervention for maintenance is easy. The insulation is sized based on temperature of the safety

1. REACTOR VAULT READY FOR ERECTION OF SAFETY VESSEL

2. SAFETY VESSEL LOWERED & SUPPORTED ON REACTOR VAULT

3. SV TEMPORARILY COVERED, REACTOR VAULT BUILT UP, SUPPORT EMBEDMENT PROVIDED

4. MAIN VESSEL WITH CSS LOWERED & SUPPORTED ON SAFETY VESSEL

5. THERMAL BAFFLES WELDED TO MV

6. GRID PLATE BOLTED TO CSS & PUMP TO GRID PLATE PIPES ERECTED

FIG. 8a Erection sequence for reactor assembly components.

114

7. INNER VESSEL LOWERED & BOLTED TO GRID PLATE

8. ROOF SLAB LOWERED & WELDED ON TO ITS SUPPORT

9. MV ALIGNED & WELDED TO ROOF SLAB

10. COMPLETION OF TOP PORTION OF SAFETY VESSEL
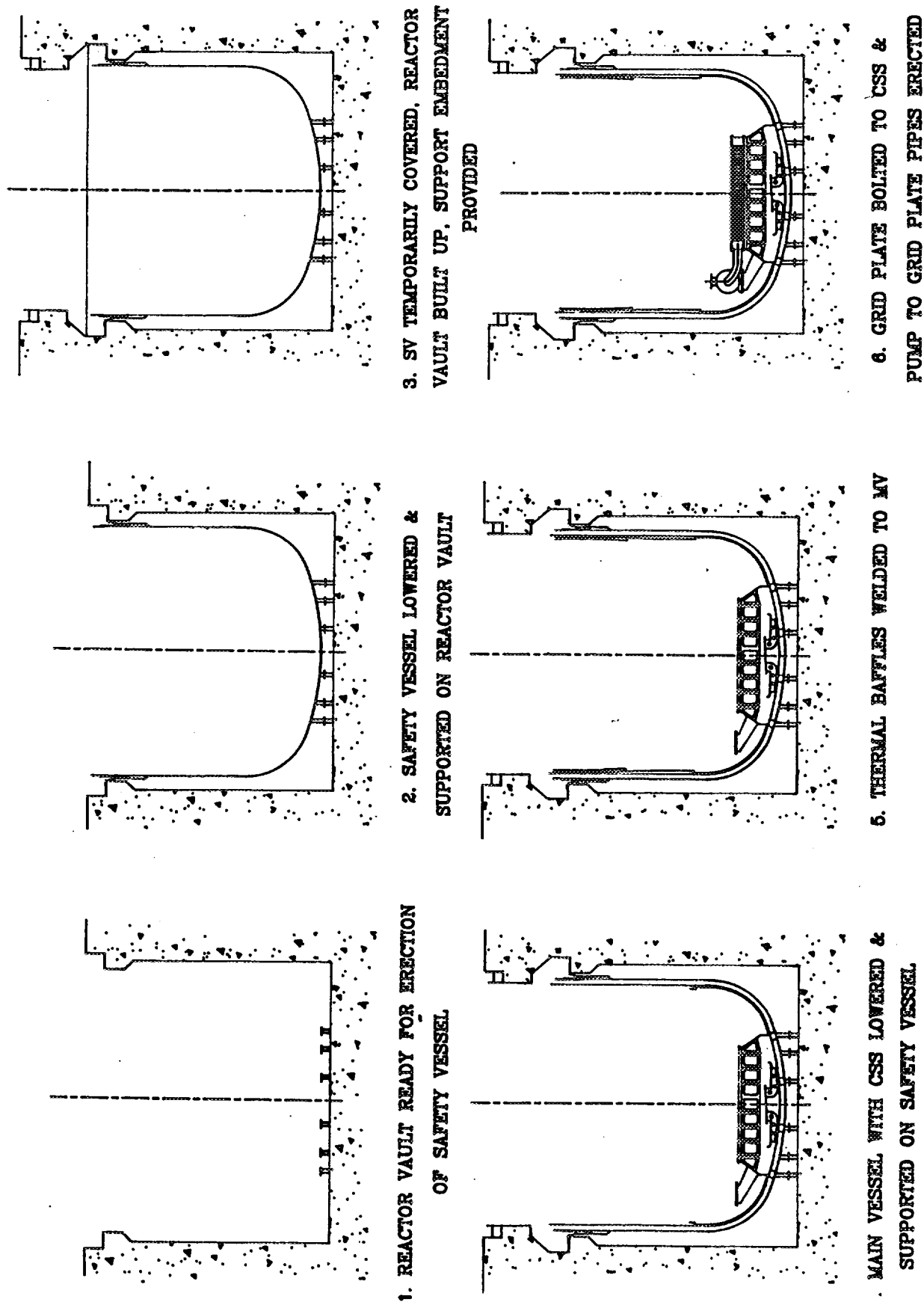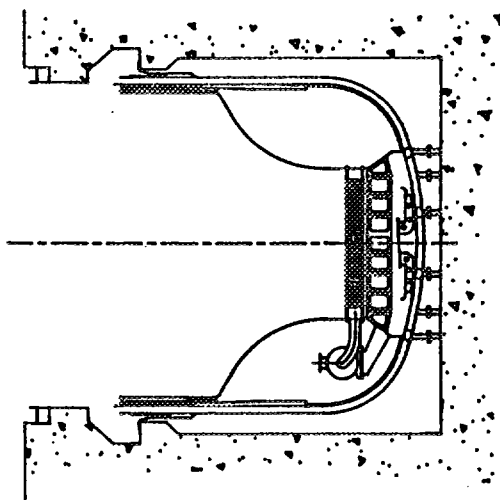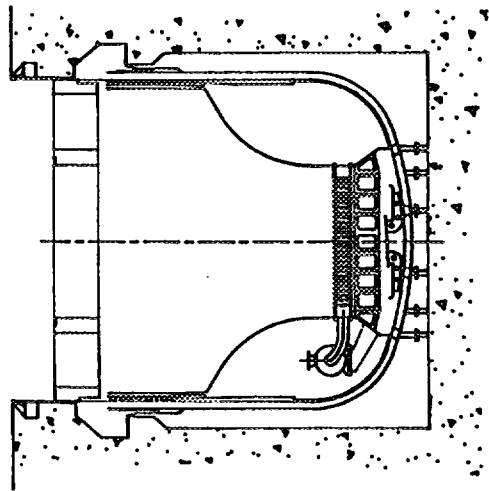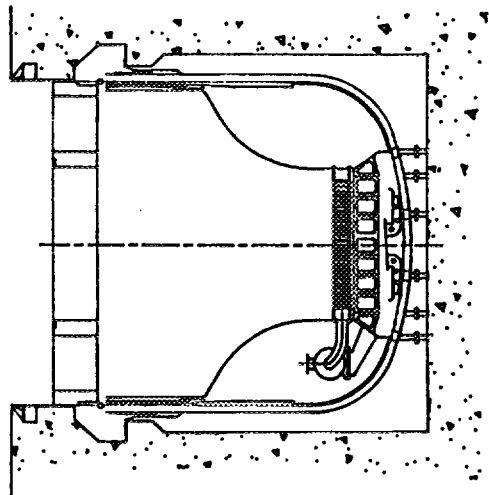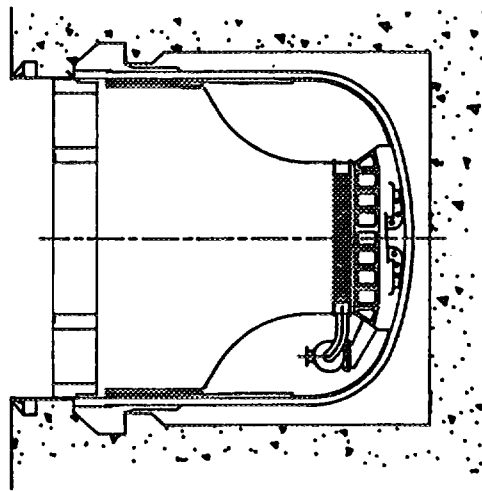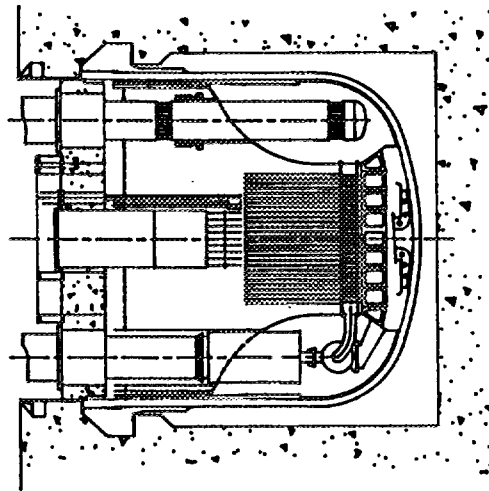
11. OTHER REACTOR ASSEMBLY COMPONENTS ERECTED

FIG. 8b. Erection sequence for reactor assembly components.

115

vessel under DHR conditions and heat removal capacity of reactor vault cooling system which uses air as the cooling medium.

3.0 ERECTION OF REACTOR ASSEMBLY COMPONENTS

Considerable simplification has been made in erection of Reactor Assembly components.

The erection sequence foreseen is as follows (fig.8).The reactor vault with lining is made ready for erection of safety vessel. Safety vessel with thermal insulation mounted on its outer surface is erected in the vault using screw jacks to support it at its bottom. The upper part of the reactor vault which supports roof slab is then cast. Collapsible spacer pads are fixed on the inner surface of safety vessel and main vessel with CSS is lowered and supported on it. The grid plate is lowered and is bolted on to CSS. The grid plate inlet pipes are erected. The inner vessel is lowered and is bolted to grid plate. The roof slab is then lowered on to the reactor vault and is welded to the embedded support structure. The combination of main vessel, inner vessel etc. is aligned to roof slab and the main vessel is welded to it. The collapsible spacer pads are removed from the space between main and safety vessels. The supporting screw jacks for safety vessel are also removed. The upper portion of safety vessel is assembled in-situ. After these operations, LRP, SRP, control plug, primary sodium pumps, IHXs , decay heat exchangers etc. are erected.

The concrete supporting structure for the Inclined Fuel Transfer Mechanism (IFTM-exvessel fuel transfer machine) is cast and IFTM is installed.

4.0 CONCLUSION

The conceptual design of reactor assembly components has been reviewed and considerable simplifications have been made in design of CSS, grid plate, inlet pipes to grid plate, primary sodium pumps, roof slab, and support structure for rotatable plugs. The resulting reactor assembly is more compact in size and offers considerable ease in erection of components.

CONCEPTUAL DESIGN OF HEAT TRANSPORT SYSTEMS AND COMPONENTS OF PFBR-NSSS

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Abstract

The production of electrical power from sodium cooled fast reactors in the present power scenario in India demands emphasis on plant economics consistent with safety. Number of heat transport systems/components and the design of principal heat transport components viz sodium pumps, IHX and steam generators play significant role in the plant capital cost and capacity factor. The paper discusses the basis of selection of 2 primary pumps, 4 IHX, 2 secondary loops, 2 secondary pumps and 8 steam generators for the 500 MWe Prototype Fast Breeder Reactor (PFBR), which is now in design stage. The principal design features of primary pump, IHX and steam generator have been selected based on design simplicity, ease of manufacture and utilisation of established designs. The paper also describes the conceptual design of above mentioned three components.

1.0 INTRODUCTION

The present power scenario in India is characterised by major contribution from fossil power stations. Under these circumstances, introduction of fast breeder reactors (FBRs) on commercial scale is possible only if their economic competitiveness is demonstrated viz-a-viz fossil power stations. This requires a set of design features which lead to lower capital cost, reduction in construction time and improved capacity factor without compromising safety.

Recent review of the 4 loop design with sodium reheat selected earlier for PFBR has revealed that NSSS constitute ~ 50 % of direct cost (50 % BoP) and "circuits" contribute to ~ 45 % of NSSS cost. Also the principal heat transport components influence capacity factor and construction time. There is thus a need to relook into the number of systems/components, simplify the design and reduce the excessive design margins. Lower capital cost and unit energy cost resulting from this modified design will provide scope for competitiveness in the long run when series construction of this design is taken for comparison. This paper deals with the modified PFBR design aspects related to heat transport systems/components and conceptual design of primary sodium pump, IHX and steam generator.

Intermediate sodium circuit is interposed between the primary sodium system and water/steam circuit in order to contain the radioactive primary sodium within the main vessel envelope, to improve maintainability of heat transport components and to avoid the consequences on account of entry of sodium-water reaction products into the primary system.

2.0 NUMBER OF LOOPS/COMPONENTS

The following factors have been considered in deciding the number of loops/components.
- Safety
- Design capability
- Raw materials availability & manufacturing capability
- LMFBR and PWR operating experience
- Construction time
- Capital cost
- Capacity factor
- Design trend abroad
- Impact on R&D

2.1 Minimum Number of Loops

As a matter of philosophy, a minimum of two primary pumps and two secondary loops is essential so that in case of non- availability of one primary pump/secondary loop, the decay heat can be easily removed by other primary pump/secondary loop. Though PFBR has an independent safety grade decay heat removal circuit consisting of 4 independent loops, it is logical to use secondary circuit for decay heat removal when off-site power is available. Hence, the requirement of minimum of two primary pumps and two secondary loops is considered essential.

2.2 LMFBR Heat Transport Components Experience

2.2.1 Pumps, IHX, Piping & Tanks

With exception of IHX in Phenix, the overall operating experience on sodium pumps, IHX, piping and tanks can be treated as satisfactory. In Phenix, the cracks in secondary sodium piping, SS 321 grade, is material related as this grade is notorious for reheat cracking. PFBR secondary sodium piping is of SS 316 LN grade and this grade does not exhibit reheat cracking phenomenon. Leak in IHX gas seal in SPX is not applicable in case of PFBR, as IHX design incorporates mechanical seal. Failure at secondary outlet header of IHX in Phenix is fairly understood now. PFBR design has design concept similar to Phenix with primary sodium on shell side. As detailed later in this paper, thermo-hydraulics of PFBR tube bundle is different as variable secondary flow distribution has been adopted in the design. Accordingly there is good confidence in PFBR IHX design. Failure due to thermal striping in auxiliary sodium circuits in many reactors, test facilities and Phenix expansion tank is proposed to be taken care of during detailed design by minimising temperature difference between two sodium streams and incorporating suitable mixers.
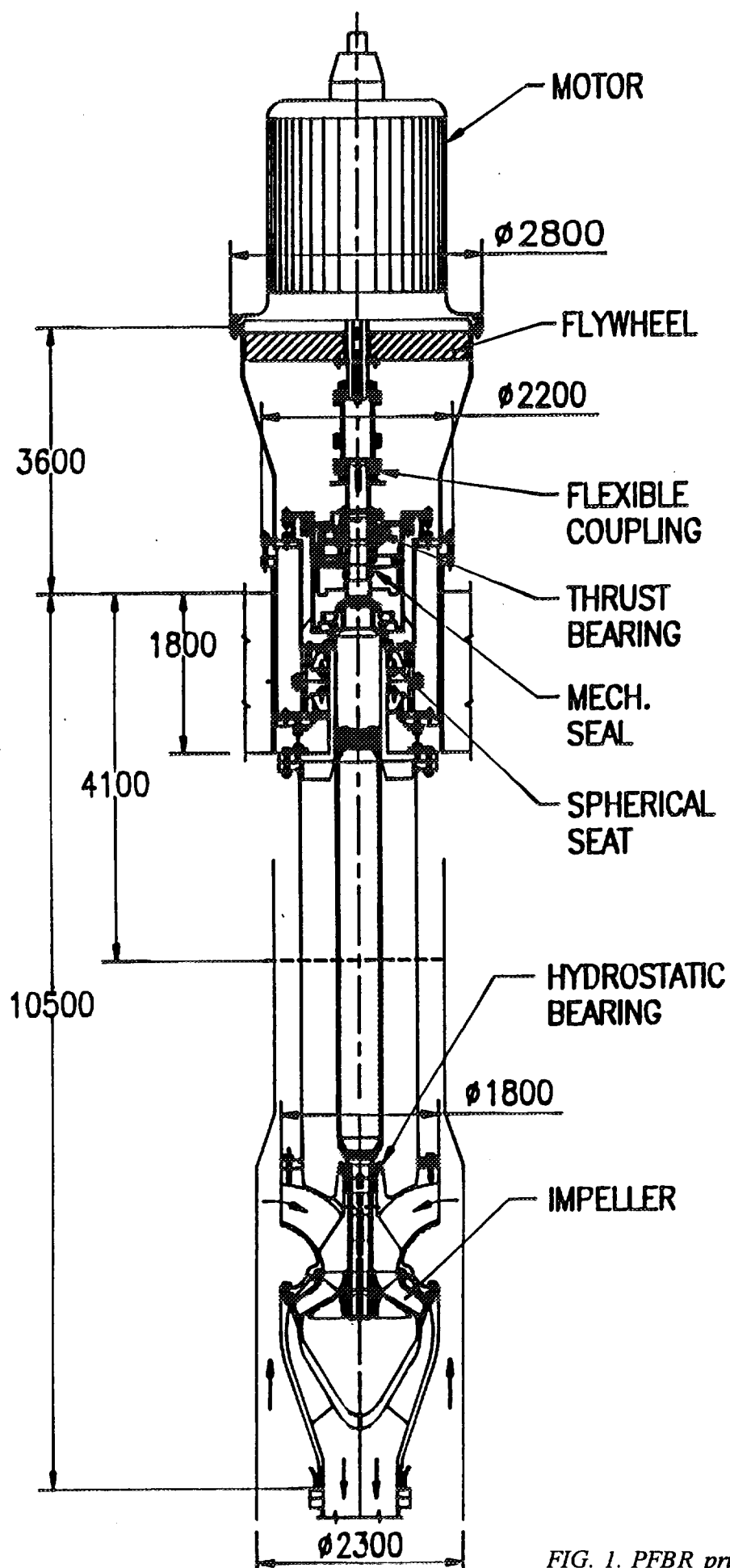
118

FIG. 1. PFBR primary sodium pump.

119

2.2.2 Steam Generator

Operating experience on single wall Steam Generator (SG) has shown that this component plays an important role in influencing the plant capacity factor. No experience is available on the type of SG selected for PFBR. However, the design incorporates the lessons learnt from the operating experiences of Phenix, PFR and BN-600. Experience of BN-600 operation with 2 loop and remaining 7 SG modules of the affected loop in case of a small water/steam leak is noteworthy which has been given due consideration.

Operating experience also shows that tube to tubesheet welds constitute the major risk of failures followed by flow induced vibrations. For gross simplification, SG reliability can be associated with failure rate of tube to tubesheet welds.

Operating experience (particularly in BN 600) points out that modular SG design would be a right choice for PFBR. There are two options of modular SG, one by increasing the number of secondary loops to match number of SG modules and second option is to have multiple SG modules per loop with number of SG modules selected so as to permit (m-1) operation in case of a small tube leak. However, the first option, though provides design simplicity and eliminates the need for large sized steam generator sodium isolation valves, is expensive and in the context of capital cost reduction is not favoured.

2.3 Impact of Number of Loops/Components on Capacity Factor

LMFBR operating experience on 2,3 and 4 loops of PFBR size does not exist to draw any conclusions. The studies indicate that minimum number of components with exception of SG provide better capacity factor due to lower maintenance time, reduced outage time in case of repairs called for due to generic design or manufacturing error. The above is also the case when the affected component is replaced by spare one on occurrence of incident. Non-generic (isolated) failures favour increased number of loops/components. But these constitute small outage period. Operating experience on US Westinghouse and French PWR also indicate that reduction in number of loops provides enhanced capacity factor.

2.4 Impact of Number of Secondary Loops on Reactor Safety

As secondary circuit takes part in DHR, first impression is that the reactor safety is enhanced by number of secondary loops. However, safety limits under DHR can be met comfortably even with 1 secondary loop when off-site power is available.

120

2.5 Selection of Number of Components

Minimum number of primary pumps/secondary loops is favoured from capital cost reduction and improved capacity factor. 2 IHX/primary pump has been selected based on the operating experience of pool type reactor designs and recognising that IHX size influences reactor assembly size. Accordingly design with 2 primary pumps, 4 IHX, 2 secondary loops and 2 secondary pumps has been selected.

Number of SG/loop is influenced by acceptable temperature differences at primary sodium outlet of IHX and steam temperature outlet of SG of the unaffected and affected loop. This sets a limit of minimum of 3 SG per loop with preference for more number of modules per loop. Based on capital cost, cost of spare unit, outage associated with SG leak leading to (m-1) operation till next planned shutdown (with tube to tubesheet weld failure rate of 2 to 5×10^{-5} welds/a) and considering the impact of number of SG on overall construction schedule, it is estimated that 4 modules/loop would be a good choice and hence 8 SGs have been selected .

The components in the above option can be manufactured indigenously.

Preliminary cost analysis has shown that 2 loop design stated above provides an advantage of ~ 4 % over 3 loop design (3 pumps, 6 IHX, 3 SG/loop) and ~ 9 % over 4 loop design (4 pump, 8 IHX, 3 SG/loop) in terms of total investment cost including interest and escalation during construction.

The overall design selected with 2 primary pumps, 4 IHX, 2 secondary pumps and 8 SG in comparison to earlier design with 4 primary pumps, 8 IHX, 4 secondary pumps and 12 SG has less hardware, reduced construction time, lower investment cost, improved capacity factor and is in line with the present design trends.

3.0 PRIMARY SODIUM PUMP

As the main vessel diameter is influenced by the larger of the two diameters of the pump or the IHX, as the case may be, efforts have been directed towards achieving a compact pump.
The following factors were considered in evolving a compact pump.

- Single stage top suction impeller for simplicity of design
- Lower cavitation margin than adopted in the earlier design
- Fixed intake skirt welded to the inner vessel redan.
- Elimination of the Non return value at pump discharge. This permits increased submergence under the free sodium level in the pool and improves NPSHA.

With an available NPSH of 15.4 mNa and a reduced cavitation margin of 1.4, a single suction primary pump with a rated capacity of 4.25 m3/s at 75 m Na head has an operating speed of 590

rpm. This corresponds to a specific speed of about 48 (metric units) for which a mixed flow type impeller is well suited. The choice also helps to reduce the maximum hydraulic dimension of the pump to less than 1800 mm (Fig 1) by adopting an axial diffuser. The pump impeller is driven through a 9600 mm long shaft guided by a sodium lubricated hydraulic bearing and supported at the top by an oil lubricated thrust pad/journal bearing. Mechanical seals are used for shaft sealing. Each of the pump is powered by a 3600 KW induction motor and can be operated at a speed that can be varied continuously between 20- 100% of nominal speed.

The pump is supported from a flange at the penetration in the roof slab and a design with an additional upper compliant support (plain spherical bearing) allowing the whole assembly to tilt through a very small inclination in order to absorb the lateral differential expansion at the level of the bottom joint (removable) between the pump nozzle and the grid plate pipe is under study. In this the maximum permissible inclination of the pump is consistent with the safe operation of pump hydrostatic bearing. A flexible geared coupling is utilised between the motor and the pump shaft to take care of this small inclination of the pump assembly.

As regards pump coast down performance, a flow halving time of 8 s is presently specified to sustain very short time (transient) power failure (2 s) and to ensure that clad hotspot temperature does not exceed 1073 K (800 °C) in case of on-site power failure and before the reactor is shutdown by the reactor trip signals . However, this calls for a flywheel of large size and weight. The problems of direct mounted flywheel on pump/motor shaft are being studied.

As regards manufacture, pump can be manufactured in the country. Castings in CF3 grade to the required quality can be made through development efforts. The pump shaft can be heat treated and machined to the required tolerances with the facilities available in the industries. The pump motor and associated speed control system is also indigenously available.

4.0 IHX

The principal design options are :

- Primary sodium on shell side or tube side
- Straight tube vs expansion bend
- Tube bundle sizing (tube OD and WT)
- Tube to tubesheet joint

Owing to the familiarity with design of IHX with primary sodium on shell side, from FBTR, the same has been adopted for PFBR. Straight tube design has been favoured from ease of manufacture and cost. Any defective tube to tubesheet weld revealed during manufacturing stage would be repaired by cutting the tube ends leaving the tube in position to provide proper tube bundle belt support. Thus no plugged tube situation is foreseen in design and this eases the duty on straight tube design. It is well recognised that, in light of the Phenix incidents, the thermohydraulics of tube

122

TABLE I. OPERATING CONDITIONS AND MAIN DIMENSIONS OF IHX

Parameter	Data
Thermal Rating, MWt	314.7
Sodium Temperature, K (°C)	
Primary Inlet	817 (544)
Primary Outlet	667 (394)
Secondary Inlet	628 (355)
Secondary Outlet	798 (525)
Tube OD x Thickness, mm	19 x 0.8
Radial/Circumferential Pitch, mm	25/26.2
No. of Tubes	3600
Heat Transfer Length of Tube, m	7.440
Shroud OD, m	1.840
Type of Seal Between Hot & Cold Pool	Mech. Seal
IHX Ring Flange OD, m	2.52
Overall Height , m	15.63
Weight, t	38

bundle with limited shell side pressure drop is extremely important to minimise the thermal loading on secondary sodium outlet header. Variable flow distribution inside the tubes with higher flow on outer row tubes has been incorporated with a distributor plate below the bottom tubesheet to provide dual advantage of minimising the temperature differences between inner and outer shell of secondary sodium outlet header and reducing the temperature difference among the tubes. In addition, a mixing device is incorporated at secondary sodium outlet header as a defence in depth measure.

Tube bundle sizing has been performed with primary side pressure drop limited to 1.5 m from the consideration of design of inner vessel, pump shaft length, main vessel height, external pressure load on main vessel cooling baffles and IHX economics. Based on parametric studies with different tube OD and WT and selection basis as overall economics which includes the IHX capital cost, influence of IHX size on main vessel size and cost, secondary pump capital cost as influenced by design head and amortization of running cost of secondary sodium pressure drop, 19 mm OD x 0.8 mm WT tube size design has been selected. 0.8 mm wall thickness design provides a capital cost advantage of ~ 10 % over 1 mm tube thickness design. Manufacturing capability exists in the country to manufacture resulting IHX design in SS 316LN grade. Mock-up trials will be needed for tube to tubesheet rolled and welded joint.

Fig 2 shows the general assembly of IHX and Table I indicates various parameters. The penetration of the inner vessel by the IHX is provided with a mechanical seal instead of gas seal to minimise the gas entrainment reactivity incidents and offers some reduction in main vessel diameter as IHX size is controlling the main vessel size. Sleeve valve is provided at IHX primary sodium inlet to permit operation at 50 % with 1 secondary loop (with both primary pumps in operation) in case of non-availability of IHX, secondary pump or piping to isolate the IHXs of the affected loop.

5.0 STEAM GENERATOR

The principal design options are:

- Sodium or steam reheat; split-up or integrated in case of steam reheat
- Unit size - Single unit, modular unit
- Tube configuration
- With or without cover gas
- Tube length
- Tube sizing
- Tube to tubesheet joint

5.1 Sodium or Steam Reheat Cycle

Discussions with the turbine supplier have indicated that reliable turbine can be offerred for both the options. With sodium reheat design there is imperative need to have separate evaporator, superheater and reheater. This is not so with steam reheat cycle where an integrated evaporator-superheater can be used. Split-up design offers flexibility in choosing separate tube size for the evaporator and superheater resulting in lesser number of tube to tubesheet welds. This design is good from flow instability considerations as well. Since the steam pressure is 17.2 MPa, the problem of instability is not much. As the cost of SG and associated circuits is 12 % more for split-up design in case of steam reheat, comparison has been made between a sodium reheat cycle with separate evaporator, superheater and reheater vs steam reheat cycle with an integrated unit.

SECONDARY
SODIUM INLET

SECONDARY
SODIUM OUTLET

FLANGE
OD 2520

MIXING
DEVICE

SLEEVE
VALVE

PRIMARY
SODIUM INLET

MECHANICAL
SEAL

SHROUD
OD 1840

TUBE
OD 19 x 0.8THK

PRIMARY
SODIUM OUTLET

THERMAL SHIELD

DISTRIBUTOR PLATE

15630

FIG. 2. Intermediate heat exchanger.

125

STEAM OUTLET

MANHOLE

TOP TUBESHEET

THERMAL SHIELDS

FLOW DISTRIBUTION PLATES

SODIUM INLET

TUBE

TUBE BUNDLE SUPPORT

S G SUPPORT

~25000

SODIUM OUTLET

EXPANSION BEND

THERMAL SHIELDS

BOTTOM TUBESHEET

ORIFICE

MANHOLE

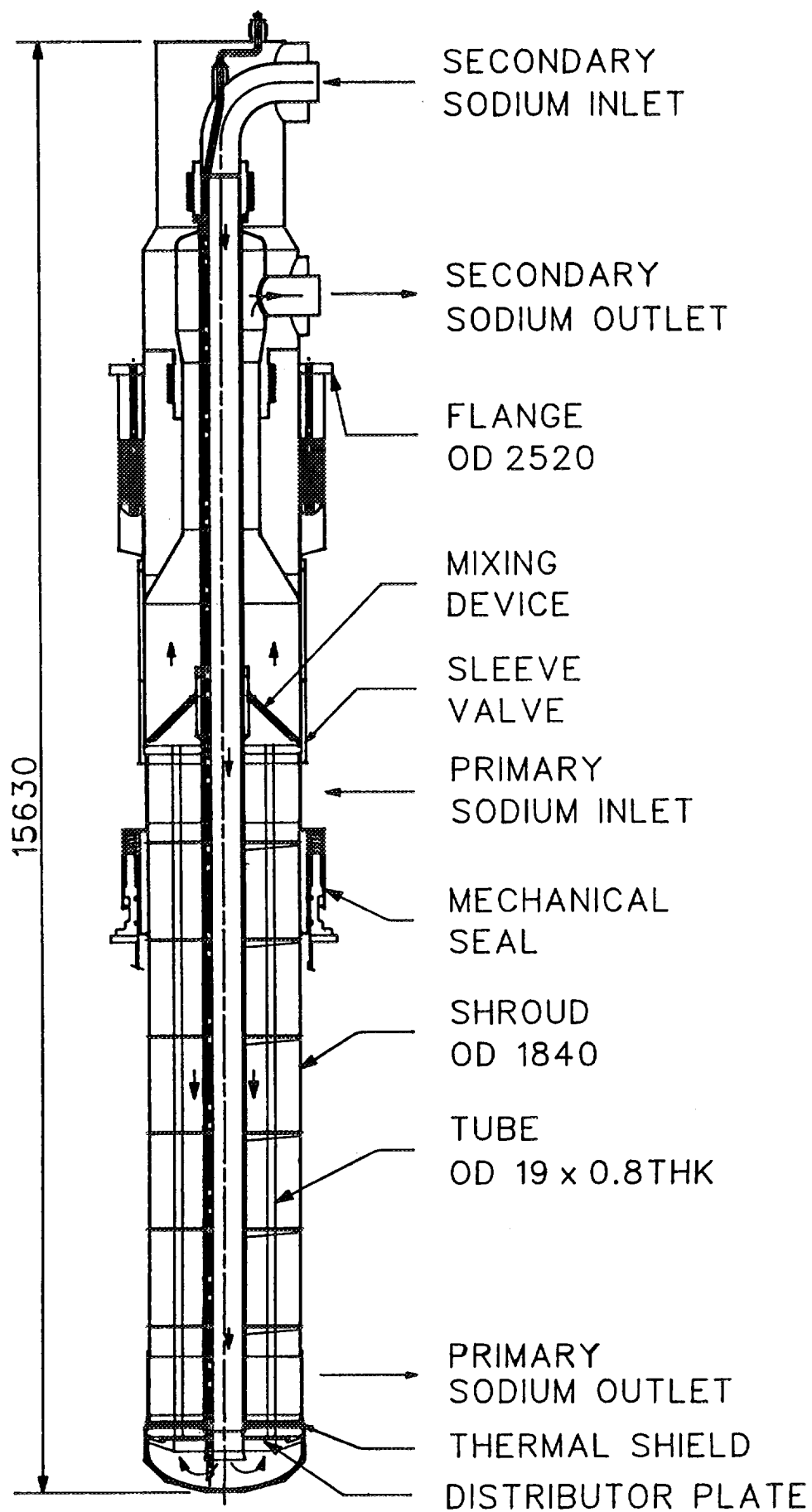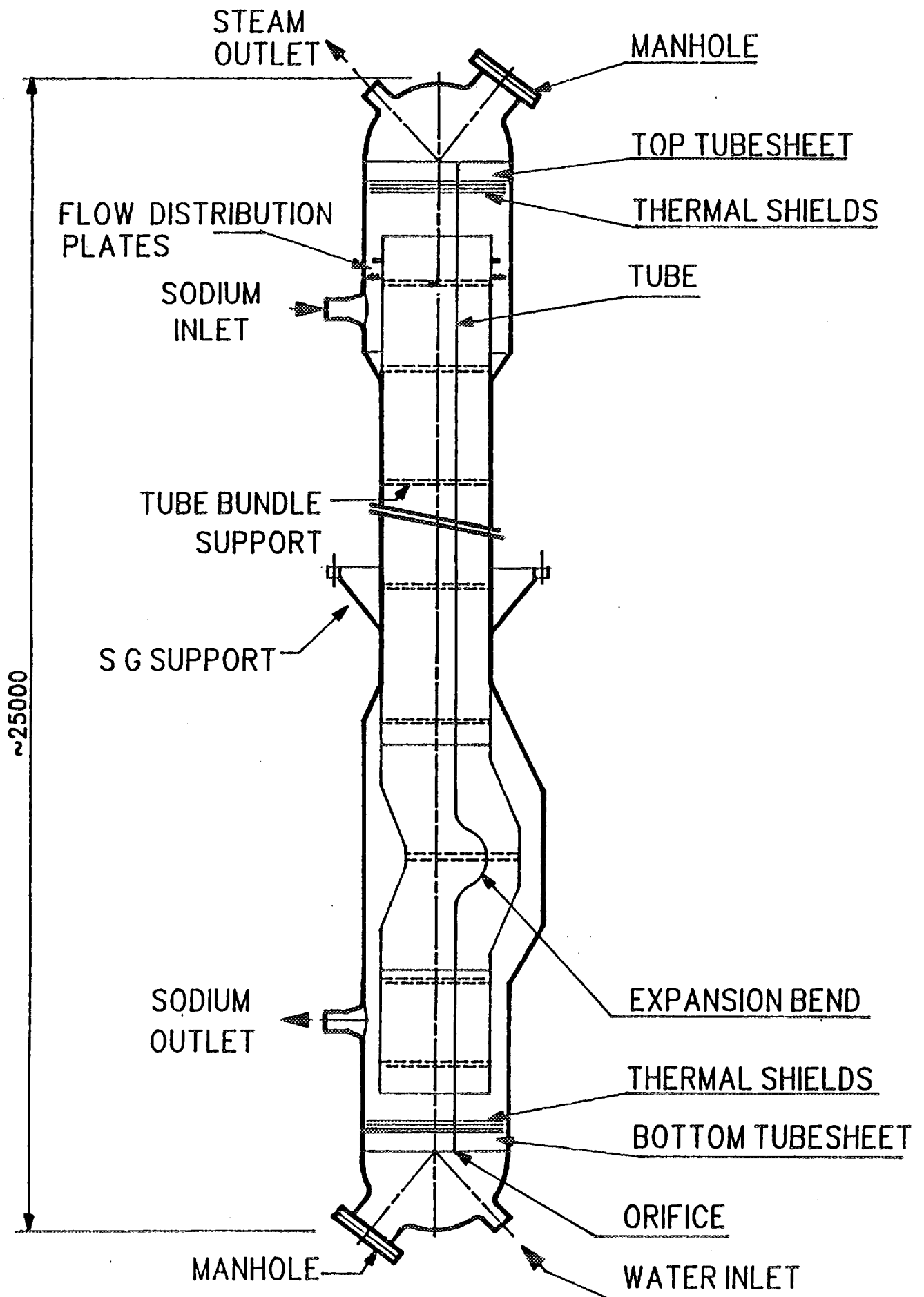WATER INLET

FIG. 3. Steam generator.

Sodium reheat cycle provides advantage of cycle efficiency by 1.33 %, reduction in total number of tube to tubesheet welds by 24 % and marginal reduction in heat transfer area (7 %). It has drawbacks associated with thermal fatigue of superheater reheater junction on sodium side and risk of sodium entry in reheater in case of tube leak during startup/shutdown.

Steam reheat cycle provides advantage in cost reduction for SG and associated circuits, reduction in construction schedule which is important in the Indian context of higher rate of interest, simplicity in plant operation due to reduction in number of SG leak detection circuits and water-steam valves. Steam reheat cycle has therefore, been selected from the point of view of lower construction time and ease in plant design and operation. This option is expected to result in lower project cost once the increased value of interest during construction in sodium reheat option is included.

5.2 Unit Size

From water/steam leak detection sensitivity and design analysis, in particular at top tubesheet-steam outlet junction, small unit has distinct advantage. Rationale behind the selection of 4 SG/loop has been dealt earlier.

5.3 Tube Configuration

Adoption of different tube configurations in LMFBR SG makes the choice difficult. The principal candidate configurations are straight tube (with or without shell bellow or flexible shell), U tube, helical coil, bayonet tube, J tube, hockey stick, serpentine type etc. Configuration of straight tube with an expansion bend in the sodium flow path has been favoured for the following reasons:

- to take care of differential temperature between shell and tube and among tubes
- identical tubes
- ease of inservice inspection
- economics

Straight tube design with no means for accommodating differential expansion or by providing bellows for accommodating differential expansion between shell and tube calls for greater duty on the flow distribution in order to avoid the risk of buckling of plugged tubes. Expansion bend is located in the sodium path instead of locating it at stagnant sodium zone. This helps in minimising ineffective tube length from the point of heat transfer consideration. Its location is in lower portion of single water phase at lower temperature to avoid creep effects.

The possibility of avoiding the expansion bend is being explored.

5.4 Steam Generator With or Without Cover Gas

With the selected option of 4 SG per loop, the next choice is between incorporation of cover gas in SG or provision of separate single surge tank in the upstream of 4 SGs or option with no surge tank.

Incorporation of cover gas blanket simplifies the mechanical design of top tubesheet from the point of fatigue damage due to thermal transients. Along with the incorporation of provision for hydrogen leak detection in sodium in each SG, additional provision for hydrogen leak detection in cover gas facilitates identification of leaky SG under all plant operating conditions.

With cover gas there is an ineffective tube length above the free sodium level, resulting in ~ 10 % increase in heat transfer area and each SG will require additional level probe instrumentation. There is also fear of interaction among gas spaces leading to sodium flow oscillations giving rise to thermal transients in the tubes and shell in the region of sodium-argon interface.

With the design basis leak resulting in the rupture of both the rupture discs, located at sodium inlet and outlet of each SG, studies indicate that the cover gas incorporation in SG does not provide advantage of decreasing pressure transients.

Operating experience on SG with multiple units per loop is available from EBR II, Phenix and BN-600. All these designs are without cover gas in SG. Design of 3 SG/loop for SNR-300 was also without cover gas.

Giving due credence to the operating experience and weighing the pros and cons of incorporation of cover gas in SG, a design without cover gas in SG is selected.

5.5 Tube Length Selection

Maximum tube length available should be used since it would result in lesser number of tubes and hence lesser number of tube to tubesheet welds. Thus a more reliable SG is possible.

Maximum tube length of 23 m has been considered from transportation of steam generator. Single tube of length 23 m is the first choice. In case single tube of length 23 m is not available, design with 23 m tube length made out of 2 tubes is recommended.

5.6 Tube Sizing

Based on parametric studies with different tube size and selection based on capital cost and pumping cost amortized with due consideration to reliability from number of tube to tubesheet welds and respecting the restraints from corrosion, manufacturing and inspection aspects it was concluded that minimum tube size of 15 mm OD x 2.2 mm WT is the most optimum and hence is selected.

TABLE II. OPERATING CONDITIONS AND MAIN DIMENSIONS OF SG

Parameter	Data
Thermal Power MW(total)	1253
Number of loops	2
Number of SG/loop	4
Sodium temperature, K (^{o}C)	
Inlet	798 (525)
Outlet	628 (355)
Water temperature, K (^{o}C)	
Inlet	508 (235)
Outlet	766 (493)
Steam pressure at SG outlet, MPa	17.2
Number of tubes	547
Tube OD x Thickness, mm	15 x 2.2
Pitch of tubes : equilateral, mm	30
OD of sodium inlet box mm (D1)	1150
OD of sodium shell, mm (D2)	800
Overall Height, m	25
Weight of module, t	30

5.7 Tube to Tubesheet Joint

The tube to tubesheet weld has been recognised as the most critical item from the manufacture and reliability of SG. The internal bore weld has been selected over the traditional rolled and welded joint as it eliminates crevices between tubes and tubesheet and permits radiographic inspection.

From the main joint options for the internal bore weld i.e. raised spigot and insert tube type, the former has been selected to place the welds in a low stress region. Also this design provides increased ligament for a given tube pitch as the tubesheet hole is equal to the tube bore.

5.8 SG Description

There are 8 SG modules in 2 loops, each loop containing 4 modules. SG is once through integrated. Fig 3 shows SG configuration and Table II gives the principal parameters.

It is a vertical counter-current shell and tube heat exchanger. Sodium enters through a single inlet nozzle, flows upwards in the annular region and then flows down through the top inlet plenum where it is evenly distributed before entering the tube bundle. After flowing downwards on the outside of the tubes, sodium exits through the single outlet nozzle via the bottom outlet plenum.

Feedwater enters the tube side at the bottom, flows through the orifice incorporated for creating the desired pressure drop from SG stability consideration and flows upward in a counter flow direction to the downcoming sodium.

The tubes are supported at various locations by formed type tube bundle support arrangement made of Inconel 718. The tube bundle support arrangements (at straight tube portions) are themselves supported by 6 tie rods which are fixed to either top or bottom tubesheet. Tubes are also supported at the centre of the bend by Inconel 718 rods to prevent out of plane vibrations. Tube to tubesheet joining is by internal bore welding (raised spigot type). Top and bottom tubesheets are protected by thermal shields to take care of thermal stresses during steady state and transients. Sodium inlet and outlet nozzle-shell junctions are in the form of pullouts. Manhole is provided on water/steam dished heads to allow (1) access to the tubes for the inservice inspection and (2) access for the tube plugging if required. Each steam generator unit is supported by the conical skirt arrangement attached at the centre of the shell.

5.9 SG Manufacture

Tubes in modified 9Cr-1Mo will be indigenously available and order for manufacture of FBTR spare module tubes in this material is already placed with the supplier. Critical manufacturing aspects of tubesheet machining and tube to tubesheet welding and inspection have already been established during the technology development phase of prototype SG manufacture.

6.0 SODIUM PURIFICATION

Ex-vessel sodium purification has been selected for the primary system as an extension of FBTR to avoid R&D associated with in-vessel purification system and to minimise congestion on the roof slab towards ease of maintenance. However, the cooling medium for primary cold trap has been

changed to nitrogen from oil in order to minimise the usage of oil in the reactor because of fire risk. Each of the two secondary loops is provided with an independent sodium purification circuit. Cold trap cooling medium is air and cold trap is envisaged to be regenerated every 6 years.

7.0 SECONDARY SODIUM MAIN PIPING

Secondary sodium main piping is designed on the following guidelines:
- Pipe sizing is based on optimisation studies with maximum sodium velocity of 10 m/s.
- Secondary sodium circuit including piping is not of safety grade classification (i.e. non-nuclear category) as dedicated direct reactor cooling system is provided for decay heat removal. Additional requirements have been imposed on piping expansion stress at equipment nozzles to reduce stresses on equipment and on piping manufacturing specifications because of sodium. This will help in providing compact layout.
- Long horizontal runs are to be avoided to minimise stratification effects.
- Components are located as close as permissible to each other and their support locations are so selected to provide compact piping layout.

Use of bellows instead of expansion loops and use of sodium leak collection trays instead of guard piping for piping outside RCB are under investigation.

8.0 DECAY HEAT REMOVAL SYSTEM

In case off-site power is available, the decay heat will be removed through normal heat transport path of secondary sodium and water/steam circuits. In case of loss of off-site power or non-availability of secondary circuits or water/steam circuit, the decay heat is removed via a Class 1 safety graded passive direct reactor cooling system consisting of four independent circuits of 6 MW nominal capacity each. The sizing of heat removal capacity is based on availability of three circuits i.e. (N-1) to meet the "upset" limits and (N-2) circuits to meet the "emergency" limits. Each of this circuit comprises of one sodium to sodium heat exchanger dipped in reactor hot pool, one sodium to air heat exchanger, associated piping and tanks. Except for the dampers on the air side, this system is passive. In order to enhance the availability of dampers, they are motorised with provision for manual operation. Dampers will be crack opened during normal plant operation in order to have smooth transition to natural convection mode. Possibility of incorporation of EM pumps and blowers in two circuits to aid forced circulation and extent of diversity in the principal components viz sodium to sodium and sodium to air heat exchangers are under investigation.

9.0 PART LOAD OPERATION

Studies have been made for the incidents of one primary pump trip and one primary pump seizure with reactor at full power. In case of one primary pump trip without non-return valve (NRV) the

maximum clad hotspot temperature is only 1026 K (753 °C) and is 1086 K (813 °C) in case of one pump seizure without NRV. These temperatures are acceptable. However, the occurrence of seizure being very rare and also with an idea to reduce the complexities of NRV design and operation, no NRV is provided in the primary pump. Also, the operation with one primary pump is not envisaged as the seizure of the operating pump at 50 % power leads to clad hotspot temperature beyond the acceptable value of 1173 K (900 °C).

Part load operation after reactor shutdown is expected to be feasible in case of non-availability of either a secondary pump or IHX. Part load operation is feasible in case of non- availability of one SG as the temperature difference at IHX primary sodium outlet of the affected and the unaffected loop is estimated to be 25 °C with sodium flow/SG module maintained same and water flow varied in the two loops to result in same steam outlet temperature from SG for both the loops. This differential temperature is acceptable. To facilitate part load operation, sleeve valve on the IHX primary inlet and isolation on sodium and water/steam as well as provision for freezing of sodium by removing insulation between isolation valve and SG are provided.

10.0 CONCLUSIONS

The economic scenario demands to explore avenues for reduction in the reactor capital costs. Design with 2 primary pumps, 2 secondary loops with each loop linked to a pump, 2 IHX and 4 SG has been selected as it combines the benefit of cost reduction wherever possible together with the cautious approach needed for SG.

A spare primary pump, secondary pump and SG are envisaged. Studies concerning part load operation after reactor shutdown in case of non-availability of secondary pump, IHX and SG are under way.

Single suction primary pump design has been selected. IHX design is with primary sodium on shell side and straight tubes of 19 mm OD x 0.8 mm WT rolled and welded to tubesheets. SG design having steam reheat, integrated once-through, straight tube with an expansion bend on each tubes, tubes of 15 mm OD x 2.2 mm WT in T91 grade welded to tubesheet with raised spigot has been selected.

CONCEPTUAL DESIGN OF CORE COMPONENT HANDLING SYSTEM IN PFBR

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Abstract

Core component handling consists of Invessel handling and ex-vessel handling subsystems. In vessel handling comprises of two rotatable plugs and a fixed Transfer Arm which handles subassemblies for invessel transfer between core, Internal storage locations and Invessel Transfer Position (IVTP). Exvessel handling comprises of storing, checking, loading of fresh core subassemblies into the reactor, unloading from the reactor, washing and storage of irradiated core subassemblies under water and shipping to reprocessing plant. From IVTP subassemblies are transferred to Ex-vessel Transfer Position (EVTP) by Inclined Fuel Transfer Mechanism. From EVTP subassemblies are handled by cell Transfer machines.

The various options such as two rotatable plugs, one or two straight pull machines, one straight pull machine and/or one Transfer arm are considered for internal handling. Water pool and sodium storage vessel are the options for irradiated fuel storage. The final choice is one transfer arm and water pool storage. The choice has been made considering safety, simplification and cost reduction aspects. It ensures that the sizes of components such as rotatable plugs, roof slab, main vessel etc are minimised to reduce capital cost of the plant.

1.0 INTRODUCTION:

Prototype Fast Breeder Reactor (PFBR) is a 1250 MWt, 500 MWe sodium cooled, Pool type reactor under design in India. The core component handling system is important because it must be capable of performing the necessary tasks in a reasonable time with required degree of safety. It must have maximum accessibility for maintenance. Core component handling in FBR is always done under reactor shut down condition because of compact core, large reactivity worth, high decay heat of irradiated subassembly and necessity to provide rotatable plugs at the top of main vessel for access to different core positions.

Sodium coolant is opaque, chemically active and becomes radioactive in reactor. No normal visual aid can be utilised. As some sodium remains sticking to SA removed from reactor it cannot be exposed to water or air during handling. Before subsequent operation cleaning of sodium becomes necessary.

During fuel handling, sodium temperature higher than 673 K causes excessive deposition of sodium vapours inside handling machine. 473 K to 523 K temperature is adequate as it provides sufficient margin above sodium solidification temperature also.

The core component handling systems in earlier FBR were designed keeping in view 50,000 MWd/t burnup as target because beyond this, the clad and wrapper materials distorted excessively and the fuel handling forces became unmanageable. With this low level of burnup, large number of fuel SA needed handling requiring frequent shut down. The designers put more emphasis on reducing fuel handling time. In Superphenix two IVTMs and two shuttles are used in parallel. In PFR, attempt was made to minimise the effect of swelling by rotating the fuel SA by 180° within the core when it reached half of the targeted burnup. Gradually improved clad and wrapper materials are put in use and 100,000 MWd/t burnup is attained without much difficulty. Today it appears that 2,00,000 MWd/t burnup can be achieved in near future. Due to this, importance of the fuel handling time is reduced considerably.

Studies done for PFBR has indicated that the shut down period for planned maintenance on yearly basis is more than the time required for fuel handling.

Operating experience of Phenix shows that reactor was shut down for planned maintenance for 24% of total shut down period where as fuel handling shut down was only for 13% of time. Experience from BN 600 also confirms the same trend. Thus it is certain that for the expected burnup, fuel handling can be completed within the time needed for planned annual maintenance shut down if fuel handling interval is about one year. Therefore emphasis has shifted from fuel handling time to simplicity of equipment and overall reduction in capital cost. As the fuel handling operations inside the main vessel are done without any visual aid under sodium it is important that the number of operations and the number of machines must be as small as possible. Reduction in number of machines reduces the capital cost also.

2.0 CORE COMPONENTS HANDLING IN PFBR: (Fig. 1)

The core components which are handled include fuel, blanket, control and shielding subassemblies. Like all other fast reactors, core components are handled with reactor shut down and when sodium temperature is brought down to 473 K (200° C).

The handling system has been divided into two parts i.e invessel handling and exvessel handling. Invessel handling is done with the help of two rotatable plugs and an invessel transfer machine which is off set (fixed) arm type. This configuration has been selected to give reduced diameter of main vessel as compared to straight pull type machine. The weight of SA varies from 1800 N to 3500 N. The maximum extraction and insertion forces taking into account the deformation are estimated to be 15 KN and 10 KN respectively. The reactor is designed for life of 30 y. Over its lifetime 4250 SA will be handled. The machines have been designed for handling 10,000
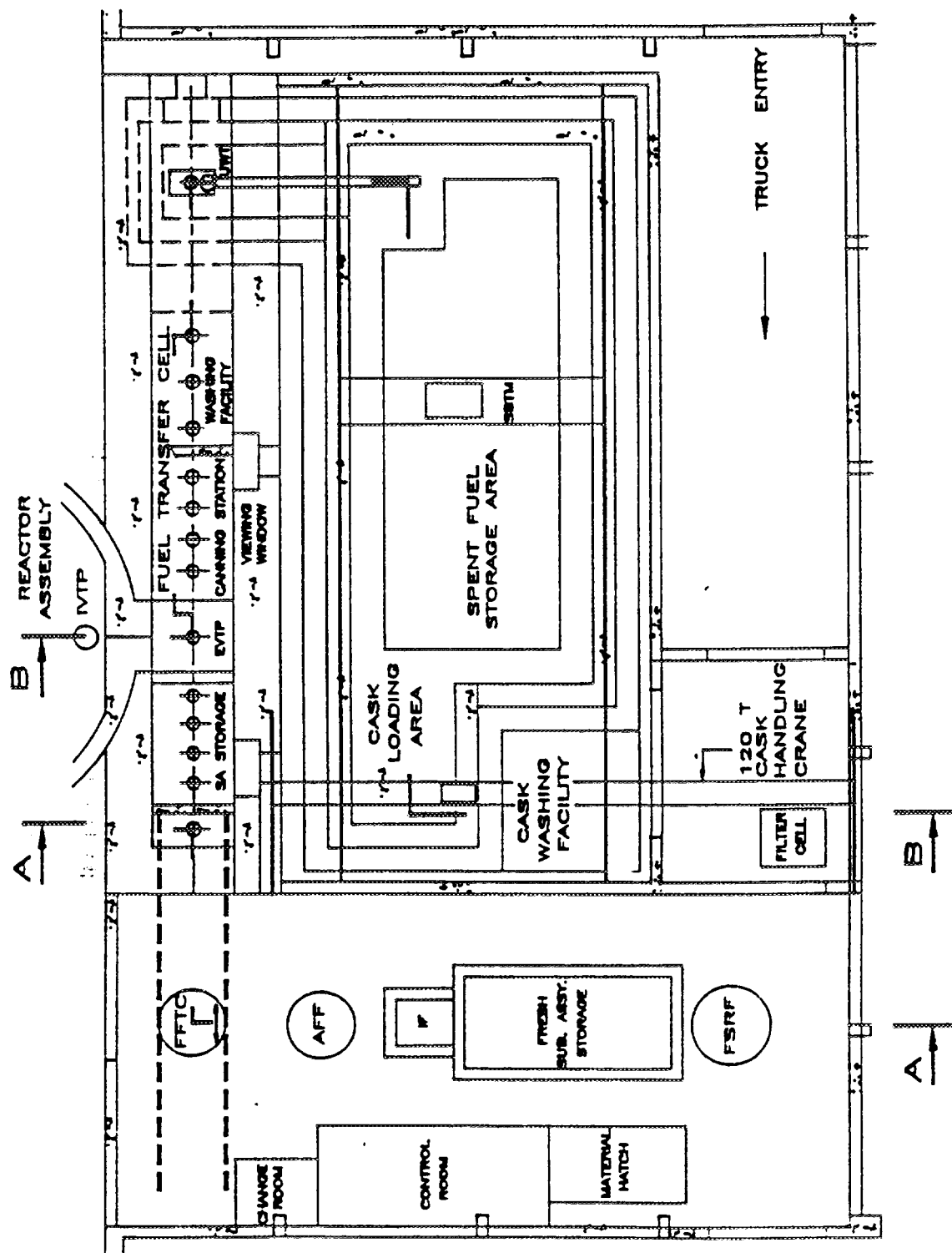
FIG. 1. Layout of fuel handling area.

135

TABLE I. CONCENTRIC CORE AND TWO ROTATABLE PLUGS

	One SPM	One TAM	Two SPM	One SPM + one TAM
Dia. of SRP flange (mm)	4955	4685	4870	4530
Dia. of LRP flange (mm)	7700	6930	7300	6470
Dia. of Main Vessel shell	13700	12900	13300	12450
No. of SAs needing double handling	NIL	NIL	169	19
Time taken to replace one spent SA with one fresh SA				
Double handling (min)	--	--	262	262
Single handling (min)	184	184	184	184

SA. The radioactivity of spent fuel SA varies from one to seven million curies. The invessel handling machine must be designed to take care of these conditions.

The decay heat of fuel SA after 1,00,000 MWd/t burnup is 4.3% and 2% of its full power after 1d and 10d of shut down respectively. In PFBR these are 35 KW and 15.5 KW respectively. Fuel SA with this decay heat needs forced cooling when it comes out of sodium of main vessel. This makes the design of fuel handling system more complicated. Sodium vapours present in cooling gas create additional maintenance problems. To avoid such problems and to simplify the fuel handling system invessel storage has been provided at periphery of core. Reduced sodium flow is provided at these locations. Spent fuel discharged from core is stored here till next fuel handling compaign when its decay heat reduces to below 5 KW. At this level of decay heat no forced cooling is required normally during transit of SA from main vessel to external storage. In case it is stuck in cover gas of transfer cell where the heat transfer condition is the most adverse, argon jet cooling has been provided. It keeps the temperature of clad below the maximum permissible limit of 923 K (650 Deg.C). Subassemblies other than fuel are not stored in internal storage as their decay heat is less than 5 KW.

Fuel handling is done after 240 calender days of reactor operation. This is equivalent to 180 EFPD. 62 fuel, 25 blanket and 5 control subassemblies are replaced during every fuel handling campaign. The control philosophy is planned to be based on 1/2 or 2/2 logic depending on the importance of the sequence. If all the steps needed for fuel handling are done by automatic control system it is estimated to take 110 min for replacement of

one fuel SA in the core by a fresh one. A semiautomatic approach has been adopted for PFBR so that periodic control by the operating personnel is done after a set of steps. A number of consecutive steps have been grouped together. Steps within a group are done automatically. With semiautomatic operation, replacement of one spent fuel subassembly with a fresh one takes about 3 h. It takes about 18 d for each campaign. Shutdown period is slightly longer as planned maintenance work is also carried out simultaneously during shut down period.

2.1   **FRESH SUBASSEMBLY HANDLING**: (Fig. 2)

Fresh SA is handled in Fresh SA storage area which has following facilities;

- Fresh Subassembly Receiving Facility (FSRF)

- Fresh Subassembly Storage (FSS)

- Inspection Facility (IF)

- Argon Filling Facility (AFF)

The fresh SA is removed from the shielded shipping cask in which it arrives at the plant and checked in Inspection Facility. Their weight, over all dimensions and absence of gross blockage of flow path in them are ascertained.   They are then stored in underground fresh SA storage in individual containers.   The ceiling of the storage cell provides necessary shielding for the operating personnel, against gamma and neutrons to limit the dose to 10^{-2} m Sv/h   (1 mR/h)   on   contact   and 10^{-3} m Sv/h (0.1 mR/h) at 1 m from surface.   The shielding has been designed for a neutron source of 4 X 10^{6} n/s per SA.

During preparation for refuelling campaign the SA is removed from fresh fuel storage with Fresh Subassembly Transfer Flask (FSTF) and introduced into Fuel Transfer Cell with the help of Transfer Chamber after repeating the checks in Inspection Facility.   The air in Transfer Chamber is evacuated and argon is filled into it at Argon Filling Facility.   The Transfer cell is always kept under atmosphere of argon.   The mobile   Transfer chamber makes a leaktight coupling with a port provided at the bottom of Transfer cell and Fresh subassembly Transfer machine picks up the subassemblies into the cell and keeps it in a temporary storage inside the cell.   Two transfer chambers are used to facilitate fuel handling operations.   The cell has two identical transfer machines.   One for fresh and other for spent subassemblies.   In case of break down of one, second can perform both the functions.   No preheating of fresh SA is envisaged as it can withstand the thermal shock produced due to insertion in molten sodium of EVTP.

2.2   **INVESSEL HANDLING**:

Handling of core SA inside the main vessel consists of transfer between the core, internal storage and invessel transfer position.   Two rotatable plugs and one or two In-Vessel Transfer Machines (IVTM) were considered.
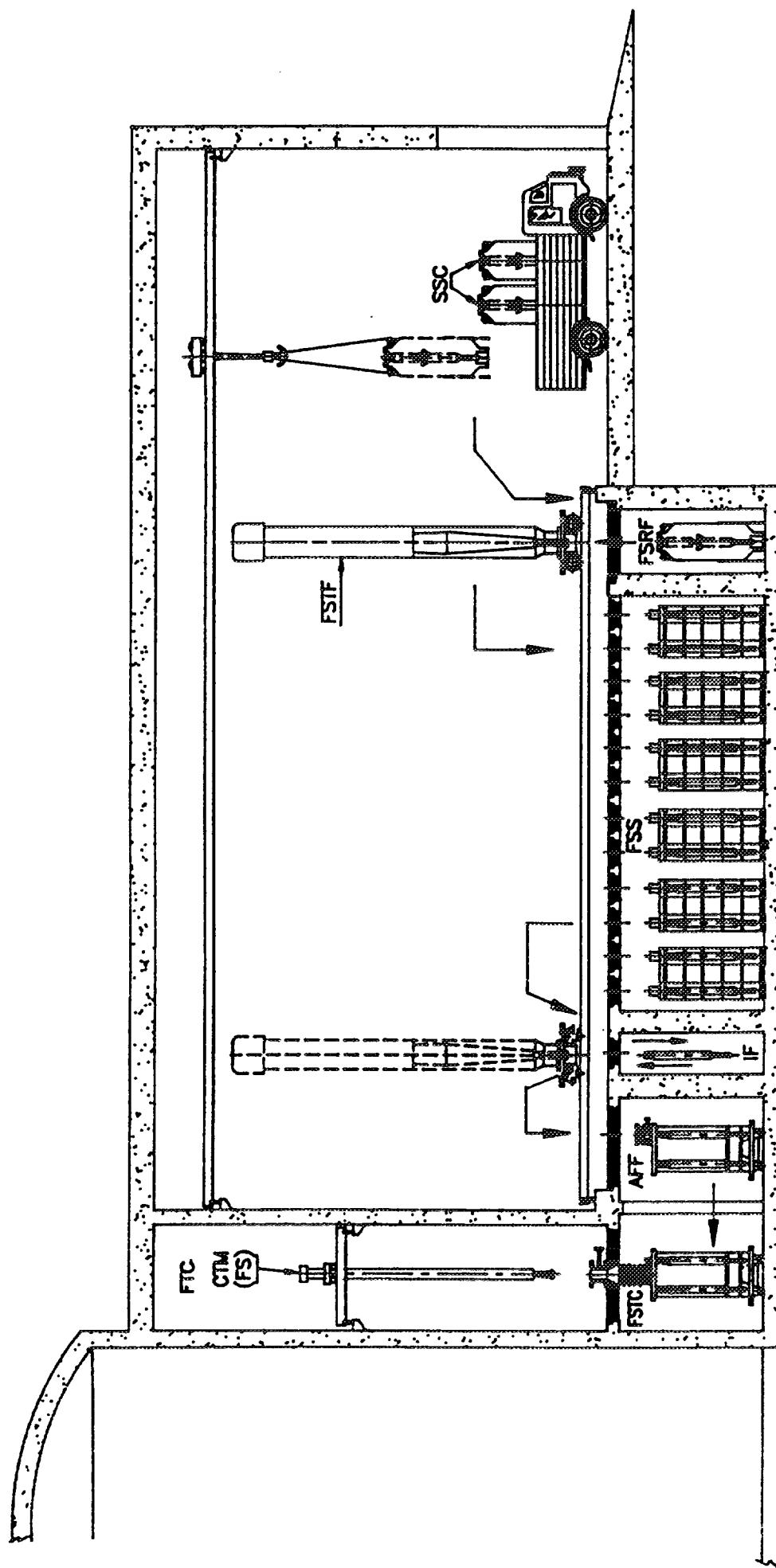
FIG. 2. Fresh subassembly handling arrangement
(section: AA).

138

A fixed offset arm type IVTM is preferred over straight pull type as it needs smaller diameter of main vessel and LRP though it necessitated the addition of auto-orientation to each subassembly and called for free rotation of lower part of gripper. Another option with one straight pull machine and one off set transfer arm would have further reduced the main vessel diameter. But in this option some of the SA located in inner portion of core need double handling as they are transferred between core and an intermediate location by straight pull machine and then further moved to internal storage by transfer arm. As this option involves additional operations inside main vessel and need development, operation and the maintenance of two different types of machines and the cost benefit is marginal, this option is not selected. The option of using two straight pull machines was discarded as it needed larger main vessel and LRP, SRP diameters and necessitated double handling of large number of fuel SA.

The entry and exit of subassembly from in vessel transfer position inside the main vessel is done through Inclined Fuel Transfer Mechanism (IFTM).

Different concepts were considered for IFTM satisfying the various constraints. The main options were Swing Type system, Rotating 'X' type system and Rotating shielded leg system.

The Rotating shielded leg system is finally selected based on following main benefits;

a)   Economic, because of lower overall size.

b)   It affords better safety provision and maintainability. The system, however is slower than the other two systems. But it does not affect the reactor shut down period.

IFTM consists of a primary ramp which is inclined at an angle of 17° with vertical plane, a Rotating Shielded Leg (RSL) on top of ramp which pulls up or lowers a sodium filled pot containing subassembly into the main vessel through the ramp and another inclined secondary ramp which connects RSL and Ex-Vessel Transfer Position (EVTP). EVTP has cooling/heating arrangement which is switched on if the SA is stuck and has to remain there for prolonged time. Heating or cooling is resorted to maintain sodium in molten condition or to maintain the pin clad temperature within specified temperature limit respectively.

2.3   **FUEL TRANSFER CELL (FTC) :**

It is a rectangular cell in fuel handling building with concrete walls lined with stainless steel. All the subassemblies pass through this cell. It is in communication with main vessel through IFTM, with Fresh fuel storage through Transfer chamber and Spent fuel storage through inclined under water trolley (UWT). It houses two identical cell tranfer machines for handling core SA, washing facility, temporary storage positions for pot and fresh SA. As the spent fuel SA picked up by Cell Tranfer Machine from EVTP has sodium and its atmosphere comes in contact with that of main vessel cover gas, the cell is kept filled with argon.

Argon pressure is maintained at about 1 KPa (10 mb) above covergas pressure of main vessel. This prevents spread of contaminated cover gas into the cell and minimises tranfer of water vapours from Spent Fuel Storage Bay (SFSB) and sodium vapours from EVTP to the FTC. All the working positions in the cell are provided with gate valves to isolate the cell from external atmosphere. All the operations of the cell are done remotely from the control panel.

Sodium of the spent fuel SA is washed in washing facility by circulating nitrogen, water mist mixture. As washing takes more time, three washing positions have been provided in FTC.

Both the cell transfer machines are identical straight pull type and can reach any working position in the cell. They operate simultaneously, but if one is not available, the other can complete the campaign though it may take additional time. There is provision for argon flow in the gripper which cools the SA in case it gets stuck in the cell. A trolley carries one SA at a time from FTC to a pool type Spent Fuel Storage Bay (SFSB).

To prevent the spread of contamiantion failed SA is not washed and is stored in a leaktight pot filled with demineralised water. Storage Bay Transfer Machine (SBTM) handles SA between under water trolley, storage racks and shipping cask.

The design of the system is such as intervention in FTC and SFSB will be possible in case of failure of any of its components.

2.4 SPENT FUEL STORAGE BAY (SFSB) (Fig. 3)

Here two options, sodium tank storage and water pool storage were considered. In case of former, the design of vessel, its cooling and heating circuits and other components is complex because of large hold up of molten sodium in the vessel. Maintenance of components working in sodium is also difficult. Therefore, water pool type storage is selected even though its needs more space.

It is a demineralised water filled double concrete walled tank. Inner wall is lined with stainless steel liner. Stainless steel channel network touching the liner helps in detecting any leak from the tank. Inner space between the two walls provides access to the personnel for inspection and isolates the pool from cells located around the pool. The bare SA is stored in racks. The spacing is such as to maintain Keff less than 0.8. The capacity of pool is for 5 years of discharge from reactor plus one emergency core unloading.

2.5 IN-VESSEL TRANSFER MACHINE (IVTM): (Fig. 4)

It transfers the core components such as fuel, blanket, shielding and control subassemblies between core, IVTP and internal storage. It is off set arm type where the off set is 592 mm. It is permanently located on small rotatable plug and is raised up by about 3.5 m during reactor operation to avoid activation of its gripper and guide tube. About 1 m of its
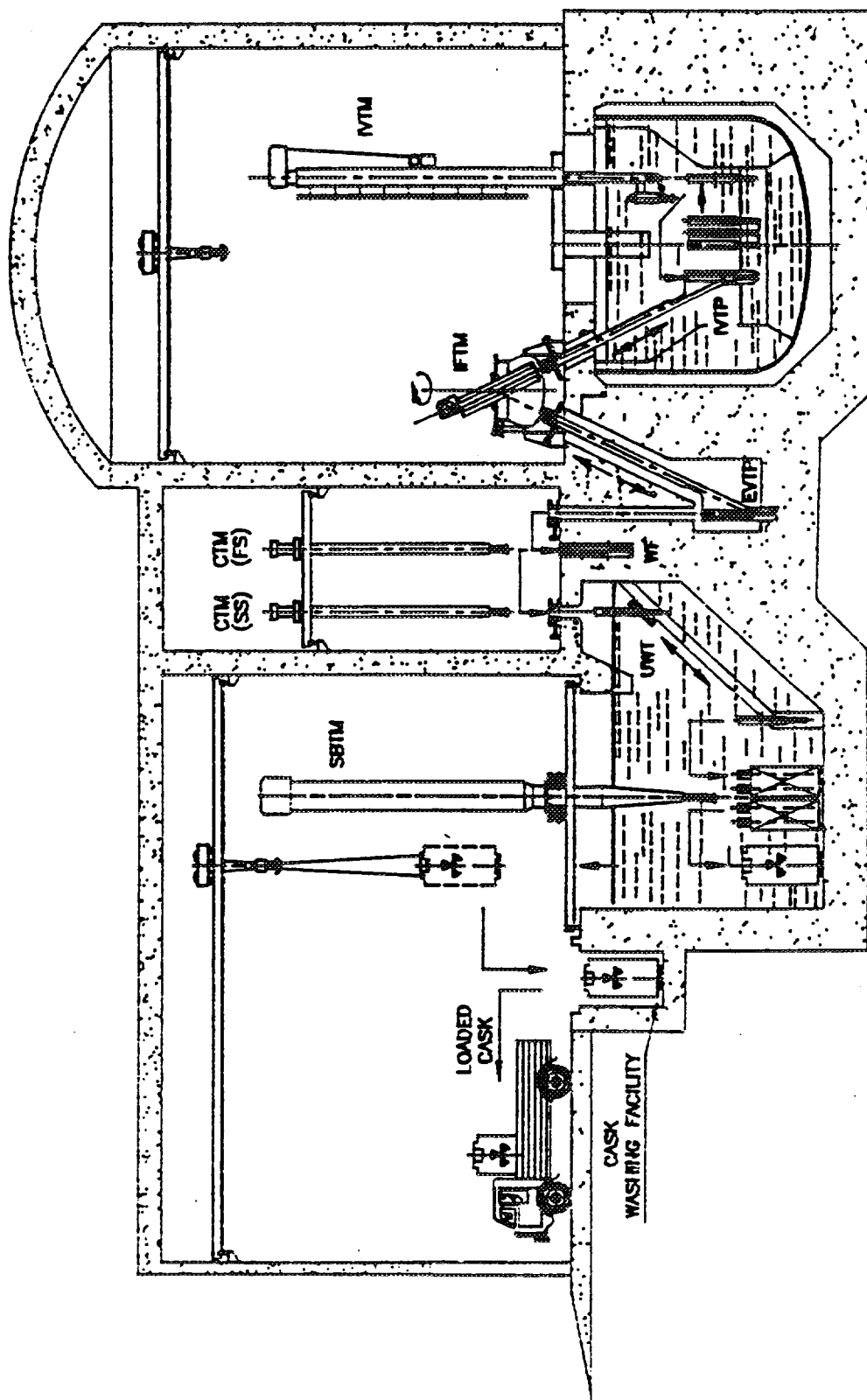
FIG. 3. Spent fuel handling arrangement (section: BB).

TENSION SENSING DEVICE

LOAD CELL

TOP STRUCTURE

44520

SAFETY BRAKE

HOISTING DRUM

HOISTING MECHANISM (HM)

TRANSFER ARM ROTATING MECHANISM (TARM)

SHIELDING ARRANGEMENT

γ-DETECTOR

GRIPPER

30500

GUIDE TUBE

27530 Na LEVEL (DURING FUEL HANDLING)

24300

23200

GEARED MOTOR

SYNCHRO

HM WITH SYNCHRO

GEARED MOTOR
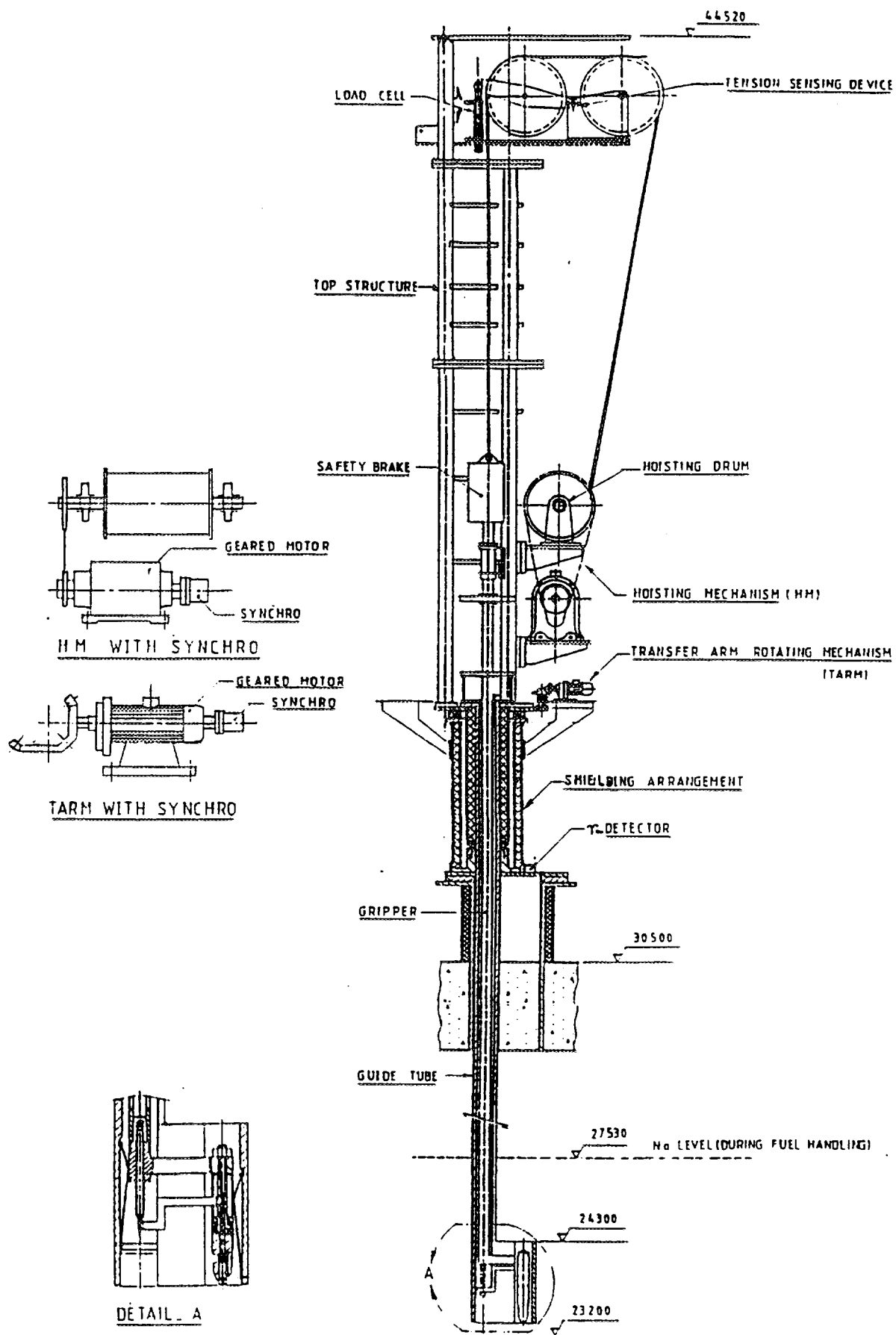
SYNCHRO

TARM WITH SYNCHRO

DETAIL A

FIG. 4. Transfer arm.

lower end remains immersed in sodium to keep stainless steel bellows of the gripper always immersed in sodium. This prevents exposure of sodium contaminated bellows to argon cover gas.

The machine has been designed for the following conditions:

- Weight of heaviest SA to be handled (steel reflector) } = 3500 N

- Weight of fuel SA } = 2400 N

- Deviation on elevation of head of SA due to tolerances, thermal expansion swelling etc. } = + 20 mm − 5 mm

- Permissible misalignment in core } = 35 mm

- Maximum insertion force on SA } = 10 KN

- Maximum extraction force on SA } = 15 KN

As it is not possible to reorient hexagonal sheath of SA by rotating the machine around its own axis auto-orientation has been provided on the head and foot of each SA. The lower part of gripper is mounted on roller bearings and rotates alongwith SA automatically when subassembly is being lowered into the core.

To differentiate control rod from other SA the level of control rod is kept lower by 40 mm with reference to nominal level of other core SA. In addition, the weight of gripper alogwith the SA held by it is constantly monitored to distinguish between various SAs. There is a tension sensing device which cuts off the motor if there is excessive load on the wire rope. A self energising brake has been provided on the gripper which limits the free fall of gripper under gravity to less than 500 mm in case of failure of hoist rope. All the operations of the machine are done remotely from a control panel.

2.6 INCLINED FUEL TRANSFER MECHANISM (IFTM):

IFTM will be installed on reactor roof slab within the reactor containment building. Its total height from the bottom of secondary tilting mechanism to top of hoisting arrangement will be 24 m approx. (Ref. Fig. 5). It will weigh 265 t. Design has been done in such a manner that no component will weigh more than 20 t. The internals of this equipment coming in contact with argon and sodium vapour will be made of austenitic stainless steel.

It consists of primary and secondary ramps, Rotating shielded leg (RSL) and tilting mechanisms. The tranfer pot can be tilted to the vertical by gravity in the tilting mechanism. The transfer pot can be hoisted up in the ramps and into a shielded leg which is mounted on a support table resting on a large slewing ring. In between a leak tight container (LTC) is there which, along with bellows and gate valves on two sides and a shielding plug on primary side, forms a leak tight enclosure
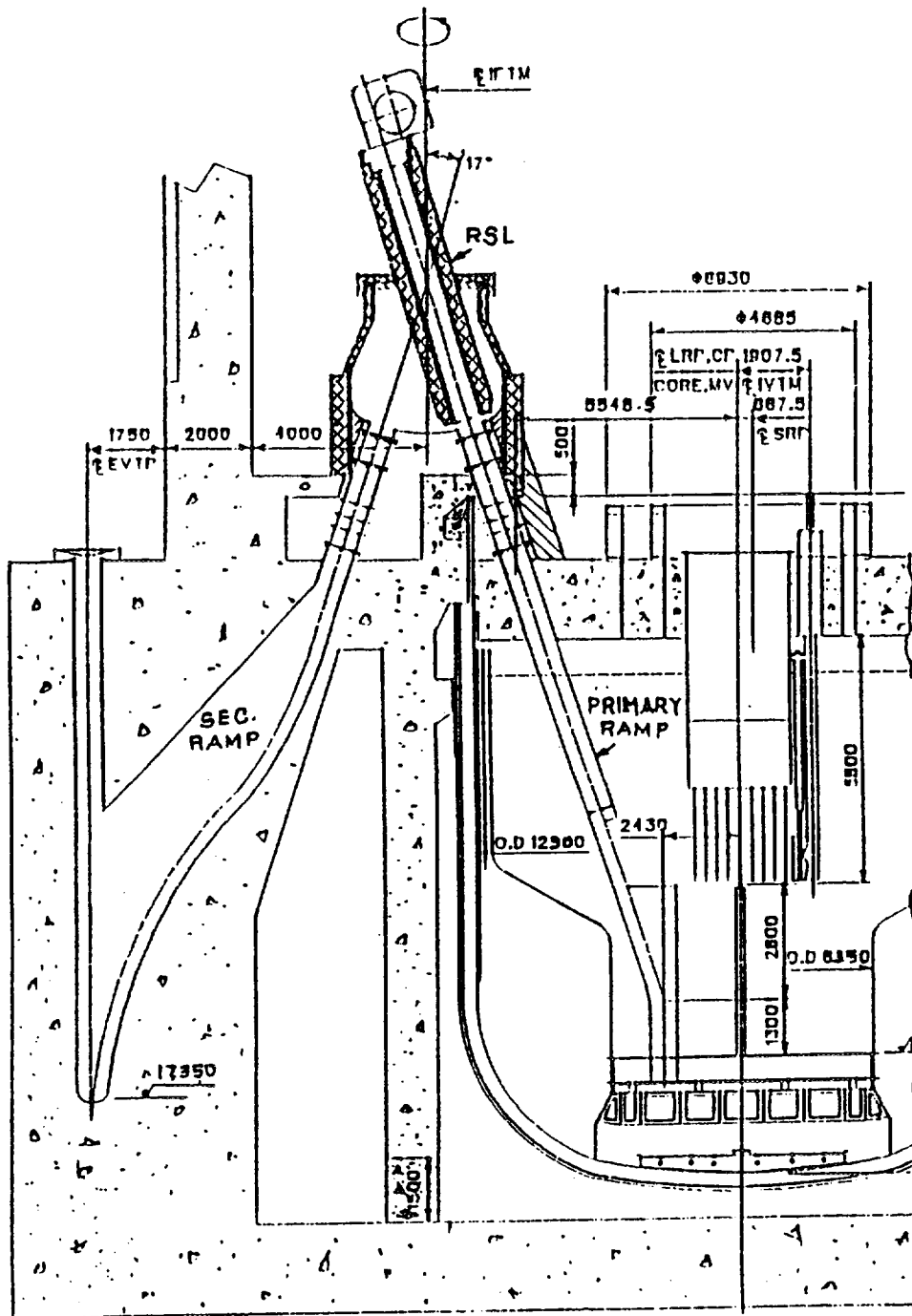
FIG. 5. IFTM.

for the argon & sodium vapours. The shielding plug is provided to allow for maintenance of primary gate valve and to protect it from direct radiation and heat. Bellows allow for thermal movement of LTC and other components. The hoisting/lowering of transfer pot as well as rotation of shielded leg will be by oil hydraulic motors through reduction gear box. IFTM will be provided with two heaters each of 10 KW for air and argon heating to facilitate preheating of internals before refuelling operations. Hot argon will be circulated inside the LTC and other parts and hot air outside them for the purpose.

Provisions have been made on transfer pot and ramps to arrest motion of transfer pot in case the hoisting link fails. This is to prevent damage to reactor internals since the transfer pot can attain a velociy of about 3 m/s in its fall. Along with safety brake a shock absorber has been provided to reduce the impact. Transfer pot will also have an anti-evaporation lid to reduce sodium vapour deposition on the internals and a siphoning arrangmement to prevent spilling of sodium during rotation. An internal shock absorber is also provided to absorb the shock due to accidental fall of a core sub-assembly from a height into transfer pot.

Tilting mechanism does not have any moving components in it and the tilting of transfer pot will be effected by gravity only.

The design takes into account the large amount of thermal movement which will take place at IVTP from fuel handling to reactor operating temperature. It also takes into account the thermal movement of LTC which affects alignment of shielded leg with ramps. A special eccentric sleeve has also been introduced to correct the misalignments introduced because of cumulative effects of dimensional tolerances of various large components during assembly.

IFTM has been designed to guard against following different types of abnormal situations and errors which can take place:

a) Simultaneous movement of transfer pot and rotating shielded leg.

b) Hoisting/lowering of tranfer pot in unaligned position.

c) Movement of tranfer pot with gate valve closed.

d) Stalling of transfer pot in ramp or rotating shielded leg with hot fuel sub-assembly inside.

e) Hoisting link failure.

Apart from the above, the design of IFTM allows remote unlocking of ramps to facilitate removal of transfer pot and ramp with spent fuel sub-assembly inside without need of a separate handling flask. Also provision for evacuation of sodium from the transfer pot before it can be removed for any maintenance work has been kept.

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CONCEPTUAL DESIGN OF SAFETY INSTRUMENTATION FOR PFBR

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Abstract

Instrumentation systems enable monitoring of the process which in turn enables control and shutdown of the process as per the requirements. Safety Instrumentation due to its vital importance has a stringent role and this needs to be designed methodically. This paper presents the details of the conceptual design for PFBR..

1. INTRODUCTION

Safety Instrumentation is designed to ensure safe operation of the reactor in all its states of operation such as normal operation, anticipated operational occurrences and accident conditions. For this, various parameters required to be monitored are identified. The options for the types of sensors and processors, their redundancy and diversity are evaluated. Finally, the adequacy of the safety instrumentation provided is evaluated. This paper presents these details.

2. DESIGN CRITERIA FOR SAFETY INSTRUMENTATION

Safety Instrumentation is designed to protect the plant from various postulated initiating events. For this purpose monitoring of the following is done: Neutron flux, Sodium flow through the core, Temperature at the core inlet and outlet, Failure of fuel pin.

The systems that monitor these safety related parameters have been provided with redundancy, diversity, fail-safeness, on-line testability and maintainability.

Two reliable and independent shutdown systems are provided, each of which has its own set of absorber rods.

Shutdown is effected by two means, the fast shutdown (SCRAM) by dropping of absorber rods into the core by gravity and gradual shutdown i.e. lowering of these absorber rods into the core.

The parameters for shutdown are duplicated for the two shutdown systems. SCRAM occurs from both the systems. However, controlled shutdown is initiated by only one system.

There is a main control room and emergency control room. Safe shutdown of the plant will be effected from emergency control room whenever it is not possible to exercise control from the main control room.

The neutron flux monitoring system meant for shutdown is triplicated and operates in a two out of three mode with a hot standby.

For core temperature monitoring, two sensors are provided for each subassembly.

The hardware of the neutron flux monitoring systems of the two independent safety monitoring systems are located in separate rooms and independent power sources.

3. PARAMETERS TO BE MEASURED

Following are the parameters that are proposed to be monitored :

1. Neutron flux - from shutdown to 150% of full power.
2. Rate of change of neutron flux i.e. period, form +3 sec to ∞
3. Temperature at the inlet of the core and the outlets of each subassembly.
4. Inlet flow to the core.
5. Failure of a fuel pin
6. Sodium leak and Steam Generator leak.

4. NEUTRON FLUX MONITORING

The fastest means of monitoring the fission reaction in the core is by neutron flux measurement. Neutron detectors placed in the closest vicinity enable this. These signals are processed by pulse or dc electronic channels in the control equipment rooms. The signals from these channels enable shutdown of the reactor, control of power and also provide data for analysis.

4.1 Range of Flux Measurement

The thermal neutron flux at the core center varies from 1.92×10^3 nv at shutdown to 2.5×10^{13} nv at full power. Locations around the reactor vessel have so low fluxes that even the most sensitive neutron detector cannot measure this flux particularly in lower ranges. In view of this, a location under the reactor vessel where the thermal flux at shutdown is around 0.2 nv and at full power is 2.9×10^9 nv has been considered in addition to the locations in the core. The radial flux profile under the reactor vessel does not vary significantly and thus the neutron flux measurement at that location is uniform.

The choice of monitoring this 10 decade long neutron flux is either to have in-core neutron detectors or detectors below the vessel.

The high temperature (550° C) and high gamma activity (10^5 R/h) at the core center prevents using any other type of detectors than the fission counters which require to be operated in pulse mode during low flux and dc mode during power range with electronics designed on Campbell mode. Since permanently installed neutron detectors in the core for monitoring results in non-availability of a valuable subassembly location, this option was not pursued. However, for initial start-up, neutron detectors are to be installed in the core. As these are to be utilized at lower temperature and gamma activity and will be withdrawn later their requirement is not stringent.

The most sensitive fission counter (1 to 2 cps/nv) can only give 0.2 to 0.4 cps. Helium-3 detectors give 50 cps/nv sensitivity and give around 10 cps at shutdown with full core. These cannot withstand higher gamma activity and also the pulse channel associated with them get saturated. Thus compensated ionization chamber (CIC) of sensitivity 10^{-14} A/nv along with dc channels have to take over from this range.

The possibility of providing neutron guides to achieve higher flux at detector location (below the vessel) was studied so that fission counters which withstand high gamma can be

used. The study indicated an increase of 3 to 10 times the neutron flux at the detector location with neutron guide. This is still not considered adequate and thus this option was not pursued.

Figure 1 shows range of coverage of neutron flux and the detectors.

4.2 Special Instrumentation for Start-up

For the initial core loading and first start-up stage of the reactor, two options are considered for flux monitoring. These are, provision of in-core detectors and their channels and provision of a strong neutron source giving adequate flux at the ex-core location. The latter option requires a source with a strength of 10^{11} n/s in the core.

In-core fission counters with sensitivity of the order of 0.1 cps/nv are required to be located at core center during this stage. safety actions are actuated from these channels during this phase of operation and then transferred to ex-core helium-3 counter channels after obtaining sufficient counts. These in-core fission counters are withdrawn when core reaches equilibrium. A special fuel subassembly that accommodates three fission counters and also fuel pins is provided at the core center.
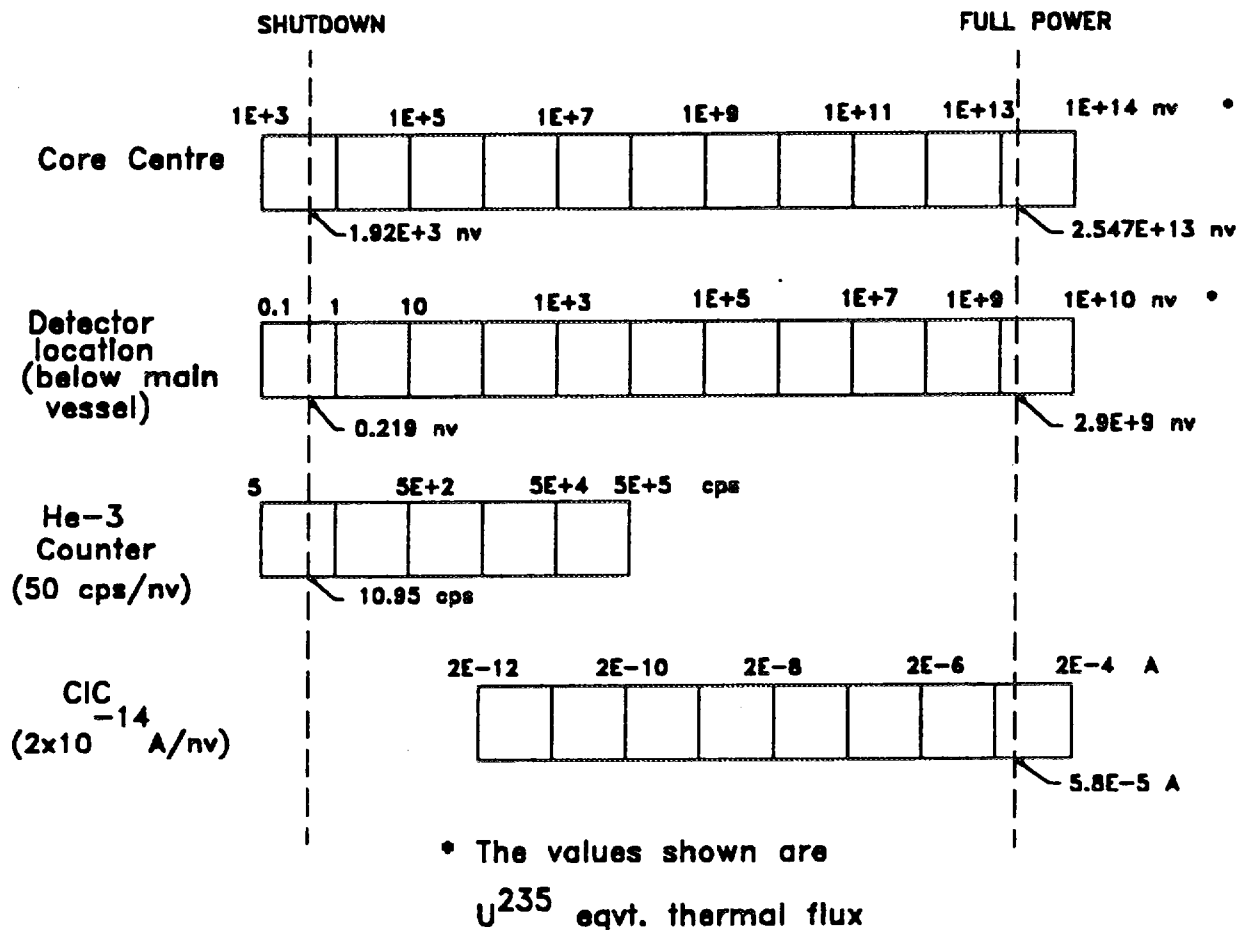

FIG. 1. Neutron Flux-Coverage.

4.3 Configuration of Neutron Detectors

As it is required to have total independence of the two shutdown systems, separate neutron detectors are provided for each shutdown system. As the protection logic envisaged is 2 out of 3 with a hot standby, 4 detectors are provided in each range for each shutdown system. The configuration proposed is shown below :

	Range	Type of Detector	No. of Detectors
1.	Initial Start-up	In-core fission counter	3
2.	Start-up	Helium-3 prop. counter	4 + 4
3.	Intermediate & Full Power	Compensated ion chamber	4 + 4

4.4 Parameters Causing Shutdown from Neutron Flux Monitoring System

The outputs of helium-3 counters are processed by pulse channels and those of the CICs by d.c. channels. The actual hardware for each of the channels can only be decided at the appropriate stage taking into account the reliability and state of art. For achieving diversity the comparators for the first system are analog and the second digital.

In the start-up range, SCRAM on high counts (LOG N) and low period (TN) are provided. In the power range, SCRAM on high power measured by the logarithmic and linear processing channels (Log P & LIN P), low period (TP) and reactivity (RHO) are provided.

5. CORE TEMPERATURE MONITORING

Measurement of core temperature provides information on the heat generated in fission and the flow blockages. Though neutron flux measurement enables a fast means of shutdown, shutdown by measurement of temperature is also considered necessary as a diverse method. This also enables measurement of flow blockages. The individual fuel element failure detection enables detection of clad ruptures but not all blockages.

RTDs are ruled out for fast reactors as they undergo transmutation in the fast neutron and high gamma fields. Gamma heating of the resistance element drastically changes the temperature resistance characteristics thereby affecting accuracy.

The passive K-type chromel-alumel thermocouple with a response time of 4 to 6 sec., even though slow, becomes the inevitable choice. It withstands radiation upto 10^{14} nv without any transmutation. Exposed junctures of thermocouples may suffer mechanical damage due to corrosion due to long duration of operation, their small size also poses some problems in insertion. In view of this, the thermocouples are housed in thermowells. The grounded junction thermocouple offers faster response but provides no isolation of lead wires from static noise pick-up. Thus ungrounded sheathed thermocouples are preferred.

Two thermocouples are used for each subassembly. As thermocouples are considered passive, no separate thermocouples are provided for two shutdown systems.

Two independent dedicated fault tolerant computers are provided. Each of them receiving the temperature signals from both the thermocouples of a subassembly. Each of

150

these computers give the signals for both the shutdown systems, but after validating the two thermocouple signals thus acting in a 2 out of 2 mode.

The three shutdown parameters generated from each computer are - increase in outlet temperature of an individual subassembly, increase in mean temperature of the outlets and rise in temperature between the mean outlet and inlet. For the purpose of inlet temperature signal, thermocouples are provided at discharge of the primary pump.

Individual subassembly monitoring provides information on adequacy of cooling for each subassembly. The other two signals give protection against global power rise or flow reduction.

6. FAILED FUEL DETECTION SYSTEM

The detection of a fuel pin failure is necessary to avoid contamination of the primary circuit, fuel and cooling disturbances in the subassembly which may occur due to deposited fuel particles. The failure of a fuel pin can start as a pin hole leak and end as an open hole in the clad. Both these stages cannot be detected by the same sensors. Leaker failure can be detected as increase in the fission gas activity and failure exposing fuel to coolant can be detected by monitoring delayed neutron activity of the precursors.

6.1 Delayed Neutron Detection System

Delayed neutron detection can only be done by taking a sample of sodium away from the pool as the prompt neutron flux is high in the pool. The other option is to find a location in the pool where prompt neutron flux is low.

The locations in the pool behind the heat exchangers offer a good choice for this. However, only fission counters that can withstand high temperature can be considered for this. These signals need to be transmitted by mineral insulated cables to the electronics located outside. In view of this, measurement of delayed neutron activity by a sample taken from the main vessel and brought above the vessel is preferred.

In view of the dilution of the delayed neutron signal in the pool, for a definite and reliable detection, either a large sample of coolant has to be collected from the pool or multiple samples are to be collected from different locations in the pools. Since the very large volume of sample being brought away from the pool will have high gamma activity, multiple samples to be collected from the pool are preferred.

Sampling of the sodium is done with the help of four e.m. pumps. The d.n. blocks housing the boron coated counters are located on the roof tap as shown in Fig 2. The system is designed to monitor a rupture of around 2 cm^2 recoil area.

Shutdown of the plant is actuated with the failure of a fuel pin. But no separate failed fuel detection systems are provided for the two shutdown systems.

6.2 Localization of Failed Fuel Subassembly

As cover gas analysis by gamma detection gives a delayed signal and is only an indirect coolant sampling for d.n. activity measurement is preferred. This can be done by a selector valve which sequentially chooses the samples from the heads of subassemblies. A single

DND Localisation Block (3 Nos.)

IHX

IHX

DND(Bulk) Block
[4 Nos.]

LRP

SRP

PUMP

PUMP

CSRDM (9 Nos.)
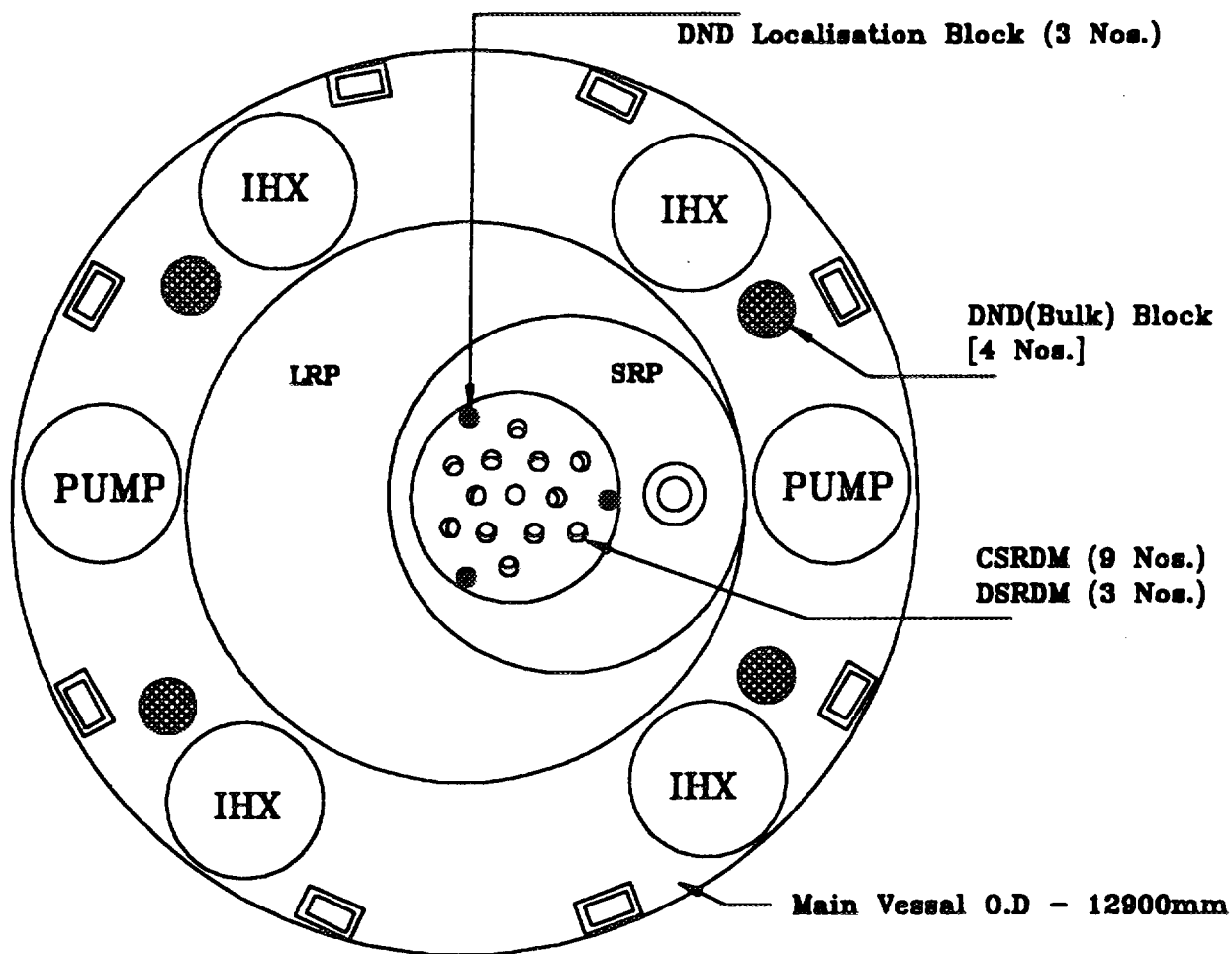DSRDM (3 Nos.)

IHX

IHX

Main Vessel O.D — 12900mm

FIG. 2. Location of DND blocks on top of reactor.

selector valve results in a longer cycle time for the 198 samples and also is considered mechanically unwieldy.

Thus, three selector valves drawing sodium samples simultaneously at 0.05 l/s from one of the 66 subassembly heads with the help of DC conduction pumps helps in localization of fuel failure (Fig. 3). A 200 cc sodium capacity gives around 25 nv of flux enabling detection of a 1 cm^2 rupture. The scanning rate for a full cycle is around 24 minutes.

6.3 Gaseous Fission Product Detection

The gaseous fission product detection system consists of on-line gas flow ion chamber and a gas chromatography along with a Ge-Li detector for spectral analysis. The cover gas sample is drawn with a 15 mm pipe at a flow rate of 0.1 l/s and the response time is of the order of 15 min. Variation in the current measured by this dc channel gives an early indication of gaseous rupture. analysis of this signal by spectral analysis helps identification or characterization of the fuel failure.

7. FLOW MEASUREMENT

Measurement of sodium flow through the core is required to find out the adequacy of· heat removal, flow blockages and heat balance. Flow measurement is done either at the outlets
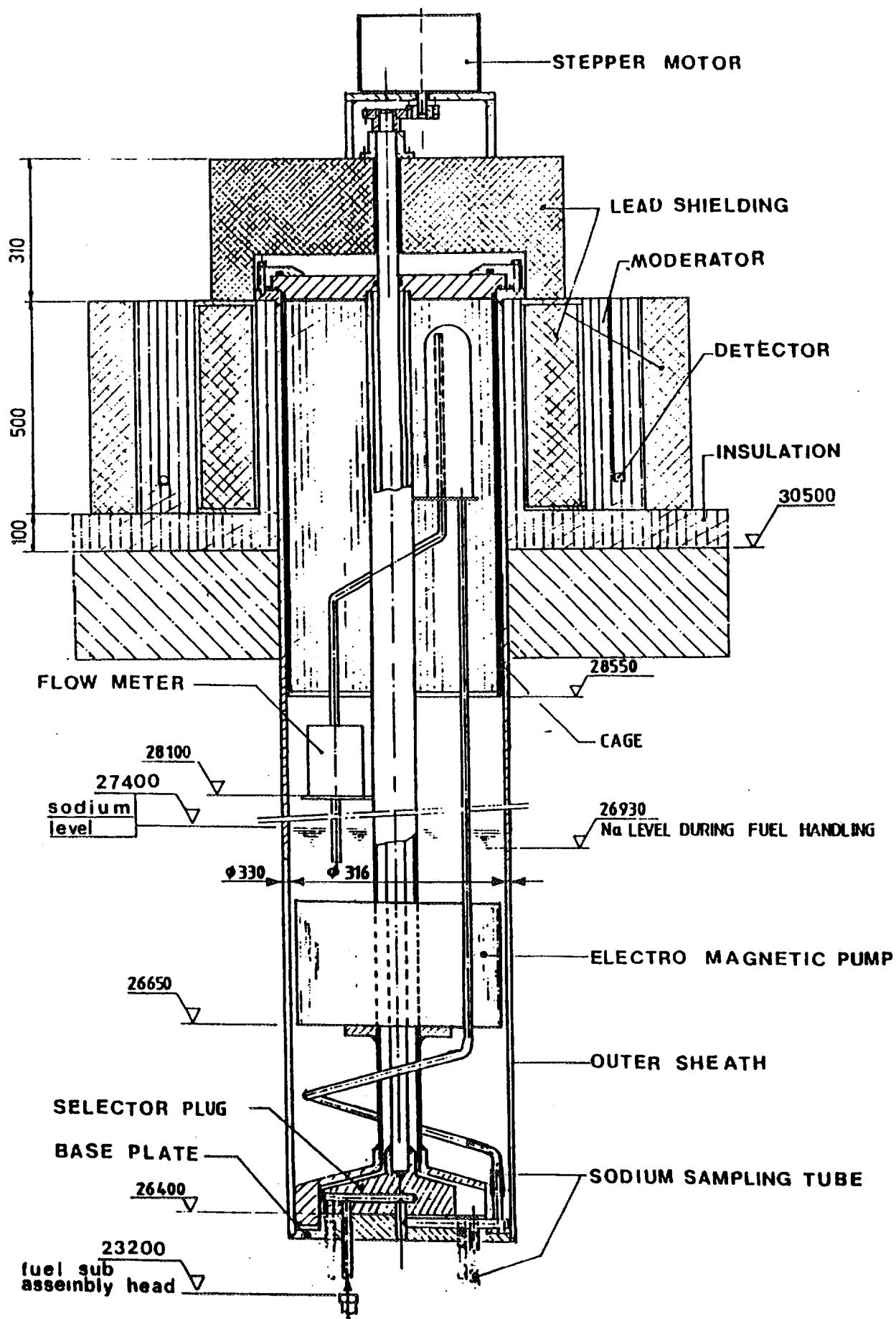
FIG. 3. Failed fuel localization module.

STEPPER MOTOR

LEAD SHIELDING

MODERATOR

DETECTOR

INSULATION

30500

28550

CAGE

FLOW METER

28100

27400

sodium
level

26930
Na LEVEL DURING FUEL HANDLING

⌀330   ⌀316

ELECTRO MAGNETIC PUMP

26650

OUTER SHEATH

SELECTOR PLUG

BASE PLATE

SODIUM SAMPLING TUBE

26400

23200
fuel sub
assembly head

310

500

100

of all subassemblies or a the inlets to the core. miniature eddy current flow meters which are fast in response when located at subassembly heads help in detecting flow blockages.

Flow measurement at the inlet to the core can be done by a direct measurement of providing magnetic flow meters at the discharge of the primary pump or an indirect method of measuring the pump speed. Flow meters placed in the interconnecting pipe between pump discharge and the dia grid are not replaceable. Thus, bypass type PM flow meters are proposed which are integral to the primary sodium pump assembly and hence replaceable.

However, miniature eddy current flow meters are also proposed to be located at the outlets of four subassemblies which could be representative of the bulk flow in that quadrant.

Inadequate flow to the core is taken as a SCRAM parameter.

8. SODIUM LEAK

As leakage of sodium can result in fires and sodium-concrete reactions, double walled pipes are provided for all primary and secondary sodium main pipes. The nitrogen in the annular spaces of these pipes are sampled continuously to monitor a leak by the Sodium Aerosol Detector (SAD). As a diverse measurement spark plugs are provided in the double envelopes but modified to avoid bottom installation. The reactor vault and space between safety vessel and main vessel are also monitored using the above two diverse monitoring systems.

9. STEAM GENERATOR LEAK DETECTION

SG leak detection system monitors water leak into sodium, as this leak can develop from small crack to a tube rupture. Small leaks are monitored using mass spectrometer based systems in which hydrogen generated by the sodium water reaction diffuses through the nickel membrane into vacuum. Four steam generator modules in each loop are provided with one leak detection system by individual sodium sampling and one leak detection system is provided on the common discharge header. Hydrogen in Argon Detector (HAD) is provided in argon space of pump tank for leak detection during low temperature operating conditions.

Pressure rise in pump tank is indicative of intermediate leaks. Triplicated spark plug type leak detectors on 2/3 logic initiate all SG safety related actions to contain effects of large leak.

10. PROTECTION LOGIC

The various parameters that enter the SCRAM system which consists of dropping the absorber rods into the core are listed in Table 1. The parameters are identical for both the shutdown systems. The various logic systems that were considered and their features are tabulated in Table 2. To achieve diversity, the logic of the two SCRAM systems are proposed to be on different principles.

Solid state static logic are not fully failsafe. This drawback is generally overcome by providing on-line test circuits.

TABLE 1. INPUT PARAMETERS FOR SCRAM

1. High neutron flux in the start up range   LOG N

2. High rate of flux in start up range (period)   TN

3. High power - linear in power range   LIN P

4. High power - log in power range   LOG P

5. High rate of power change in power range (period)   TP

6. Reactivity   RHO

7. High mean core outlet temperature HT_m

8. High individual S/A outlet temperature HT_i

9. Difference in mean core outlet temperature &
   inlet temperature DT_m

10. Power to flow ratio   P/Q

11. Fuel clad failure by DND   DND

TABLE 2. COMPARISON OF LOGICS

	Relay	Solid state discrete compo- nents	ICs	Compu - ters	PCL	Magnetic core (Laddic) devices
Reliability	High	Very Good*	Very Good*	Very Good**	Very Good**	Very Good
Testability	No	Yes	Yes	Yes	Yes	Yes
Failsafeness	Yes	No	No	No	Yes	Yes
Flexibility	No	No	No	Yes	No	No
Response	Slow	Medium	Fast	Fast	Fast	Medium
Ruggedness	High	Medium	Medium	Low	Low	Medium
Power Consumption	High	Medium	Low	Low	Low	Medium
Used in	Earlier generation of reactors	FBTR Rapsodie	Dhruva	PWRs in France, FFTF, USA	Super Phenix	Super Phenix, Gas cooled reactors in U.K.

*   depends on the components chosen
**   depends on the software reliability

TABLE 3. DESIGN BASIS EVENTS & SCRAM PARAMETERS

Trip Parameters		Design Basis Events			
		1	2	3	4
High neutron power	LOG P	*			
High neutron power	LIN P	*			
High rate of power change	TP	*			
Reactivity	RHO	*			
High mean core outlet temperature	HT_m	*	*	*	
High individual S/A outlet temperature	HT_i				*
Difference in mean core outlet temperature and inlet temperature	DT_m	*	*	*	
DND(Bulk)	DND				*
Power to flow ratio	P/Q			*	

1. Inadvertent withdrawal of one absorber rod
2. Failure of one primary sodium pump
3. Failure of one secondary sodium pump
4. Subassembly fault / pin failure / flow blockage

Development of a dynamic or tri-state logic dispenses with the need for on-line testing as well as offers fail-safeness. Thus, the logic for one of the shutdown system is to be built on this dynamic pulse-coded-logic.

The choice of the logic for the second system could be either microprocessor based or computer based as they offer flexibility. However, the reliability of the software and their acceptance to the safety authorities is yet to be established. In view of this, despite the lack of flexibility and the robustness the second logic system may be built with proven and rugged relay logic.

The overall configuration of the safety instrumentation that causes SCRAM is shown in Fig. 4.

Due to the adoption of pulse coded logic for one of the systems, the need for on-line testing of the SCRAM logic is not there. Due to the low shutdown flux in fast reactors, on-line testability of the neutronic channel from the detector stage is difficult. However, due to the adoption of a 2 out of 3 majority logic individual channel can be tested on-line.

To achieve the required reliability, the two shutdown systems may be optically coupled and the last stage, the SCRAM is initiated from any of the first shutdown system parameters actuates the second shutdown system also and vice-versa.

11. DIVERSITY IN SCRAM PARAMETERS

An attempt has been made to analyze SCRAM parameters that cause shutdown for some of the incidents. This is summarized in Table 3. It is seen that for each of the events, more than one parameter actuates SCRAM.
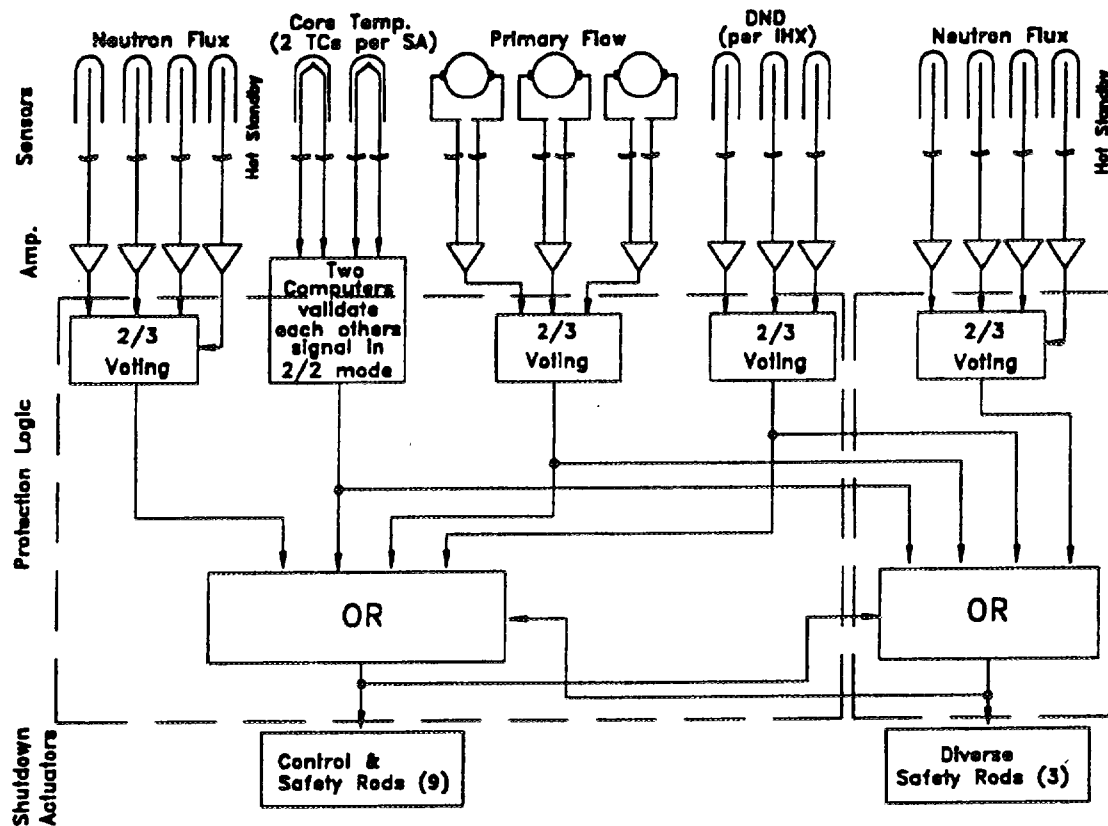
FIG. 4. Safety instrumentation - Typical configuration.

12. CONCLUSION

Safety instrumentation causes the reactor shutdown in the fastest manner. For this the incidents which are considered probable are listed, the instrumentation that is needed for monitoring these is provided. Ultimately an analysis of the adequacy of this instrumentation with respect to the probable incidents is done. As evolution of electronics is continuing, the actual hardware to be provided cannot be frozen at an early stage and this can only be done at the detailed design stage.

SAFETY CONSIDERATIONS IN THE DESIGN OF PFBR

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Abstract

Prototype Fast Breeder Reactor (PFBR) is a 500 MWe reactor under design in India. The overall safety approach adopted is based on the defence-in-depth principle. Design features have been incorporated to minimise occurrence of unsafe conditions. A plant protection system comprising reliable core monitoring to detect the off-normal condition, a reliable shutdown system to ensure safe shutdown and a passive decay heat removal system are provided. Containment is provided to prevent any release of radioactivity to the environment in case of failure of the protective devices. This paper provides a brief outline of the safety considerations in the design of PFBR.

1. INTRODUCTION

Prototype Fast Breeder Reactor (PFBR) is a 1250 MWt, 500 MWe mixed oxide fuelled reactor under design in India. The overall safety approach adopted for PFBR is based on the defence in depth principle which incorporates safety provisions in three stages. The first step is to provide inherent safety features in the design of the core and components in the various systems of the reactor. The second step is to incorporate a diverse and redundant protection system which will ensure safety actions when an off-normal event is detected by monitoring various parameters like neutron flux, coolant flow, temperature and leak to prevent any plant damage. Finally as a third step, adequate containment is provided to prevent any leak of radioactivity to the environment in case the protective devices provided as above fail, resulting in a core power excursion. This paper provides a brief outline of the various safety features in the design of PFBR.

2. GENERAL SAFETY APPROACH

The following principles govern the general safety approach adopted for PFBR:

- Use principles of redundancy and diversity besides segregation/independence of components wherever necessary to achieve overall high reliability.

- Incorporate fail safe design features wherever possible.

- Minimise the occurrence of the incidents through sound design, construction and inspection.

- Take into account the operating experience of different LMFBR especially with respect to sodium fire and sodium water reaction.

The components of NSSS have been classified into four groups according to their influences on radiological safety and these safety classes have been linked to four different classes of design, construction and inspection.

The design basis initiating events have been identified and divided into four categories based on the frequency of their occurrence i.e. normal, upset, emergency and faulted. For each of these categories, design limits have been proposed for fuel, cladding and plant components. Single failure is considered in the analysis of incidents and credit for manual action is taken half-an-hour after the occurrence of the event.

3. FEED-BACK REACTIVITY COEFFICIENTS

In PFBR, both temperature coefficient (2.75 pcm/deg C) and power coefficient (0.8 pcm/MW) of reactivity are negative so that any increase in temperature or power leads to a reduction in reactivity and the consequent reduction in power. The negative reactivity comes from the expansion of fuel, radial expansion of core and grid plate, control rod expansions and from the Doppler effect of the fuel. The expansion of coolant sodium and structural steel result in small positive reactivities which are compensated by the former negative reactivity effects so that the net reactivity coefficient is negative. The net negative reactivity coefficients of temperature and power, thus provide the intrinsic safety.

4. CORE MONITORING

The objective of core monitoring is to ensure that reactor power and adequate cooling of all subassemblies is maintained within their permissible limits during reactor operation. A companion paper to this meeting [1] deals with the details of core monitoring. It is essential to detect a fuel pin failure at the earliest to prevent large scale propagation and minimise contamination of the primary sodium. For this purpose, different parameters both neutronic and thermal are monitored and actions initiated in case of detection of off-normal conditions. The safety actions are either lowering of all absorber rods (LOR) or drop of all rods (SCRAM). Table I gives a list of parameters that order reactor trip. For detection of failed fuel, the bulk detection is based on assessing the presence of delayed neutrons in the pool sodium drawn from behind the IHX. Localisation is based on sampling sodium from individual fuel subassemblies using selector valves and checking for presence of delayed neutron. Temperature monitoring of all the fuel subassemblies is done and processed by computer for ordering any trip action.

5. SUBASSEMBLY BLOCKAGE

To prevent total instantaneous blockage (TIB) at inlet of the subassembly multiple openings are provided in the support sleeves and subassembly feet. To minimise effect of total blockage at outlet of the subassembly, an adaptor at the top of the subassembly has been conceived. This adaptor provides a

TABLE I. PARAMETERS CALLING FOR SAFETY ACTIONS

No.	PARAMETER	SAFETY ACTION	
		LOR	SCRAM
1.	High neutron flux in start-up range	-	yes
2.	Short period in start-up range	-	yes
3.	High neutronic power (LOG P)	-	yes
4.	High neutronic power (LIN P)	-	yes
5.	Short period in power range	-	yes
6.	High reactivity in core	-	yes
7.	High individual S/A outlet temp.	yes	yes
8.	High mean core outlet temp.	yes	yes
9.	Difference in mean core outlet temp. & inlet temp.	yes	yes
10.	Fuel pin failure by DND	-	yes
11.	Power/primary flow rate	yes	yes
12.	Primary pump trip/reduction in flow	yes	-
13.	Secondary pump trip	yes	-
14.	Feed water pump trip	yes	-
15.	Turbine trip	yes	-
16.	Water/steam leak into sodium in SG	yes	-

radial flow path when blockage occurs. The flow during blocked condition is approximately 30 % which is sufficient to avoid sodium boiling. Studies have shown that instantaneous blockages resulting in flow reduction of 32 % through the fuel subassemblies can be detected and reactor tripped to limit clad hotspot to 800 deg C, with 4 s time constant thermocouples. With limiting condition on sodium boiling, 71 % step flow reduction can be detected. In case of zero time constant thermocouples the safe detectable flow reduction value is 37.5 % for clad hotspot limit and 77.5 % for sodium boiling limit. Thus there is no advantage to have a fast response thermocouple for this event.

6. SHUTDOWN SYSTEM

Reactor safety is assured by two independent, fast acting, diverse shutdown systems each comprising sensors, logic circuits, drive mechanisms and absorber rods. First system comprises of a bank of 9 control and safety rods (CSR) while second system has 3 diverse safety rods (DSR). CSR is for reactivity and power control as well as shutdown while DSR is only for shutdown. The scram release electromagnet of CSR drive mechanism is housed in the upper part in argon atmosphere while for DSR it is at a lower end of drive mechanism which is immersed in sodium. For DSR there is a curie point magnetic switch which gets demagnetised on reaching a temperature of 873 K under LOF or TOP and triggers drop of the rods. In view of two shutdown systems with a high reliability and passive feature like curie point magnet, shutdown is assured for all design basis events.

7. DECAY HEAT REMOVAL

If offsite power is available, decay heat is removed through the normal path through steam generator and steam water system.

In order to have a high reliability of decay heat removal (DHR), the DHR system must be closest to the heat source i.e. core as this will involve minimum components between the heat source and the sink. With this in view 4 DHR circuits each comprising a dip heat exchanger (DHX) is provided in the hot pool of the reactor. Each DHX is linked on the tube side to a intermediate sodium circuit and ends in a sodium-air heat exchanger (AHX). During loss of off-site power, the sodium flow in the primary and intermediate circuits and air flow over AHX is by natural convection. The layout and relative elevations have been fixed to ensure this passive feature. As a defence in depth measure, battery power to drive the primary pumps at 20 % of the speed for about 1 h is also provided. It is sufficient if 2 out of the 4 DHXs operate to assure fuel and clad integrity. The degree of diversity to be provided amongst the DHR circuits to take care of common cause and common mode failures is being studied.

8. SODIUM FIRES

Sodium fire is one of the important areas in safety which needs to be tackled right from the design stage. Some of the design features considered to prevent sodium fires are:

a) Material selection with sufficient ductility to assure leak before break and eliminate the possibility of sudden development of a large leak and sodium spray fire.

b) Adequate inspection of all main pipe welds.

c) Provision for dumping sodium wherever possible.

d) Provision for leak detection at important weld joints.

e) All sodium pipelines inside containment building have been provided with double walled construction.

f) Provision of leak collection trays.

g) Development of high temperature concrete and liners.

In addition to the above, following measures will be incorporated into the detailed design.

-   Providing partition walls to limit the effect of fire on the other equipment.

-   Provision of proper vents in the different areas to relieve pressure developed due to sodium fire.

9. SODIUM-WATER REACTION IN SG

The possibility of sodium water reaction in the steam generator cannot be ruled out. With a good sensitivity of the hydrogen leak detection system in secondary sodium, timely detection is possible. There is, however, a finite probability that an undetected leak may cause wastage of more than one tube. The under-sodium leak event in PFR superheater in 1987 has

indicated that overheating of tubes can result in multiple tube failures. For PFBR, the design basis leak (DBL) which was double ended guillotine (DEG) rupture of 1 tube in the initial design has been modified to a value which will cause the rupture of both top and bottom rupture discs on the sodium side. For the present SG concept without cover gas, it is seen that DBL is 5 DEG failure for a bottom leak or 3 DEG failure for a top leak.

10. SODIUM VOID REACTIVITY EFFECT

Sodium voiding can occur due to the temporary passage of argon gas or fission gas from pins, through the core. It may also occur due to boiling of sodium in the case of blockage in the subassembly or loss of flow due to trip or seizure of both primary pumps and transient over power event. Loss of coolant which also may lead to voiding is not considered in view of provision of a safety vessel around the main reactor vessel, the gap between the two vessels being monitored. In all the cases, protective action of reactor scram and consequent decay heat removal will take place and no voiding of core is expected. Nevertheless, the incidents of loss of flow (LOFA) and transient over power (TOPA) without reactor trip have been studied to assess the propagation and energy release expected and to estimate containment requirements to limit the radioactivity release.

The total sodium void reactivity effect of PFBR is 1200 pcm (3.5 $). The void worth of the central assembly is 0.066 $. Void worth of central and surrounding 6 subassemblies is 0.4 $. In order to have reactivity addition of 1 $ it is necessary for voiding to take place in 18 fuel subassemblies.

The time dependent reactivity effects due to fuel axial expansion and structural expansion of grid plate, control rod etc. have been neglected in the analysis as a conservative measure. For LOFA, two studies have been conducted with 2 s and 10 s flow halving times. The former refers to the seizure while the later refers to the trip of the primary pumps due to loss of off-site power.

For the 2 s flow halving time, the time taken to reach fuel melting is 16 s for the reference core. When the upper axial blanket in the core is replaced by sodium, the total void effect is 706 pcm (2.06 $) and time taken to reach fuel melting is 720 s. For the 10 s flow halving time, fuel melting starts at 79 s for reference design and for the case of sodium in place of axial blanket, no fuel slumping occurs upto 2500 s.

For other accidents like TOPA, reduction in void coefficient has not much significance. It must be emphasised that unprotected events are of a very low probability due to provision of diverse shutdown systems. Removal of upper axial blanket is not favoured from consideration of loss of breeding. Other methods of reducing sodium void effect without affecting breeding like axial heterogenous core are still being studied.

11. PUMP TO DIAGRID PIPE CONNECTION

Each primary pump is linked to the diagrid by two pipes of 600 mm dia. With the use of stainless steel having high ductility, leak before break concept can be applied to this pipe. High quality of manufacturing will be maintained for this component. Detailed analysis [2] has been carried out with conservative estimate on the loadings. It has indicated that for the initial crack size of a/t (crack depth/thickness) 0.3, and 2c/t (crack length/thickness) 10, leakage less than 1 kg/s is possible after many plant lifes. There exists large safety margin between the critical crack length and the maximum crack length attained at the time of leakage. This indicates that the LBB argument can be applied to this critical component.

Nevertheless, the guillotine rupture of one of the pipes can be managed by very fast detection through reactivity, primary coolant flow measurements and high sensitive quick response (approximately 1 s time constant) core outlet thermocouples for some selected fuel subassemblies. It is also planned to have core flow monitored by 4 eddy current flowmeters in the control plug and these would provide one more signal for core flow decrease.

12. CORE CATCHER

Whole core meltdown is considered a residual risk event in the state-of-art design of LMFBR. However, local accidents like gross blockage are considered credible and safeguards against such accidents are essential. It is very difficult to quantify the magnitude and rate of blockage. The studies carried out in France regarding such an accident lead to a conclusion that the reactor can be safely shutdown before the accident progresses beyond the adjacent ring of 6 subassemblies. A provision for post accident cooling of fuel debris from 7 subassemblies is thus considered adequate [3, 4]. With the fuel melt down from 7 subassemblies, there is no fear of criticality as the K effective is only 0.87. For criticality to occur, calculations have shown that about 25 subassemblies need to melt down. A core catcher located below the core support structure covering the area of entire grid plate has been provided to safely contain the molten debris and cool them by natural convection.

13. REACTOR CONTAINMENT

With the various design provisions, core meltdown can be considered as an event of extremely low probability and can be classified as beyond the design basis. Yet this has been considered in determining the containment capability of the system. Detailed analysis of LOFA with seizure of both primary pumps gives a mechanical energy release of 23 MJ. For TOPA represented by hypothetical continuous addition of 4 pcm/s, the total energy release comes to 135 MJ. Hence the energy release for containment has been taken as 200 MJ as a conservative measure. It has also been checked that the energy release of this magnitude can be absorbed without failure of the main vessel and the roof slab [5]. Though no sodium is envisaged to be released into the containment building during such an accident, it has

been assumed that about 500 kg of sodium may be ejected into the containment building. This ejected sodium can catch fire and give rise to an overpressure of 25 KPa.

The containment building has been designed to withstand such pressure build-up and limit the release of the products to less than 0.1 % of the building volume per hour. The resulting dose due to this release at the site boundary of 1.6 km from the reactor building is found to be within the permissible levels.

14. SITE RELATED EVENTS

Following site related external events are considered in the design:

- Aircraft impact: considering the distance from airport and frequency of flights, a fall of commercial aircraft is a very very low probability event, hence not a DBE.

- Flood: There is no flowing river in the vicinity of the site and hence no risk of flooding due to river water.

- Cyclone: Every year 4 to 5 cyclones visit the coast of Kalpakkam. Under extremely severe cyclonic conditions, the maximum tidal wave may inundate portion of land about 5 m above mean sea level. It is therefore proposed to protect building complex by means of peripheral roads with an elevation > 5 m above mean sea level. The maximum possible wind speed which must be considered in the design is 280 km/h at 10 m level.

- Chemical explosion: No chemical industry exists nor any transportation of explosives takes place near the site and hence chemical explosion is not considered in design.

- Seismicity: The seismicity values of the site are as follows:

  0.078g(S1) and 0.156g(S2) for peak horizontal ground acceleration.

  The peak acceleration in vertical direction is 2/3 of the peak acceleration in the horizontal direction.

15. SUMMARY

This paper has brought out the important safety approaches adopted in the design of PFBR. It is seen that with the application of the defence in depth principle, the reactor can be adequately designed to cater to incidental and accidental conditions. The operating experience of the different LMFBRs have been used in the design and specific design features for protection against sodium fires and sodium water reaction incorporated. Core melt down which is a very low probability event has been considered and a core catcher and containment have been provided. In summary, considerable progress has been made in the safety case for PFBR.

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DESIGN OF THE ADVANCED REACTOR BN-600M

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Abstract

This paper summarizes the design and performance characteristics at the BN-600M nuclear power plant as developed at this stage of the advanced conceptual design. The approaches being taken in the design in order to meet modern safety standards also outlines.

1. INTRODUCTION

The concept of nuclear power development in the Russian Federation was approved by the Minatom Board on July 14, 1992 [1]. This concept determines the general goal of the programme, its main tasks within the determined time period, the main stages of programme implementation, primary NPP projects, considers conditions of the NPP fuel supply and proposes various options of nuclear power development including fast reactors.

The analysis of the design and operating experience, the assessment of the state of research and test-design work and the possibilities of improving the economic and safety characteristics and the prediction of prospects for the commercial use of fast reactors in the Russian Federation could be summarized as follows:

- the structural-layout and circuit designs worked out for the equipment and systems and verified by operating experience demonstrated the maturity of the technology of sodium-cooled fast reactors. In regard to operating indicators the BN-600 has reached the level of commercial reactors;
- the design and licensing of the BN-800 project have been completed and the conditions have been created for building the fourth power unit of the Beloyarsk Plant and three units of the South Ural Plant with BN-800 reactors;
- the large-scale commercial introduction and creation of a two-component nuclear power system on the basis of fast and thermal reactors are linked today with the development of the next-generation intermediate power BN-600M reactor. The project for the improved BN-600M ensures the economic competitiveness of fast reactors with light-water reactors.

2. DESIGN OVERVIEW

The BN-600 is of middle size advanced nuclear power concept that combines application of high degree proven design solutions from: previous LMFR projects BN-600 and BN-800, and new design approaches. Key design features which have been validated by the experience of BN-600 construction and operation are:

- pool type-reactor design;
- bottom support of the reactor vessel;
- use of main equipment: PSPs, SSPs, IHXs, in-vessel handling system, CRDMs and fuel subassemblies from BN-600 reactor.
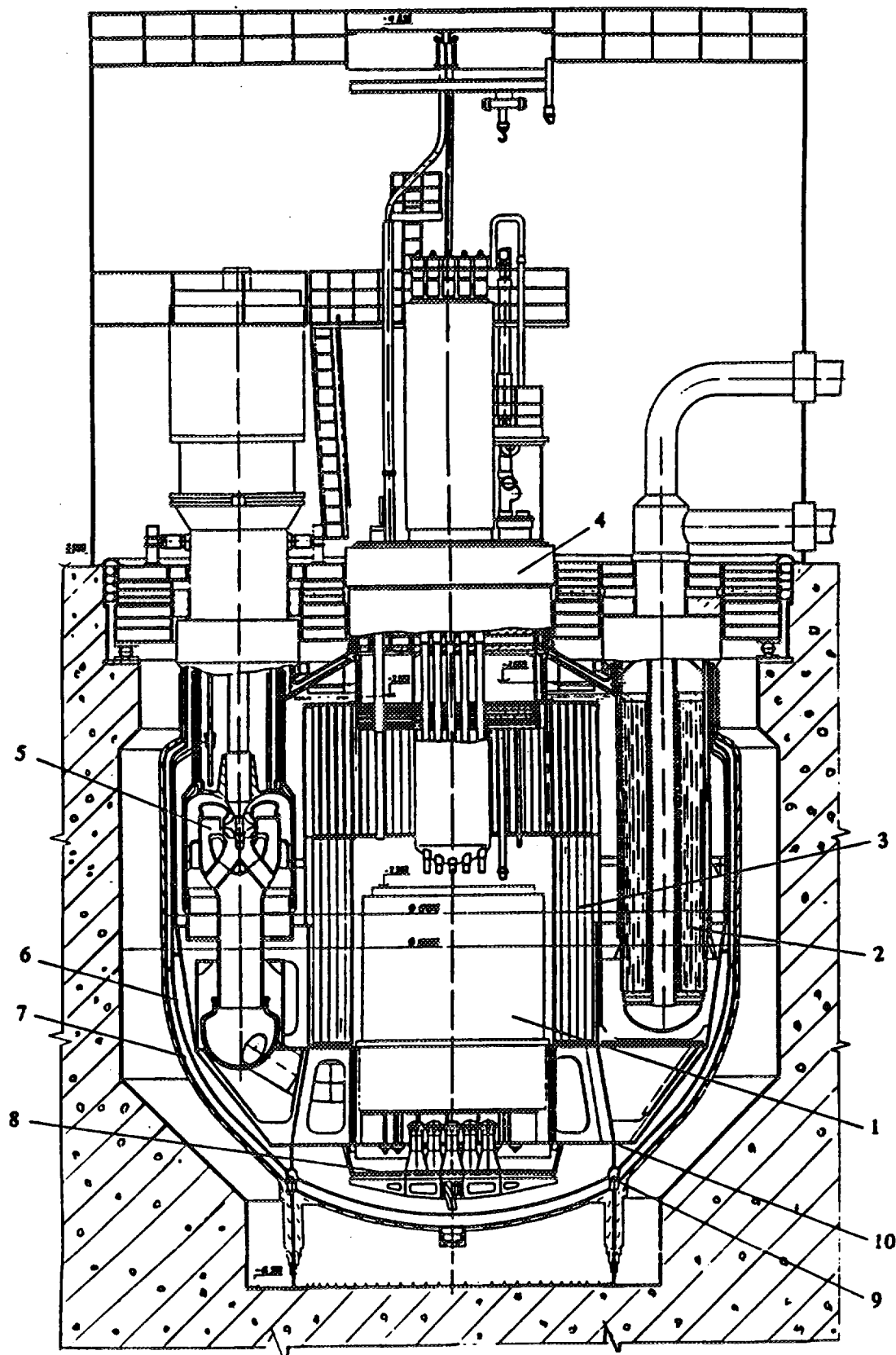
Fig. 1. General view of BN-600M Reactor

1 - core; 2 - intermediate heat exchanger; 3 - in-vessel shielding; 4 - rotating plug; 5 - main primary pump; 6 - main reactor vessel; 7 - reactor guard vessel; 8 - core catcher; 9 - reactor vessel support; 10 - in-vessel support structure

168

New design approaches are also here listed as follow:

- completely integral primary radioactive circuit due to cold trap location inside of the reactor vessel;
- two loops primary cooling system: two PSP and four IHX;
- core design with zero or negative sodium void reactivity coefficient;
- bellows for compensation of thermal extension of secondary piping system.

Figures 1 and 2 show a cross-section at the BN-600M reactor.

A brief description of reactor features is given below. The core has a negative sodium void reactivity effect due to the introduction of sodium plenum and boron carbide shielding instead of upper axial blanket. An additional group of absorber rods is used for reactor
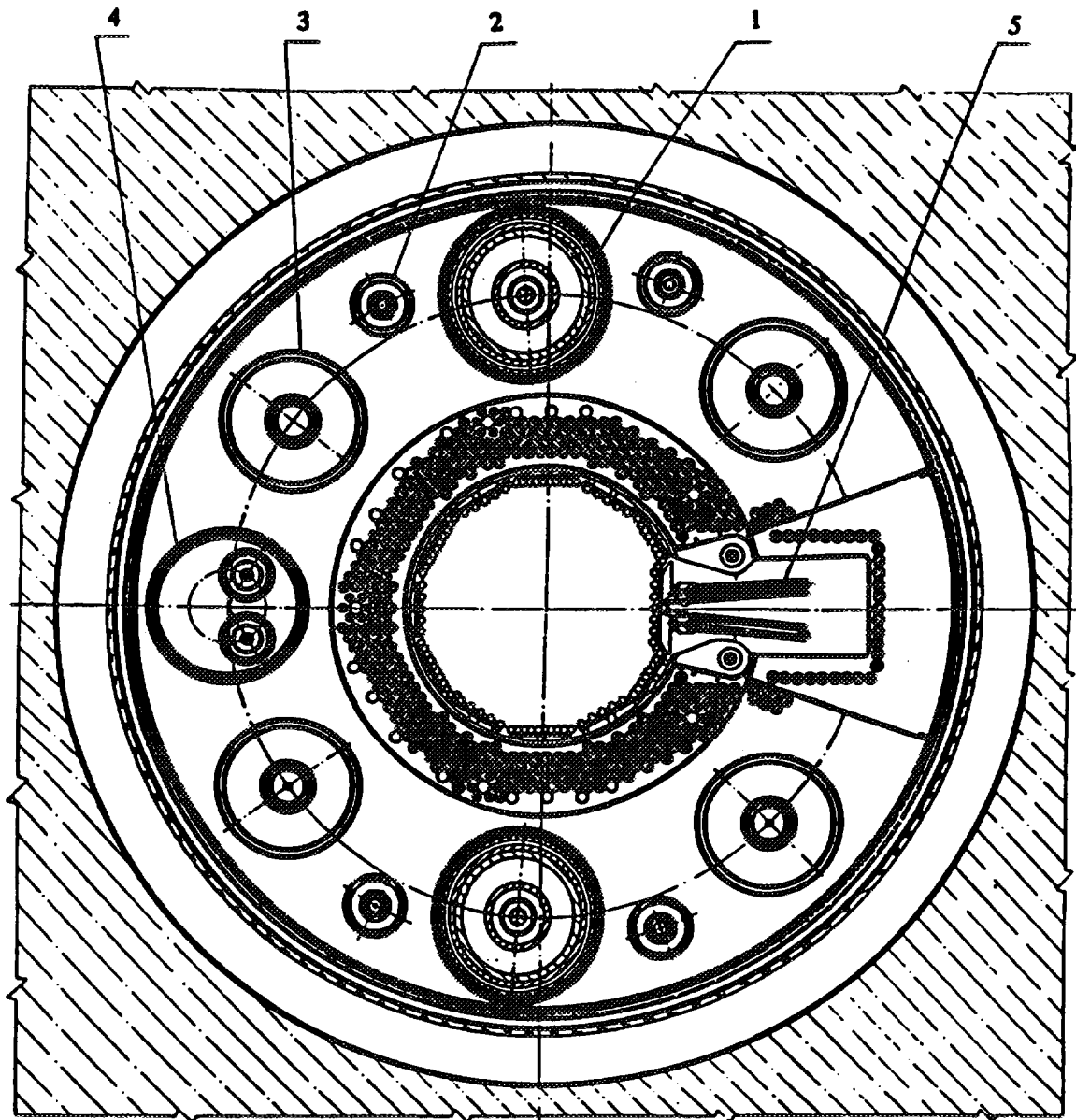
Fig. 2. Cross sectional view of BN-600M reactor
   1 - primary circuit pump; 2 - DHRS heat exchanger; 3 - intermediate heat exchanger; 4 - cold trap; 5 - fuel assembly loading-reloading elevators

shutdown. They act according to the passive principles and drop into the core when the coolant flowrate decreases to 50% of nominal value. A catcher is arranged under the core to contain components if the core would be destructed during beyond design accidents. The catcher was adopted to hold core fuel, melted steel and uranium dioxide of the lower axial blanket. In vessel shielding includes: in radial direction removable assemblies of boron carbide and steel, neutron reflector of steel, then several rows of tubes filled with boron carbide. Using in the shielding assemblies of boron carbide allowed to reduce heat generation in fuel subassemblies (FAs) of the in-vessel store and to cool them by natural convection of sodium from inter-assemblies space. The assemblies of boron carbide are cooled in the same manner.

Three rotating plugs are used to ensure FAs handling. This provides a symmetrical arrangement of the above structure system (ASS) relative to the core. Therefore, coolant streams around the ASS will be symmetrical as well. The first loading of the core is supposed to use MOX fuel. Main characteristics and core layout are given in Table 1 and Fig. 3 respectively. Besides the above-mentioned features, a number of new engineering decisions described below, are used in BN-600M design.

TABLE I. MAIN CHARACTERISTICS OF THE CORE

Characteristics	Value	
Core height, mm	880	
Number of FAs:		
- core	469	
- radial blanket	84	
Number of shield subassemblies:		
- boron carbide	170	
- steel	226	
Number of absorber rods:		
- active	27	
- passive	3	
Number of positions in in-vessel store	252	
Size of the FA duct, mm	96x2	
Number of pins in the FA	127	
Size of the fuel pin, mm	6,9x0.55	
	1 stage	2 stage
Maximum burnup, % ha.	12,7	15,8
Refuelling interval and residence	4x150=	5x150
time, eff	600	750
Maximum linear heat rating, kW/m	45,0	46,3

The primary cooling system has two loops, because only this arrangement can locate heat exchangers of the Direct Heat Removal System (DHRS) and cold traps inside of the reactor vessel. When heat removal through steam generators is lost, residual heat is removed through DHRS consisting of four independent channels, Each channel includes a sodium - sodium heat exchanger built-in the primary circuit, an air heat exchanger, pipelines, an air damper and a vent stack. Each channel capacity is 15MWth. It allows to remove heat by two
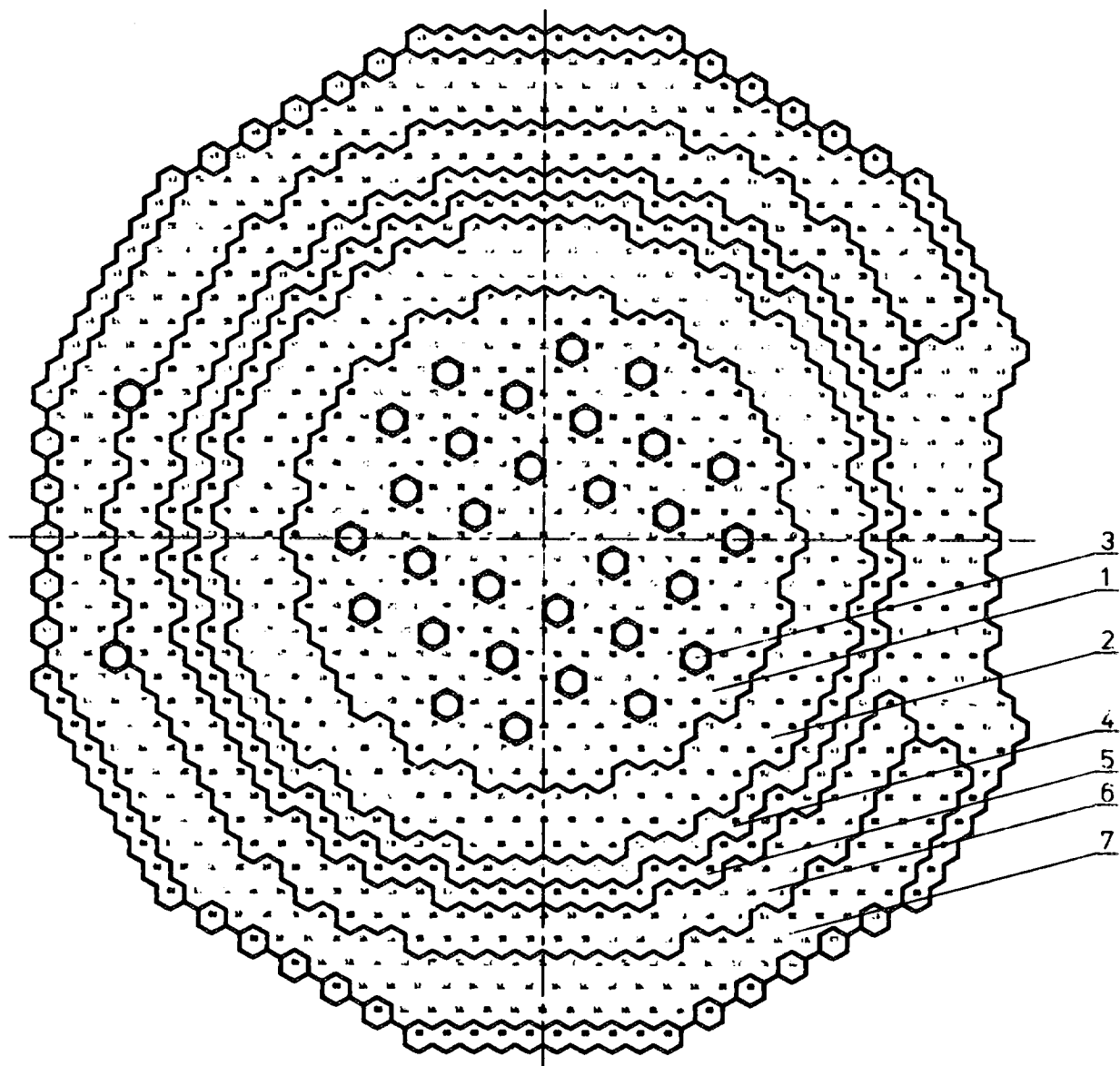
Fig. 3. Core layout of BN-600M reactor

1 - inner core zone; 2 - outer core zone; 3 - absorber assemblies; 4 - radial blanket; 5 - stainless steel shield assemblies; 6 - carbide boron shield assemblies; 7 - in-vessel storage positions

channels without loss of integrity of fuel pins over design basis limits. Cold traps for sodium purification are arranged in the reactor vessel so as to exclude branched pipeline system with radioactive sodium beyond the reactor tank and reduce leak probability. At the inlet of primary pumps there is a common chamber which was introduced in terms of safety. Indeed, such arrangement eliminates pressure increase under the strongback in case of a feeding pipe rupture. Neutron flux monitoring chambers are located in the reactor vessel. This improves control reliability, excludes hollow tanks in the vessel and served to increase neutron flux on chambers for their ex-vessel shield design. Number of in-vessel storage positions is increased to store FAs during two refuelling intervals. Together with additional shielding, it allows to decrease decay heat in FAs before refuelling to 15W and transport them to the cleaning facility without storage in ex-reactor store which is expelled from equipment of the handling system.

3. MAIN FEATURES OF THE NPP

The NPP with BN-600M reactor is designed as monoblock of reactor-turbogenerator. To provide seismic resistance of the nuclear island, the reactor building together with auxiliary systems rested entirely on a single foundation. The reactor with systems important for safety are preserved from external events by the containment (Fig. 4.). The reactor is located in a cavity, the roof of which is an Upper Fixed Shield (UFS). There are several holes in the UFS for stub tubers to pass through the large rotating plug, pumps, IHXs, cold traps, in-vessels handling machine and other equipment. A two-liner of the concrete vault above the UFS goes into a air-tight shell. Inside the shell, CRDMS, electric motors of the
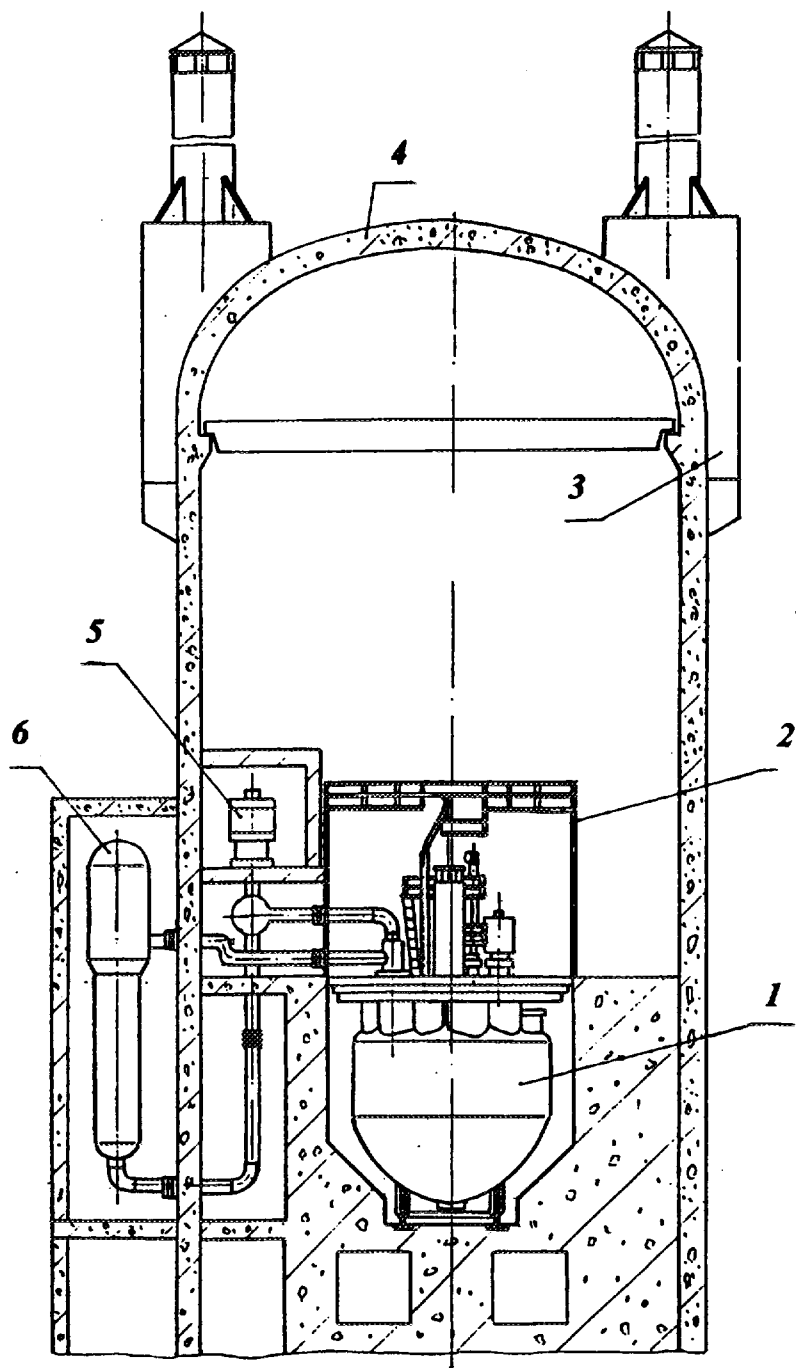
Fig. 4. Cut view of the reactor building

   1 - reactor vessel; 2 - air-tight shell; 3 - air-sodium exchanger; 4 - containment; 5 - secondary circulating pump; 6 - steam generator

main primary pumps and mechanisms of the in-vessel handling system are located. The air-tight shell serves as an additional barrier for retention of radioactivity. Sodium-air heat exchangers of the DHRS are arranged into separate boxes connected to the containment symmetrically relative to the reactor building. Mechanisms of ex-vessel handling system adjoins upper part of the reactor. The system provides transfer of fresh FAs to the ramp and irradiated FAs to the cleaning facility and for storage. Arrangement of the secondary circuit depends in a great extent on SG type. For module design of SG as used in BN-600, the secondary circuit includes two loops, each of them has ten modules. Each module consists of an evaporator and a superheater. For vessel-type design of SG, the secondary circuit includes four loops and each of them has: a SG, a pump, and IHX and pipelines. BN-600M technical data compared with BN-600 are presented in Table II.

TABLE II. REACTOR PLANT MAIN CHARACTERISTICS

Characteristics	BN-600M	BN-600
Reactor power, MWth	1520	1470
Power unit output, MWe:		
- gross	647	613
- net	595	564
Number of primary loops	2	3
Parameters of primary circuit:		
- hot leg sodium temperature, °C	550	550
- coolant flowrate, t/h	25590	24000
Parameters of secondary circuit:		
- hot leg sodium temperature °C	515	518
- cold leg sodium temperature , °C	345	328
- sodium flowrate, t/h	25300	21900
Parameters of water-steam:		
- feed water temperature, °C	240	240
- superheated steam temperature, °C	495	505
- steam pressure, MPa	13,7	13,7
- steam flow, t/h	2411	1980
Steam reheating	by steam	by sodium
Refuelling interval, effd	150	150-165
Refuelling outage, day	12	12
Design plant capacity factor	0,85	0,80
Thermal efficiency, gross %	42,5	40,7
Design lifetime, years	60	30
Seismic stability, magnitude on MSK scale,	8	7
Specific weight of reactor plant, t/MW	8,23*	13,0**

* - When using vessel-type SG   ** - When using module-type SG

4. SAFETY DESIGN APPROACH

Enhancement of the reactor plant safety is provided through the following design decisions. A sodium plenum and boron carbide shielding is introduced instead of the upper axial blanket, to exclude positive sodium void reactivity effect in case of sodium boiling.

Residual heat under accident conditions is removed through heat exchangers built-in directly into the reactor tank.

If absorber rods with drive mechanisms faile, the reactor is shutdown by additional group of hydraulically suspected rods.

Ionization chambers are placed into the reactor pool to improve neutron flux control.

Branched pipework beyond the reactor vessels is eliminated due to cold traps are located in the reactor pool. The reactor plant is seismic resistant up to magnitude 8 on MSK Scale-Safe Shutdown Earthquake. According to preliminary assessment the probability of an accident with severe core damage is less than 1 E-6 per reactor year.

5. BN600M ECONOMICS

An important goal of advanced fast reactor plants is to attain their competitiveness with other sources of electricity production, maintaining high safety level. This goal regards BN-600M is supposed to reach by the following procedures: (1) to use a maximum degree of design and engineering decisions adopted for previous reactors and proved by experience of their long operation period; (2) to reduce expenditure of materials for in vessel shield and FAs handling system by excluding a drum to store irradiated FAs and layout optimisation, and (3) to reduce the weight of equipment and systems of the primary circuit by cutting number of heat removal loops and housing cold traps in the reactor pool, and equipment of the secondary circuit by layout optimisation and usage of vessel-type SG. The specific weight of BN-600M for the alternatives with module-type and vessel-type SGs is reduced by 1.25 and 1.58 times respectively compared with BN-600.

Design value of maximum fuel burnup is increased up to 12.7% ha (1 phase) and 15.8% ha (2 phase) that is assumed to be realistic on the basis of data obtained for experimental FAs with MOX fuel.

The results of engineering and economic analyses showed that efficiency of the reactor unit is raised through lowering all components responsible for the high electricity generation costs:

- capital costs by reduction of specific weight of the reactor equipment;
- fuel cycle cost by fuel burnup increase;
- operating experiences by cutting down the plant staff due to automation of technological processes and reduction in number of operations for the reactor refuelling and equipment replacement resulting from improvement of its reliability.

6. FUTURE ACTIVITIES

During the next phase of the design development it is planned to continue in improving the reactor equipment and systems for reduction of specific weights and dimensions, namely:

- module-type SG as regards the number of modules, tube length and arrangement of tube bundles;
- arrangement of the secondary circuit pipework to shorten length of pipelines;
- outer handling zone without a handling and washing boxes which have large weight and occupy vast area;
- more light flasks for replacement of the primary circuit equipment.

174

When developing BN-600M detailed design, R&D will be conducted to confirm new decisions, namely:

- testing of equipment and DHRS as a whole using in water and sodium test facilities;
- experimental tests of large diameter bellows for secondary piping system;
- testing of cold trap models installed directly in primary circuit;
- experimental tests of vessel-type SG design;
- experimental investigations for perfection of reactor equipment (pumps, IHX, mechanisms of handling system, CRDM, etc.).

Further, reactor plant parameter revising is not excluded, especially power level in connection with the core concept variation that will be probably intended for plutonium consumption of minor actinides destruction.

7. CONCLUSION

Design work performed for BN-600M shows the possibility to create a reactor plant of middle power range meeting modern requirements on safety and being competitive to LWRs in the Russian Federation. Rate of further work and scope of R&D will depend on funding the design as a whole.

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INTELLIGENT TYPE SODIUM INSTRUMENTATIONS FOR LMFR

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Abstract

The constructions and their performances of a lot of newly developed intelligent type sodium instrumentations that consist of the intelligent type sodium flowmeter, the intelligent type immersed sodium flowmeter, the intelligent type sodium manometer and the intelligent type sodium level gauge are described. The graduation characteristic equations for corresponding transducer using the medium temperature as the parameter are given. Because the operating temperature limit of measured medium (sodium) is wide, so the on-line compensation of the temperature effect of their graduation characteristics must be considered. The tests show that these intelligent type sodium instrumentations possess of good linearity. The accurate sodium process parameter (flowrate, pressure and level) measurement data can be obtained by means of their on-line compensation function of the temperature effect. Moreover, these intelligent type sodium instrumentations possess of the self-inspection, the electric shutoff protection, the setting of full-scale, the setting of alarm limits (two upper limits and two lower limits alarms), the thermocouple breaking alarm, each other isolative the 0—10V direct-current analogue output and CENTRONICS standard digital output, and the alarm relay contact output. These intelligent type sodium instrumentations are suitable particularly for the instrument, control and protective systems of LMFR by means of these excellent functions based on microprocessor. The basic error of the intelligent type sodium flowmeter, immersed sodium flowmeter, sodium manometer and sodium level gauge is respectively ±2%, ±2.3%, ±0.3% and ±1.9% of measuring range.

1. INTRODUCTION

Because the liquid sodium as the coolant of LMFR possesses of such features as high temperature, high electroconductivity, non-transparent, violent sodium-water reaction, inflammability in air, and solid state

at ambient temperature etc., so the transducers measuring the process parameters (flowrate, pressure and level) of liquid sodium are different considerably from those transducers measuring the process parameters of water. Moreover, the operating temperature limit of measured medium (liquid sodium) in LMFR is wide, the on-line compensation of the temperature effect of the graduation characteristic of transducers must be considered. For these reasons, various transducers of sodium process parameter are developed and their graduation characteristic equations using the medium temperature as the parameter are given. A temperature on-line compensation type process parameter (flowrate, pressure and level) indicator is developed. Furthermore, the measurement systems of the intelligent type sodium flowmeter, the intelligent type immersed sodium flowmeter, the intelligent type sodium manometer and the intelligent type sodium level gauge are composed.

2. TRANSDUCERS AND THEIR GRADUATION CHARACTERISTIC EQUATIONS

(1) Sodium flowmeter

The construction of the sodium flowmeter is shown in Fig. 1. This is an instrument of measuring sodium volumetric flowrate by means of Faraday electromagnetic induction law. This is a permanent-magnet sodium flowmeter which magnet material is Alnico-5. Nowadays, various permanent-magnet sodium flowmeters which measuring range is respectively 0. 5, 5, 20 up to 800 m^3/h can be made in our institute.

In consideration of the temperature effect, in the sodium temperature limit of 200—550℃, the general form of the graduation characteristic equation of this sodium flowmeter is

$$Q = (aT^2 + bT + c) E \qquad (1)$$

where   Q —— The flowrate, m^3/h

         E —— The output signal EMF, mV

         T —— The sodium temperature, ℃

         a, b, c —— The coefficients.

(2) Immersed sodium flowmeter

The construction of the immersed sodium flowmeter is shown in Fig. 2. The immersed sodium flowmeter is a flowmeter of all-sealed construction,
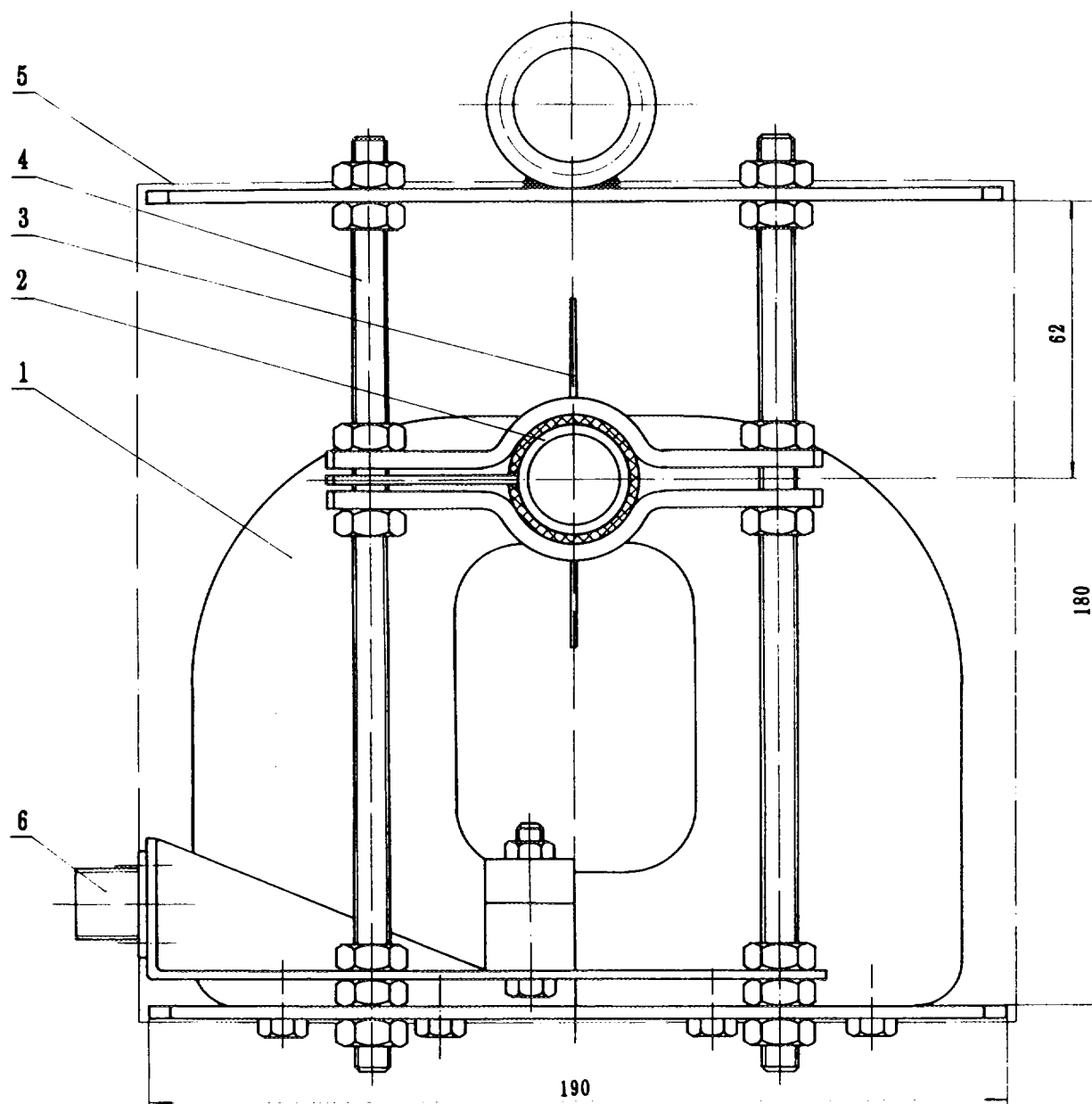
Fig. 1   The construction of the sodium flowmeter

1 —— Permanent magnet; 2 —— Sodium tube; 3 —— Signal electrodes;
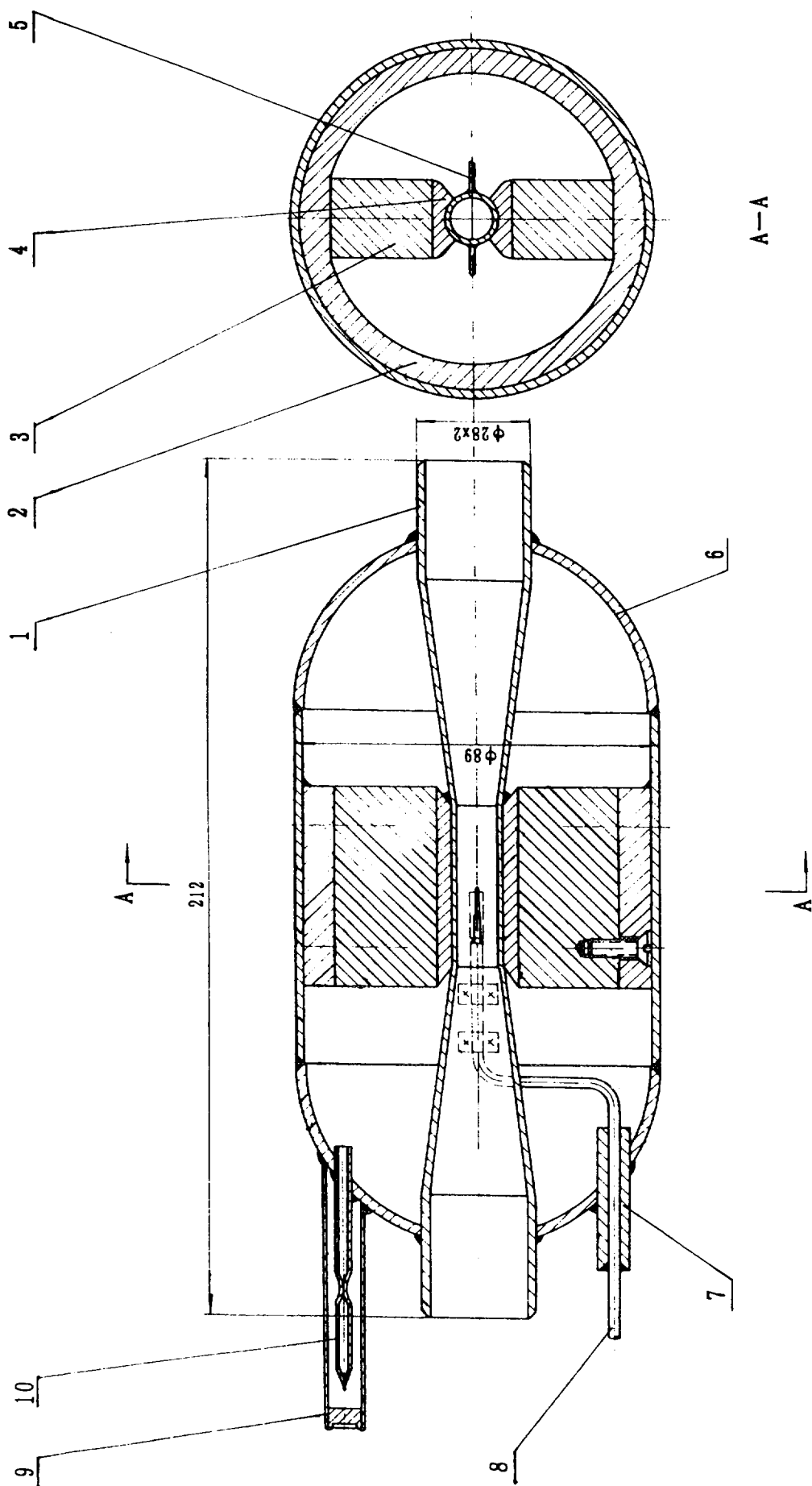4 —— Tube supports; 5 —— Bonnet; 6 —— Cable socket.

Fig. 2 The construction of the immersed sodium flowmeter

1 —— Sodium tube; 2 —— Magnetic circuit; 3 —— Permanent magnet;
4 —— Polar mass; 5 —— Signal electrodes; 6 —— Envelope;
7 —— Tight plug; 8 —— Sheathed signal cable;
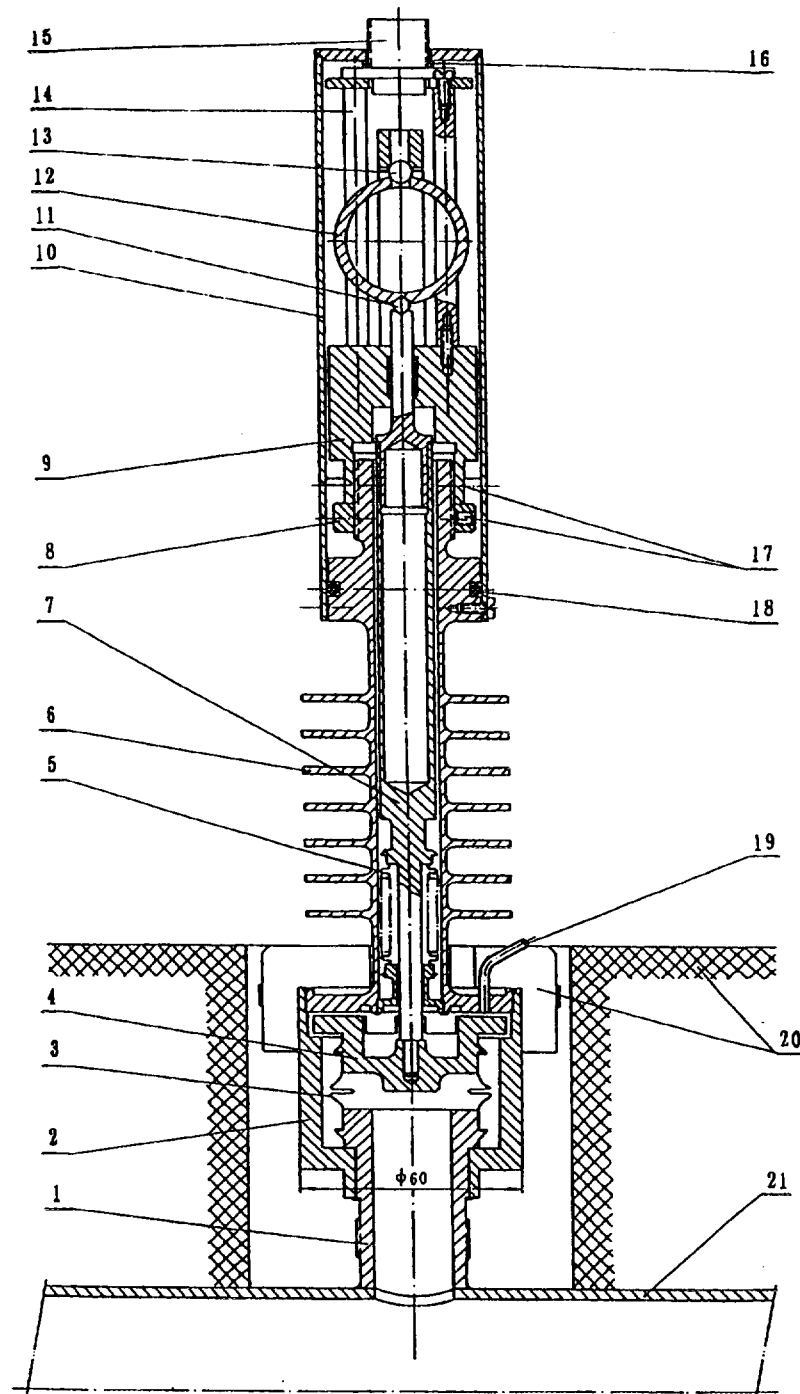9 —— Tight protective sleeve; 10 —— Pumping tube.

Fig. 3 The construction of the sodium pressure transducer

1 —— Join tube; 2 —— Enclosure; 3 —— Bellows; 4 —— Bridle support; 5 —— Bellows; 6 —— Radiator; 7 —— Central axis; 8 —— Blocking nut; 9 —— Stirrup frame; 10 —— Bonnet; 11 —— Ball; 12 —— Dynamometric annulus; 13 —— Ball; 14 —— Support; 15 —— Cable socket; 16 —— Rubber tight annulus; 17 —— Blocking screw; 18 —— Rubber tight annulus; 19 —— Evacuated tube; 20 —— Heat-insulated material; 21 —— Sodium tube.
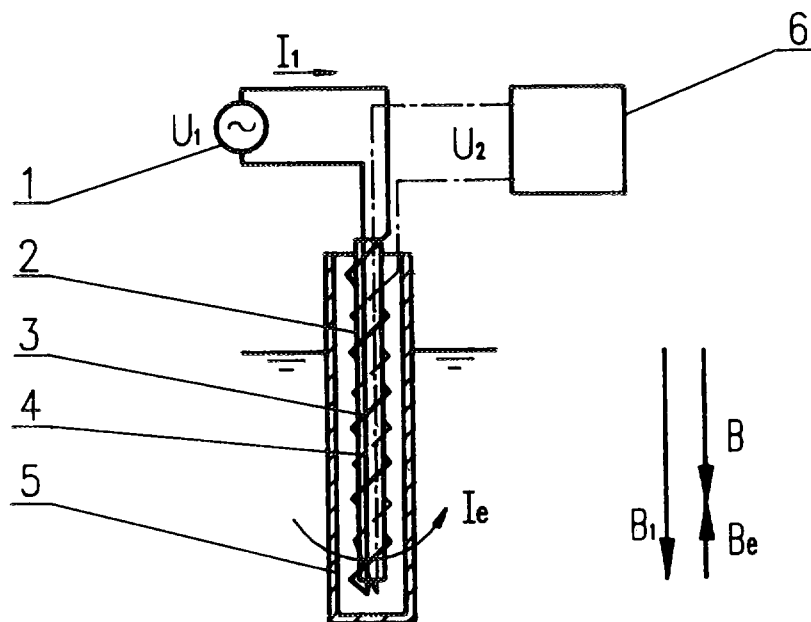
Fig. 4 The principle diagram of the mutual-inductance type sodium
level transducer

1 —— 1 kHz alternating stabilized current supply; 2 —— Pure ferro
core; 3 —— Primary winding;   4 —— Secondary winding;   5 ——
Stainless steel thimble; 6 —— Alternating millivoltmeter.

it can be operated in sodium. The immersed sodium flowmeter can be used
for resolve the flowrate measuring problems of the in-reactor
component of LMFR.

Nowadays, the immersed sodium flowmeter which measuring range is
3 m³/h has been made, and the immersed sodium flowmeter which measuring
range is 1200 m³/h has been designed in our institute.

The general form of the graduation characteristic equation of the
immersed sodium flowmeter is similar to the equation (1).

(3) Sodium pressure transducer

A high precision sodium pressure transducer whose sensor is the
dynamometric annulus is developed in our institute. Its measuring range
is 1 MPa. This sodium pressure transducer can be installed directly at
the sodium tube and resists the high temperature action up to 550℃ of
liquid metal. The construction of the sodium pressure transducer is
shown in Fig. 3. This sodium pressure transducer possesses of very good

linearity and less hysteresis error. High precision sodium pressure data can be obtained by on-line compensation of temperature effect.

In consideration of the temperature effect, in the sodium temperature limit of 120—550℃, the general form of the graduation characteristic equation of this sodium pressure transducer is

$$P = (aT^2 + bT + c) Us + dT^2 + eT + f \qquad (2)$$

where   P —— The pressure value, Pa

         Us —— The output voltage of the pressure transducer, mV

         T —— The sodium temperature, ℃

         a, b, c, d, e, f —— The coefficients.

(4) Mutual-Inductance type sodium level transducer

The principle diagram of the mutual-inductance type sodium level transducer is shown in Fig.4. Its sensor is constituted by the primary winding 3 and the secondary winding 4 that are both core wire of the stainless steel sheathed cable with MgO electrical insulation and wound round a pure ferro core. The sensor is inserted into the stainless steel thimble 5 that is immersed in liquid sodium. The primary winding is supplied by a 1 kHz alternating stabilized current supply 1, so the inductive voltage U_2 of the secondary winding is diminished linearly with the increment of the sodium immersion height. This is due to create the inductive eddy in sodium with the primary winding immersion by sodium. The direction of the eddy magnetic-flux density Be is opposed to the direction of the primary winding magnetic-flux density B_1, so that the composite magnetic-flux density B is diminished and the inductive voltage of the secondary winding is diminished linearly with the increment of the sodium level. This is an application of the eddy loss principle in the liquid metal level measurement. Nowadays, the mutual-inductance type sodium level transducers which sensitive length is respectively 800 and 1400 mm have been made in our institute.

In consideration of the temperature effect, in the sodium temperature limit of 120—550℃, the general form of the graduation characteristic equation of this sodium level transducer is

$$H = (aT^2 + bT + c) U_2 + dT^2 + eT + f \qquad (3)$$

where   H —— The insertion depth of the transducer in sodium, mm

         U_2 —— The output voltage of the transducer, mV

T —— The temperature of the medium measured, ℃

a, b, c, d, e, f —— The coefficients.

3. TEMPERATURE ON-LINE COMPENSATION TYPE PROCESS PARAMETER INDICATOR

As shown above, the measurement values of sodium process parameters (flowrate, pressure and level) are determined by the output voltage of the transducer and the temperature of the medium measured. Therefore, the temperature on-line compensation type process parameter indicator must be developed, in order to compensate on-line the variation of the transducer graduation characteristic due to the change of medium temperature.

Temperature on-line compensation type process parameter indicator newly developed consists of the digital part 1 and the analogue part 15 (see Fig. 5). The digital part is based on the microprocessor, it possesses of the functions of the time-shared measurement, the controls of the keyboard input and the digital output, etc., and the processing of the present sampling data are made. After the input of the graduation characteristic equation of transducer is confirmed, the multiple switch 10 is transformed at regular intervals by the microprocessor, and the values of E (or Us, or U_2) and T are inputed in turn. The temperature of the thermocouple cold-end is measured at regular intervals, in order that the accuracy of temperature measurement is increased.

Except the on-line compensation of the temperature effect can be made according to the graduation characteristic equation of the transducer and the value of the measured parameter is displayed directly, this indicator possesses of following functions still:

(1) The signal transformation

The output signal of transducer is transformed into the 0—10 V direct-current analogue signal, and the 10 V direct-current output voltage is corresponding to the full-scale value of the transducer. If necessary, the setting of full-scale can be changed by the keyboard input of the indicator, so that the 10 V direct-current output is corresponding to new full-scale value after the on-line compensation of the temperature effect.

Fig. 5 The block diagram of the temperature on-line compensation type
process parameter indicator

1 —— Digital part; 2 —— Microprocessor; 3 —— Alarm output
control; 4 —— Sampling control and analogue output control; 5 ——
Digital output; 6 —— Light coupler; 7 —— A / D converter; 8 ——
Compensation circuit of the thermocouple cold-end; 9 —— D / A
converter; 10 —— Multiple switch; 11 —— Monitoring circuit of
the thermocouple breaking; 12 —— AC / DC converter; 13 ——
Low-pass filter; 14 —— Amplification circuit; 15 —— Analogue part;
16 —— Keyboard and display; 17 —— Protection of power beaking;
U_2 —— Output signal of the level transducer; T —— Temperature of
the medium measured; Um —— 0—10 V direct-current analogue output;
E —— Output signal of the flowmeter; Us —— Output signal of the
pressure transducer.

185

(2) Two upper limits alarms and two lower limits alarms for process parameter

Two limits alarms are corresponding to the warning alarm and the accident alarm respectively. The alarm limit values can be set by the keyboard input. It possesses of the corresponding alarm relay contact output and the alarm symbols of display and print. This function is suitable particularly for the instrument, control and protective symtems of LMFR.

(3) The thermocouple breaking alarm

If the thermocouple breaking, it possesses of the corresponding alarm relay contact output and the alarm symbols of display and print.

(4) The self-inspection

The units that accept the self-inspection are the measuring circuit of the thermocouple cold-end, the A / D converter and the RAM, etc..

(5) CENTRONICS standard digital output

It is easy of the data acquisition and processing, and the in-service inspection of this indicator.

The main specifications of the temperature on-line compensation type process parameter indicator are following:

Measuring range of DC voltage —— 0—100 mV ;

Measuring range of AC voltage —— 0—200 mV ;

Measuring range of temperature —— 0 — 900°C (Matching K type thermocouple) ;

Measuring accuracy of DC voltage ——± (0.04% reading + 0.01% full-scale) ;

Measuring accuracy of AC voltage —— ± (0.2% reading + 0.2% full-scale) ;

Measuring accuracy of temperature —— ±2°C ;

Resolution of DC voltage —— 10 μV ;

Resolution of AC voltage —— 100 μV ;

Resolution of temperature —— 1°C ;

Display error of parameter ——± (0.1%reading + 0.05%full-scale);

Measuring speed —— 3 cps ;

Display renewal speed —— 1 cps ;

Analogue output voltage —— 0—10V DC (the load capacity is 10 mA);

Accuracy of analogue output —— ± (0.1% reading + 0.025% full-scale) ;

Digital output —— CENTRONICS parallel port ;

Display form of parameter —— Full five digits, with unit m^3/h, or kPa, or mm, or ℃, or mV DC, or mV AC ;

Relay contact output —— Normal opened contact, the contact capacity is 28 V DC and 1 A.

The input signals and the output signals of the indicator are each other isolative.

The block diagram of the temperature on-line compensation type process parameter indicator is shown in Fig. 5. The exterior of the temperature on-line compensation type process parameter indicator is shown in Fig. 6.
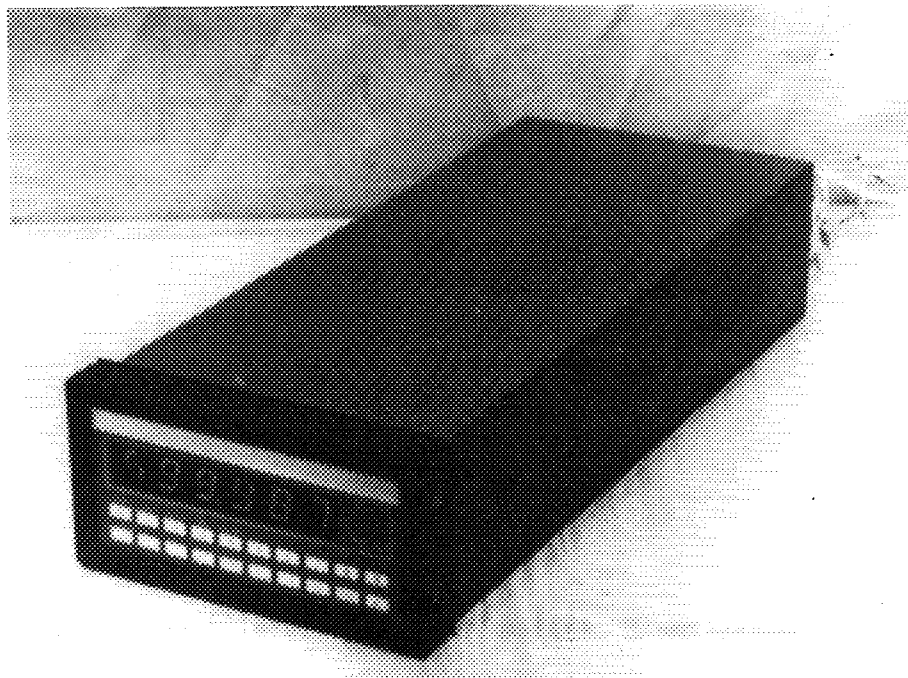
Fig. 6   The exterior of the temperature on-line compensation type process parameter indicator

187

4. MEASUREMENT SYSTEMS FOR INTELLIGENT TYPE SODIUM INSTRUMENTATIONS

The measurement system for intelligent type sodium flowmeter, immersed sodium flowmeter, sodium manometer and sodium level gauge is shown respectively in Fig. 7, Fig. 8 and Fig. 9. In that, the temperature on-line compensation type process parameter (flowrate, or pressure, or level) indicator is matched with the corresponding transducer, it compensates on-line the variation of the transducer graduation characteristic due to the change of medium temperature according to the transducer graduation characteristic equation that is inputed beforehand in it, ant it displays accurately the value of the process parameter.
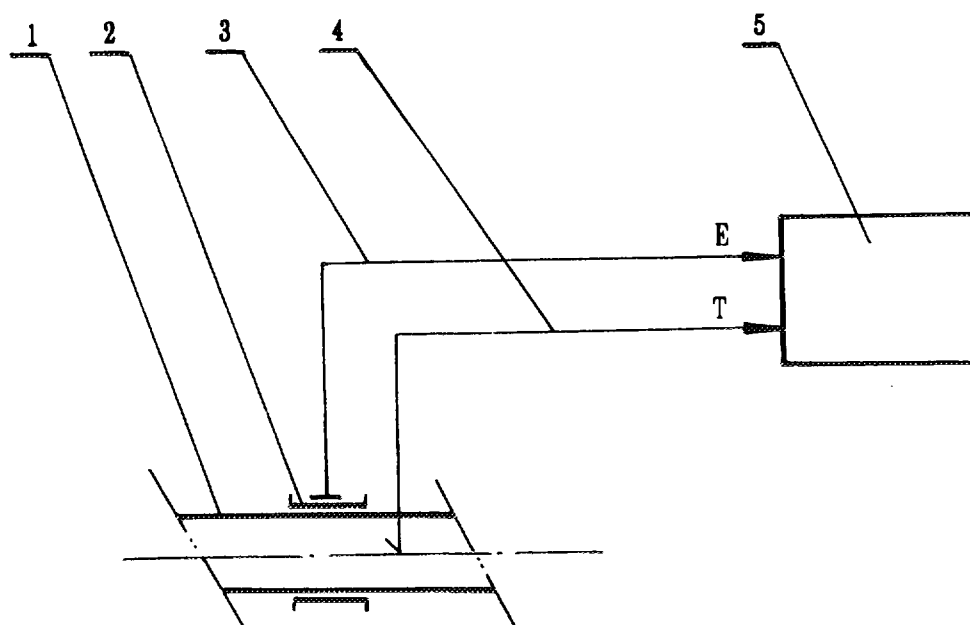
Fig. 7 The measurement system of the intelligent type sodium flowmeter (or Immersed sodium flowmeter)

1 —— Sodium tube; 2 —— Sodium flowmeter (or Immersed sodium flowmeter); 3 —— Two-lead shield cable; 4 —— Thermocouple; 5 —— Temperature on-line compensation type process parameter (flowrate) indicator; E —— Output signal of the flowmeter; T —— Sodium temperature.

Moreover, these intelligent type sodium instrumentations possess of the self-inspection, the electric shutoff protection, the setting of full-scale, the setting of alarm limits (two upper limits and two lower limits alarms), the thermocouple breaking alarm, each other isolative the 0—10V direct-current analogue output and CENTRONICS standard digital output, and the alarm relay contact output. These intelligent type sodium instrumentations are suitable particularly for the instrument, control and protective systems of LMFR by means of these excellent functions based on microprocessor. The basic error of the intelligent type sodium flowmeter, immersed sodium flowmeter, sodium manometer and sodium level gauge is respectively $\pm 2\%$, $\pm 2.3\%$, $\pm 0.3\%$ and $\pm 1.9\%$ of measuring range.

Fig. 8   The   measurement   system   of   the   intelligent   type   sodium
manometer

1 —— Sodium tube;   2 —— Sodium   pressure   transducer;   3 ——
Thermocouple; 4 —— Four-lead twisted pairs sheild cable;   5 ——
Precise DC stabilized voltage supply; 6 —— Two-lead shield cable;
7 —— Temperature on-line compensation type process parameter (pressure) indicator;   Us —— Output   signal   of   the   pressure transducer; T —— Sodium temperature.
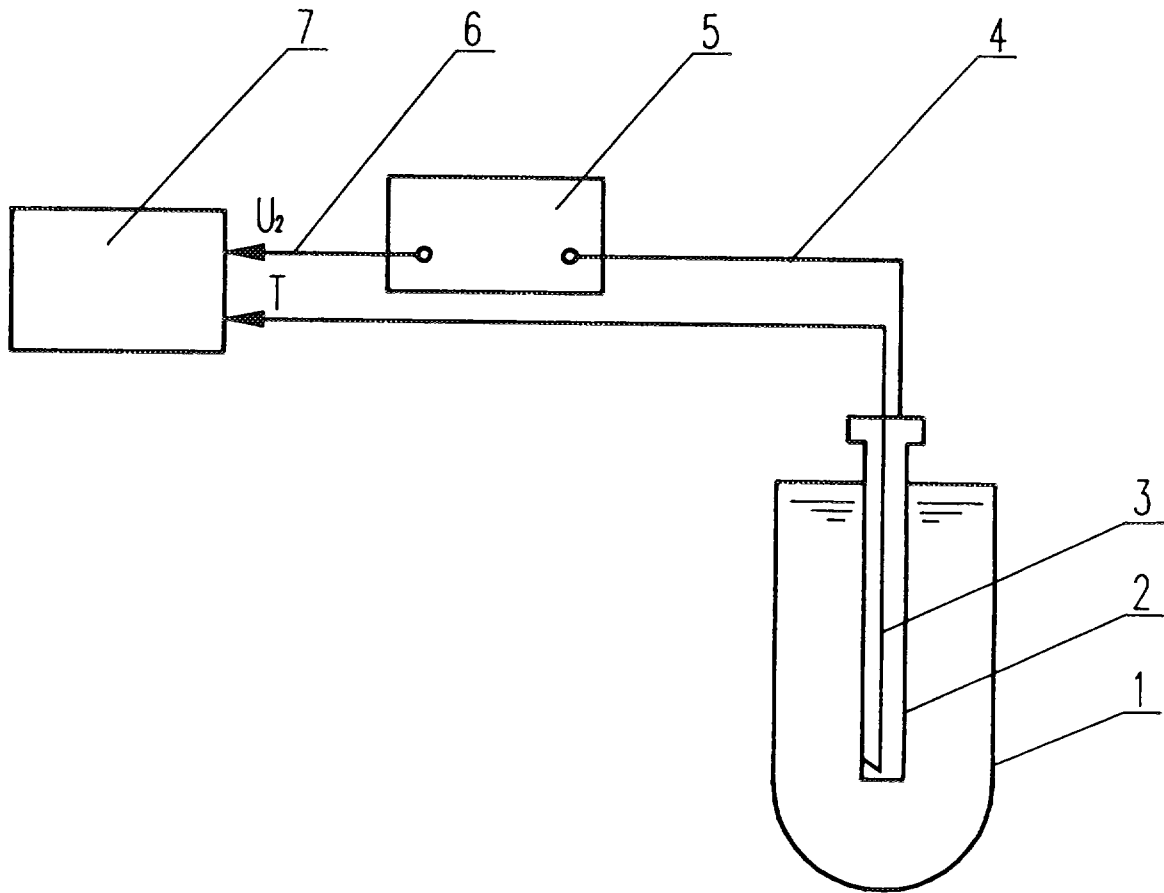
Fig. 9 The measurement system of the intelligent type sodium level
gauge

1 —— Sodium container; 2 —— Mutual-inductance type sodium level
transducer; 3 —— Thermocouple; 4 —— Four-lead twisted pairs
shield cable; 5 —— 1 kHz alternating stabilized current supply;
6 —— Two-lead shield cable; 7 —— Temperature on-line compensation
type process parameter (level) indicator; U_2 —— Output signal of
the level transducer; T —— Temperature of the medium measured.

5. CONCLUSION

(1) In order to obtain the accurate measurement data of process
parameter, the transducer graduation characteristic equation using the
medium temperature as the parameter must be made by means of the
transducer graduation calibration test firstly.

(2) The construction composition and the functions design of the
measurement systems for intelligent type sodium flowmeter, immersed

sodium flowmeter, sodium manometer and sodium level gauge are reasonable and practical. Their basic errors can satisfy the needs of LMFR engineering and its test installations.

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SAFETY FEATURES AND CORE PERFORMANCE OF KALIMER

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Abstract

KALIMER (Korea Advanced LIquid MEtal Reactor) is an economically competitive, inherently safe, environmentally friendly, and proliferation-resistant liquid metal reactor which is now being developed by the Korea Atomic Energy Research Institute. A modular, pool-type sodium cooled KALIMER is currently in initial concept study phase with the goal of its construction to be completed by the year 2011.

KALIMER produces 333 MWe per reactor module with the employment of modular single component IHTS concept which can reduce the cost of IHTS by combining the three major components, i.e., steam generator, intermediate sodium pump, and intermediate sodium expansion tank into a single vessel thereby reducing the quantity, complexity, and space required by the IHTS.

Passive safety features of KALIMER design include the Reactor Vessel Auxiliary Cooling System (RVACS) which assures safety-grade decay heat removal and the Self-Actuated Shutdown System (SASS) for reactor trip. The core nuclear design will be largely governed by passive safety and reactivity control issues. KALIMER core is fueled with metallic fuel, and the initial core will be loaded with 20% enriched uranium metal fuel.

This paper summarizes the safety features of KALIMER design and the ATWS performance of Pu and U metal core options.

1. INTRODUCTION

The Korean national liquid metal reactor development plan was approved by the Korea Atomic Energy Commission in 1992, with the goal of developing a liquid metal reactor which can serve as a long term power supplier with competitive economics and enhanced safety. The KALIMER Program is now being led by the Korea Atomic Energy Research Institute (KAERI) with the objectives of developing an economically competitive, inherently safe, environmentally friendly, and proliferation-resistant fast reactor concept. A modular, pool-type sodium cooled KALIMER is currently in initial concept study phase with the goal of its construction to be completed by the year 2011.

The KALIMER plant will compete economically with contemporaneous alternative electrical generation options including both Advanced Light Water Reactors

(ALWRs) and fossil plants. This can be achieved by the simplification of the intermediate heat transfer system (IHTS), the elimination of rotating plug with the use of variable arm pantograph type fuel handling machine, and the introduction of seismic isolators.

KALIMER has enhanced safety features with the use of metallic fuel, Reactor Vessel Auxiliary Cooling System (RVACS), Self-Actuated Shutdown System (SASS), Gas Expansion Module (GEM) in the core, and the reduction of sodium piping above reactor vessel for the prevention of major sodium fires. Utilization of these enhanced safety features eliminates the need for diverse and redundant engineered safety systems so that "walk-away" safety characteristics are achieved. KALIMER accommodates unprotected anticipated transients without scram (ATWS) events without operator action, and without the support of active shutdown, shutdown heat removal, or any automatic system without damage to the plant and without jeopardizing public safety.

Environmentally friendly KALIMER has extremely low probability and amount of accidental radioactivity releases. The KALIMER core is loaded with metallic fuel which is recycled through pyroprocessing. Recycling of transuranic elements by this process would avoid the expense and potential long-term risk of their disposal in a geological repository, and would provide increased proliferation resistance.

The costs and schedules for KALIMER development will be minimized by standardizing the design and demonstrating the plant's operational and safety features in a full-scale test of a single nuclear steam supply system (NSSS). The modular design will allow a full commercial sized module and its associated NSSS equipment to be tested, eliminating the need to scale up the size of the components in a series of costly demonstration plants. The standard KALIMER design will be such that it can be certified by the Korea Institute of Nuclear Safety.

The KALIMER will be designed utilizing the available liquid metal reactor technology base, both foreign and domestic. Wherever cost advantages can be gained, latest state-of-the-art technology will be utilized.

2. KEY DESIGN FEATURES OF KALIMER

The standard KALIMER plant consists of power blocks which comprise multiple reactor modules with the power rating of 333 MWe per reactor module shown in Figure 1. Each power block consists of one or more reactor module systems and power conversion systems, together with their associated instrumentation, controls and auxiliary systems. The reactor core is designed to accommodate the flexible core envelope which permits use of various fissile materials and allows different breeding/ conversion ratio core configurations, including actinide burning capability. Table 1 summarizes the key design features of KALIMER.

The design features unique to KALIMER include the use of integrated steam generators, elimination of rotating plugs and simplification of in-vessel transfer machine, and volume reduction of intermediate sodium above the reactor vessel.

KALIMER (333MWe)

Korea Advanced Liquid MEtal Reactor

- Enhanced Economics
- Inherently Safe
- Environmentally Friendly
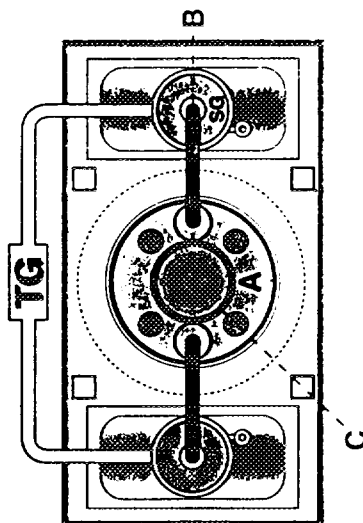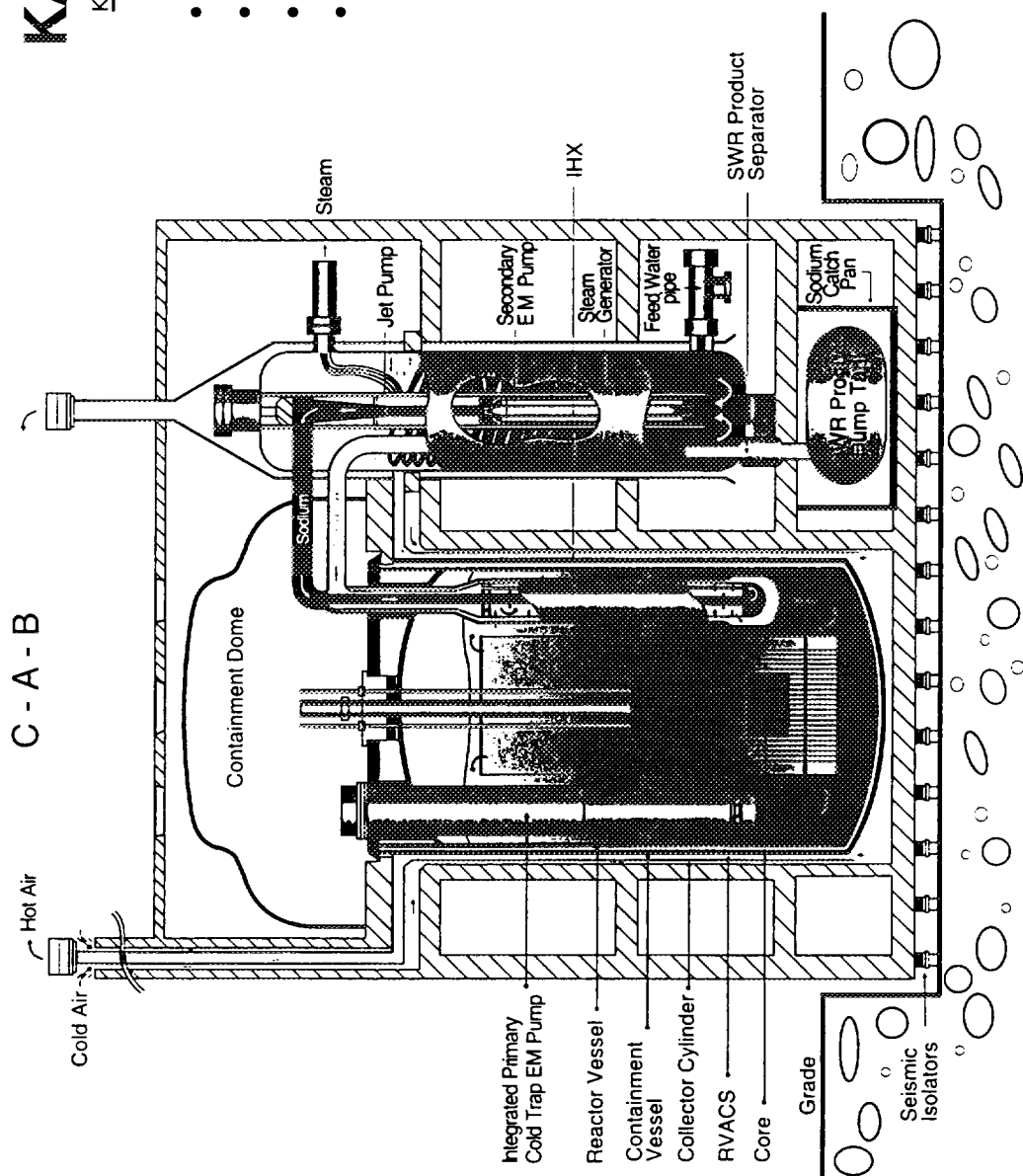- Nuclear - Proliferation Resistant

Figure 1. KALIMER Reactor Module and Steam Generator

195

Table 1. Key Design Features of KALIMER

Reactor	Modular Pool Type
Electrical Power	333 MWe / Reactor Module
Efficiency	40 % (target)
Fuel	Metal
Initial Core	20 w/o enriched U metal
Primary Sodium I/O Temp.	375 $^{\circ}$C /530 $^{\circ}$C (target)
Reactor Vessel	
Diameter	~ 9 m
Height	~15 m
Primary Pumps	
Type	Electromagnetic
Number	4
Number of IHXs	2
Steam Generator	
Number of Units	2
Type	Integrated with Secondary EM pump
Shutdown Heat Removal	RVACS
Reactor Shutdown System	SASS
Fuel Handling	Variable Arm Pantograph Type IVTM
Seismic Design	Seismic Isolation Bearing

KALIMER adopts the modular or consolidated IHTS concept which combines the three major components, i.e., steam generator, intermediate sodium pump, and intermediate sodium expansion tank, into a single compact component. This concept makes use of boiling water reactor (BWR) jet pump technology to reduce the number of electromagnetic (EM) pumps necessary for the required intermediate sodium flow rate, and thus reduces the diameter of the modular IHTS, which results in a substantial reduction in the cost of IHTS.

Elimination of rotating plugs is achieved by the use of variable arm pantograph type fuel handling machine which is plugged in during the refueling period for the in-vessel fuel transfer. In this case the upper internal structure which consists of the control rod driving guide tubes and mechanism, and the instrumentation plug will be pulled out for easy access of the fuel handling machine to reactor core.

The reactor building of KALIMER is on a 0.3g safe shutdown earthquake (SSE) seismic isolation system. This seismically isolated building is a super structure in which the reactor vessel and steam generators are located. Application of innovative seismic isolation system is expected to result in the improvement of economics and safety of KALIMER.

3. KALIMER SAFETY DESIGN OBJECTIVES

The overall goal of the KALIMER effort is to develop an advanced inherently safe, reliable, and marketable liquid metal cooled reactor power plant which will be economically competitive with alternative nuclear power plants. The safety design of KALIMER emphasizes accident prevention by using passive and natural processes, which can be accomplished by the following safety design objectives:

● Utilization of inherent safety features to eliminate the need for diverse and redundant engineered safety systems so that "walk-away" safety characteristics are achieved.

● Accommodation of unprotected ATWS events such as UTOP, ULOF, and ULOHS without operator action, and without the support of active shutdown, shutdown heat removal, or any automatic system without damage to the plant and without jeopardizing public safety.

● Low probability and amount of accidental radiation releases beyond the limits of the site boundary, which eliminates the need for detailed offsite evacuation planning, exercises, and early warning.

4. KEY SAFETY FEATURES OF THE KALIMER DESIGN

In order to achieve the above safety design objectives, the following design features are desired:

A. PASSIVE DECAY HEAT REMOVAL

KALIMER design requirements specify that each reactor module incorporates its own independent passive heat removal system which will protect the public health and safety following the complete loss of the normal heat removal system, without bulk AC power, without any operator action, following defined design basis events.

Reactor shutdown heat is normally removed by the turbine condenser using the turbine bypass. An Auxiliary Cooling System (ACS) is provided for cases when, due to maintenance or repair needs, an alternative shutdown heat removal method is required. The ACS induces natural circulation of atmospheric air past the shell side of the steam generator, and normal, natural circulation ACS operation is initiated by opening the exhaust damper. ACS operation in a natural circulation mode is expected to have the capability to maintain reactor temperatures below design limits.

In the highly unlikely event that the IHTS becomes unusable during power operation, for example, because of a main sodium pipe break or sodium dump, the reactor will scram and the RVACS will automatically come into full operation.

Temperatures of the reactor sodium and reactor vessel will rise, increasing the radiant heat transfer across the gap to the containment vessel and the heat transfer from the containment vessel to the upwardly flowing atmospheric air around the vessel. The temperatures and heat transfer by RVACS will continue to increase until equilibrium between reactor heat generation and RVACS cooling is established.

B. REACTIVITY CONTROL AND SHUTDOWN

In order to meet design requirements, two independent reactivity control systems employing different design principles will be provided. Each system ensures the reactor be maintained in a safe shutdown state under all operating and postulated accident conditions, assuming failure of the other system. The reactor protection system will have sufficient redundancy and independence to assure that no single failure results in loss of reactor function, and removal from service of any component or channel does not result in loss of the required minimum redundancy.

Negative feedback enhancers being considered include Gas Expansion Modules (GEMs) which enhance the negative reactivity feedback during a loss of flow without scram event, and a rod stop system which limits reactivity addition during a rod withdrawal without scram event.

KALIMER design adopts a passive shutdown system SASS which actuates by the naturally occurring physical phenomenon, i.e., saturation of magnetization of the ferromagnetic materials at Curie-point, without any external driving force. SASS consists of a Curie-point electromagnet and an articulated rod type neutron absorber assembly. Articulated rod can guarantee the insertion of the control absorber assembly into the reactor core even when the control rod guide tube is distorted due to the seismic load.

C. INHERENT NEGATIVE REACTIVITY CONTROL

The KALIMER is designed to provide a strong inherent negative reactivity feedback with rising temperature. This characteristic, combined with the RVACS heat removal capability, makes the KALIMER capable of safely withstanding severe undercooling and overpower transient events without scram.

As the temperature increases during an event, the negative feedback from Doppler, axial fuel expansion, radial core expansion, and control rod driveline expansion are activated, which generate a net negative reactivity for the core loaded with metal fuel. This feedback responds according to the associated time constants, to overcome the positive reactivity from the sodium density / void effect and any external source.

198

D. SEISMIC ISOLATION AND ELECTROMAGNETIC PUMPS

One of the challenges to standardized design and nuclear power plant siting has been the difficulty of incorporating a standard design on sites of differing seismic characteristics. The KALIMER reactor module design overcomes this problem by incorporating a seismic isolation to reduce the lateral seismic loads on the reactor structure and internals. The reactor module and steam generators with all safety related systems rest on seismic isolators, simplifying reactor component designs and significantly increasing safety margins. By placing the KALIMER NSSS on a seismically isolated platform, and by keeping the vessel diameter small, the seismic capability of KALIMER will allow it to be placed in most possible sites.

The primary and intermediate pumps are electromagnetic (EM). Synchronous motors, which provide coastdown power to the EM pumps, are on the same seismically isolated platform as the reactor module. The use of EM pumps and seismically isolated synchronous motors minimizes the potential for rapid coastdown in a seismic event, enhancing the natural safety characteristics of the plant.

One of the KALIMER safety design features also to be noted is the minimized volume of intermediate sodium above the reactor vessel for the prevention of major sodium fires should the sodium piping break occurs.

5. ACCOMMODATION OF ANTICIPATED TRANSIENTS WITHOUT SCRAM

KALIMER passively accommodates the ATWS events, and the plant response to the ATWS events meet criteria with adequate margins. KALIMER has inherent passive means of negative reactivity insertion and decay heat removal, sufficient to place the reactor system in a safe stable state for bounding ATWS events without significant damage to the core or reactor system structure.

In order to improve the KALIMER design and to investigate inherent safety features from the initial concept study phase, preliminary evaluation of ATWS performance for KALIMER core options has been performed.

A. CORE DESIGN OPTIONS

One of the options being considered for a KALIMER reactor core is the design which utilizes a homogeneous core configuration allowing a compact core and no fuel shuffling. The layout consists of 115 driver fuel assemblies, 42 radial blanket assemblies, 6 control rods, and 174 shield assemblies. The inner five rows of the core consist of low enrichment fuel assemblies and six control rods. The outer radial core section contains two rows of high enrichment fuel assemblies. Six control rods are located between two enrichment zones, and the driver fuel zones are surrounded by one row of radial blanket assemblies. There are no upper or lower axial blankets surrounding the core.

A comparison of the core performance parameters for the plutonium (Pu) and uranium (U) fueled KALIMER cores is summarized in Table 2. Since the U core is a

direct substitution for the Pu core, i.e., both cores have the identical core layout, and assembly design data, the differences in the core performance parameters are directly attributed to the differences in the neutronics characteristics of the U-235 and fissile plutonium.

Table 2. Comparison of Core Performance Parameters

	Pu-U-Zr	U-Zr
Average Breeding / Conversion Ratio	1.12	0.71
Loaded Fissile Enrichment, w/o	11.1	19.8
Total Beta-effective	0.00355	0.00607
Reactivity Feedback Coefficients		
Doppler, Tdk/dt	-0.00309	-0.00313
Sodium Density, delta K	-0.02579	-0.00297
Axial Fuel Expansion, Hdk/dH	-0.18237	-0.18643
Radial Expansion, Rdk/dR	-0.45955	-0.46352
Sodium Void Reactivity, $	5.21846	-0.38580

The uranium fueled core requires a much higher fissile enrichment of 19.8 % in the feed fuel than that of 11.1 % for the corresponding plutonium core. Since U-235 is neutronically less effective than Pu-239 in a fast reactor system, a higher fissile enrichment is required to achieve criticality in the uranium fueled system.

The increase in the fissile enrichment has several important effects on the performance characteristics of the uranium core. The average conversion ratio is significantly lower due to the increased fissile depletion and reduced U-238 captures.

The total delayed neutron fraction for the uranium core is about 1.7 times larger than that of the corresponding plutonium core. This is due to an inherently higher beta-effective for the U-235 fission reaction. The higher delayed neutron fraction is expected to have a significant impact on the reactor kinetics during plant transients.

The uranium core has negative sodium density/void reactivity feedback coefficients, instead of the large positive void reactivity for the plutonium core. This is because i) the variation of capture-to-fission ratio due to spectral hardening is less pronounced in U-235, ii) the contribution of the higher plutonium isotopes, especially Pu-240, to the spectral shift reactivity is significantly smaller in the uranium core, iii) the impact of the spectral shift on the fast fission of U-238 is less important in the uranium core, and iv) the increase in the neutron leakage due to sodium voiding is more pronounced in the U-235 fueled core. The negative sodium void reactivity in the uranium core should have significant impacts on the plant operation and safety related issues.

B. ATWS SAFETY CRITERIA

According to the design requirement, KALIMER is to accommodate ATWS events, specifically UTOP, ULOF, and ULOHS, so that core damage leading to a safety challenge does not occur.

Conservative safety criteria are to be established in order to insure that the requirements for ATWS events are met. Safety criteria to be considered include the limited number of cladding failures, maintenance of primary boundary integrity, no sodium boiling, and no positive reactivity addition from fuel movement. Temperature limits are then to be set, based on current knowledge of experimental data, to insure that these safety criteria are met. The temperature limits are dependent on the specific fuel and cladding compositions, and are subject to revision as additional experimental test data become available.

Preliminary temperature limits for accommodated ATWS events of the KALIMER are 1820 ^{0}F (over which for less than two minutes) and 1880 ^{0}F for peak fuel temperatures of Pu and U metal fuels, respectively. In order to prevent sodium boiling, the temperature limit for sodium is set at 1750 ^{0}F when no sodium pumps are operating.

C. ATWS PERFORMANCE

A conceptual design of the plutonium and uranium core options for KALIMER has been evaluated. The uranium core was constrained to fit within the permanent reactor structure and to produce the same power as required for the plutonium fueled KALIMER core. ATWS events of UTOP, ULOF and ULOHS are selected for the safety margin assessment of the KALIMER design.

The objectives of ATWS performance analyses are to evaluate the inherent passive safety features and to compare the performance of the Pu and U cores. It should be noted that the KALIMER core option being analyzed has not been optimized yet, and thus there is a room for the improvement in its safety performance.

Due to its rapid progression of the UTOP event, for which case the safety margin is usually determined by the fuel centerline temperature, there is little effect of the reactor configuration on the consequences of this event.

The rapid coastdown of both primary and intermediate EM pumps in the case of ULOF event would cause rapid temperature rises which introduce the negative reactivity feedbacks to come in to play for the power decrease of the reactor. Analyses have been performed for the effects of primary pump coastdown on core safety.

For a ULOHS event, a loss of intermediate EM pump is assumed with the subsequent primary EM pump trip by the Thermal Shutoff System (TSS) which actuates based upon the core inlet sodium temperature and power levels.

All Primary Rods Withdrawal Without Scram   (Figs. 2-7)

This event postulates that a malfunction in the reactivity controller causes the shim motor to continue to withdraw all of the primary control rods and that the Reactor Protection System (RPS) either fails to detect the event or the control rods fail to unlatch. The shim motors are assumed to withdraw the control rods at a rate corresponding to $0.02 per second. The secondary control rods are completely withdrawn during normal operation, and it is assumed that these rods do not contribute to the reactivities inserted for this accident.

It is assumed that the primary and secondary sodium flows remain at rated conditions for this event and that the feedwater is sufficient to keep the sodium outlet temperature from the steam generator constant.

The results of $0.20 UTOP transient for power and flow, the core temperatures, and the reactivity feedbacks are shown in Figures 2 through 7 for Pu and U cores.

For $0.20 UTOP, the Pu core power level reaches its peak at 165% of rated condition and equilibrium at 125% power. The peak fuel and peak coolant temperatures are 1880 ^{0}F and 1423 ^{0}F, respectively. Core thermal expansion, which is the sum of axial fuel expansion, radial core expansion, and rod bowing, and control rod driveline expansion provide negative reactivities. Although the sodium expansion provides large positive reactivity, the net reactivity becomes negative due to core thermal expansion, Doppler, and control rod driveline expansion.

The core power level changes are similar for U and Pu cores, however, the peak fuel and peak coolant temperatures are much lower for U core due to the higher thermal conductivity of uranium metal fuel. The sodium expansion reactivity is much smaller for U core, but the Doppler and core thermal expansion reactivities are less negative due to the smaller temperature increases of the core.

Although the design meets the performance limits for ATWS events, the limiting condition appears to be the clad temperature at the elevated equilibrium conditions. This temperature must remain below the 1300 ^{0}F limit to prevent eutectic formation because the reactor could be in that state indefinitely.

For $0.30 UTOP, the peak core power increases over 200% of rated power for both Pu and U cores. The peak fuel temperatures violates the limits of 1820 ^{0}F and 1880 ^{0}F, for Pu and U cores, respectively. However, the peak coolant temperatures are within the limits.

The performance of KALIMER Pu and U core options during selected ATWS events including UTOP is summarized in Table 3.
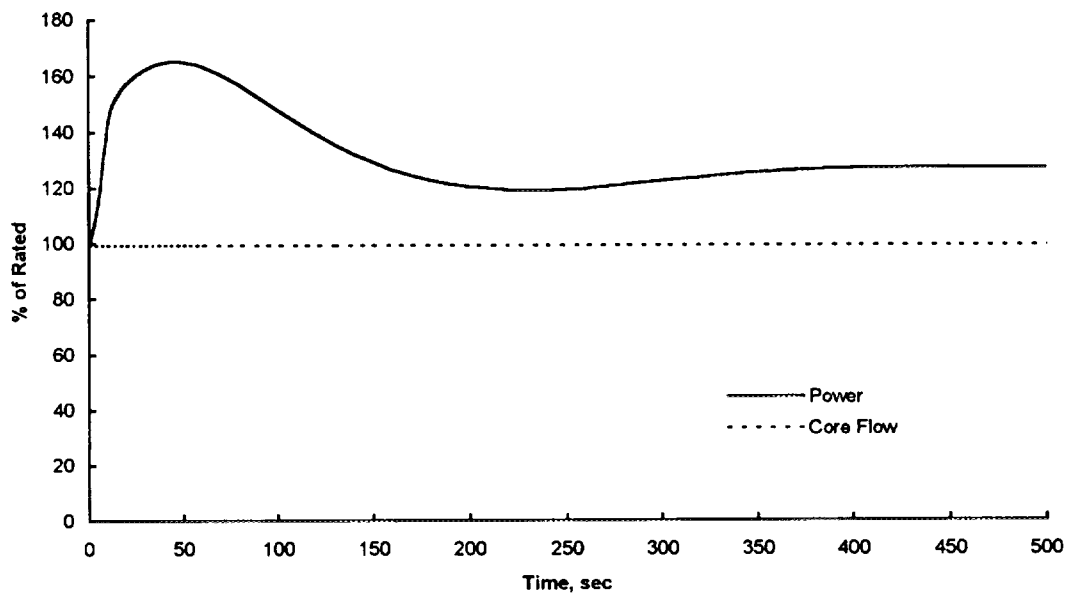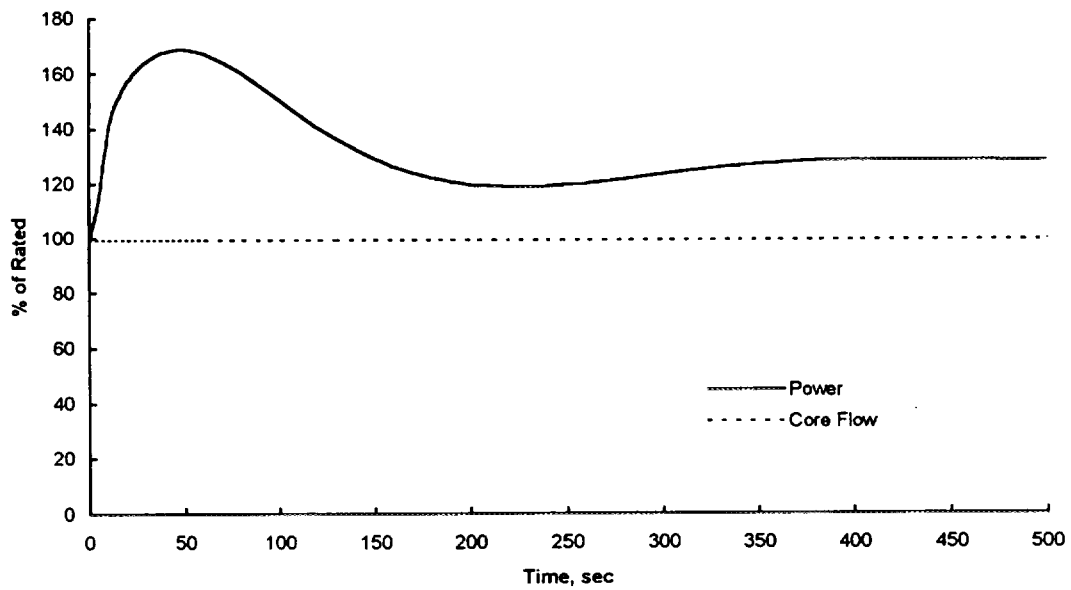
Figure 2. Power and Flow During $0.20 UTOP for Pu Core
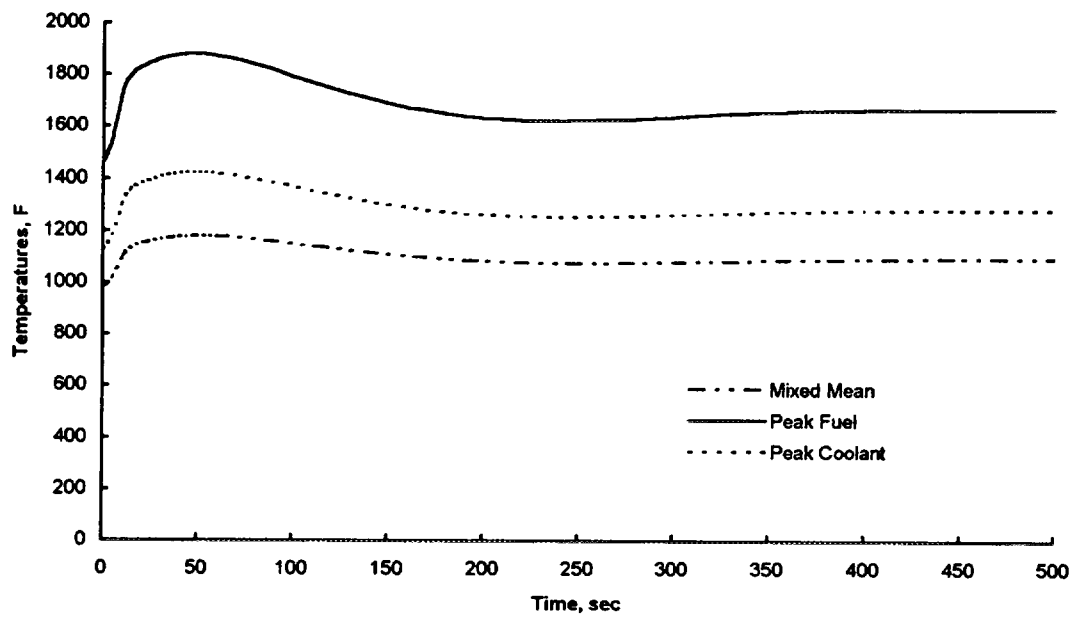

Figure 3. Power and Flow During $0.20 UTOP for U Core
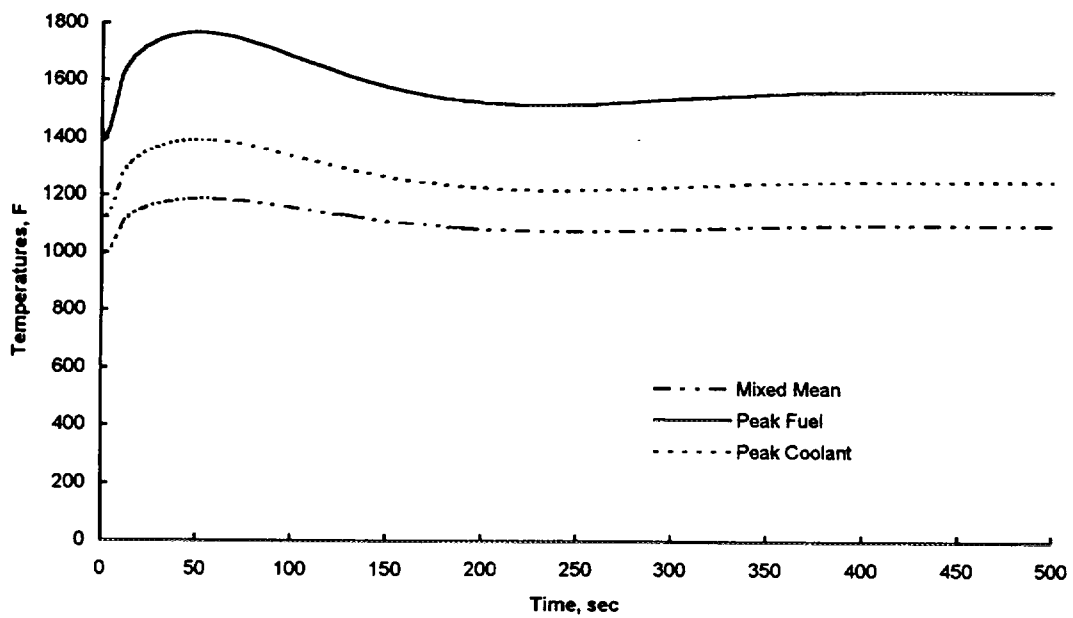
Figure 4. Temperatures During $0.20 UTOP for Pu Core

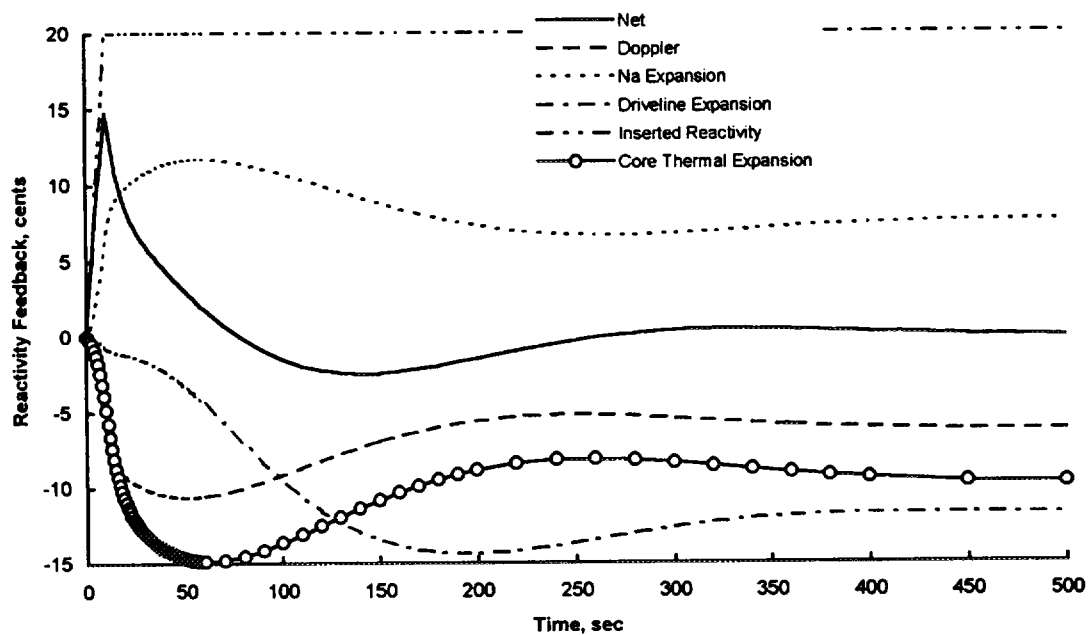Figure 5. Temperatures During $0.20 UTOP for U Core

204

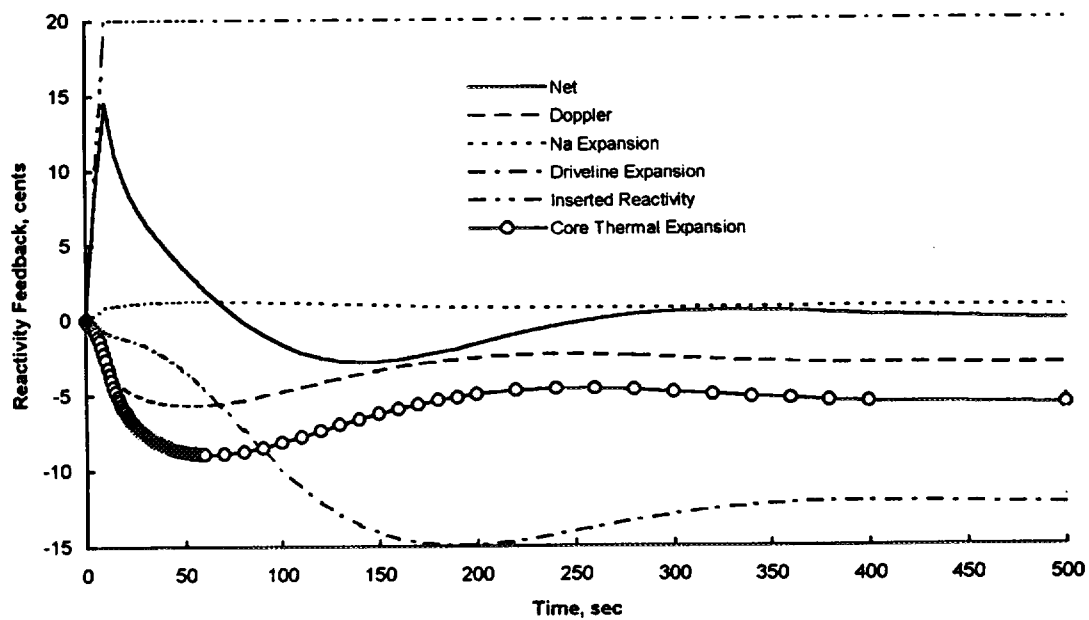Figure 6. Feedbacks During $0.20 UTOP for Pu Core

Figure 7. Feedbacks During $0.20 UTOP for U Core

Table 3. Summary of ATWS Performance of KALIMER Core Options

Events	Peak Power, %		Peak Fuel		Peak Coolant	
	Pu	U	Pu	U	Pu	U
Temperature Limits			1820	1880	1750	
100 % Power	100	100	1465	1445	1131	1131
$0.20 UTOP	165	169	1880	1767	1423	1391
$0.30 UTOP	206	209	2114	1974	1600	1553
Unprotected Loss of Primary Flow	100	100	1736	1609	1669	1544
Unprotected Loss of Intermediate Flow	100	102	1522	1445	1507	1390

Temperatures, °F

Unprotected Loss of Primary Flow (Figs. 8-13)

For the ULOF event, the IHTS flow is assumed to be at the rated condition, and the primary pumps are assumed to coastdown. The heat removal from the reactor vessel by RVACS has not been modeled and the reactor heat is removed by the normal path of IHTS. After the coastdown, the primary pumps are in a low flow operation mode, providing about 13% of the rated primary flow. Although this would normally result in a scram due to a high flux-to-flow ratio soon after the initiation of the coastdown, it is assumed that either the RPS fails to detect the mismatch or the control rods fail to insert.

The transient results for power and flow, the core temperatures, and the reactivity feedbacks during the ULOF event are shown in Figures 8 through 13 for the Pu and U cores.

The primary flows drop to 49.64% of rated flow at 10 seconds after the initiation of the coastdown, and the flow rate is maintained at 13% of the rated flow for the low flow operation mode which begins at about 150 seconds. Core power decreases gradually due to the negative feedback effects, and U core power decreases more rapidly.

The peak temperatures occur after about 50 seconds, which is mainly because of gradual coastdown of the primary pumps. The temperatures reach maximum values of 1736 °F and 1669 °F for peak fuel and peak coolant temperatures, respectively, for Pu core. These temperatures are within limits. The peak fuel and peak coolant temperatures are much lower for U core due to the higher thermal conductivity of uranium metal fuel. There is a large safety margin for peak fuel temperature, and it is noted that the peak coolant temperature is a key parameter to meet the safety limits for this event.
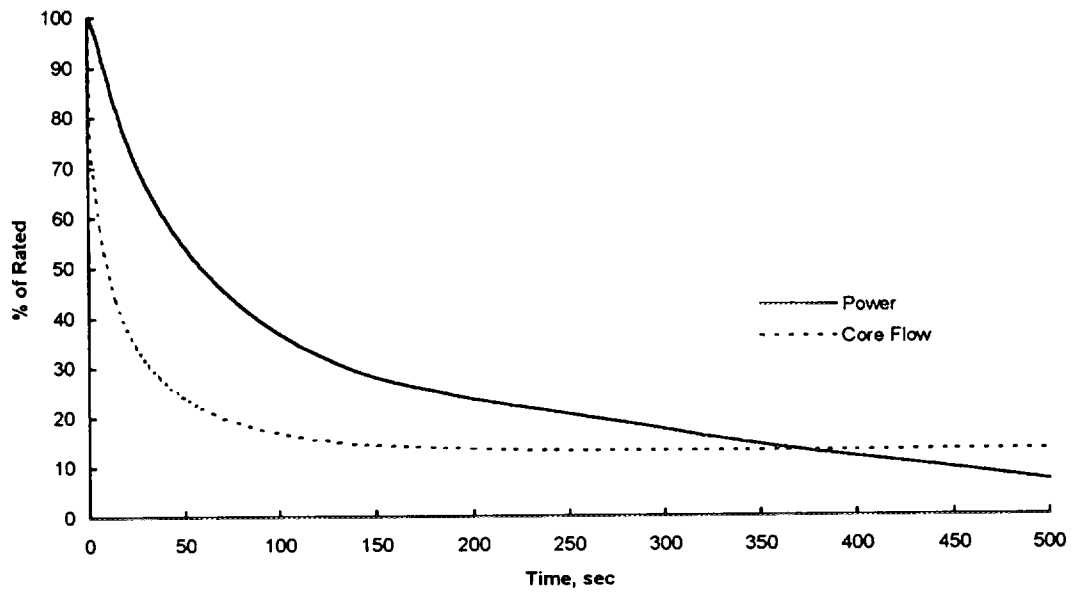
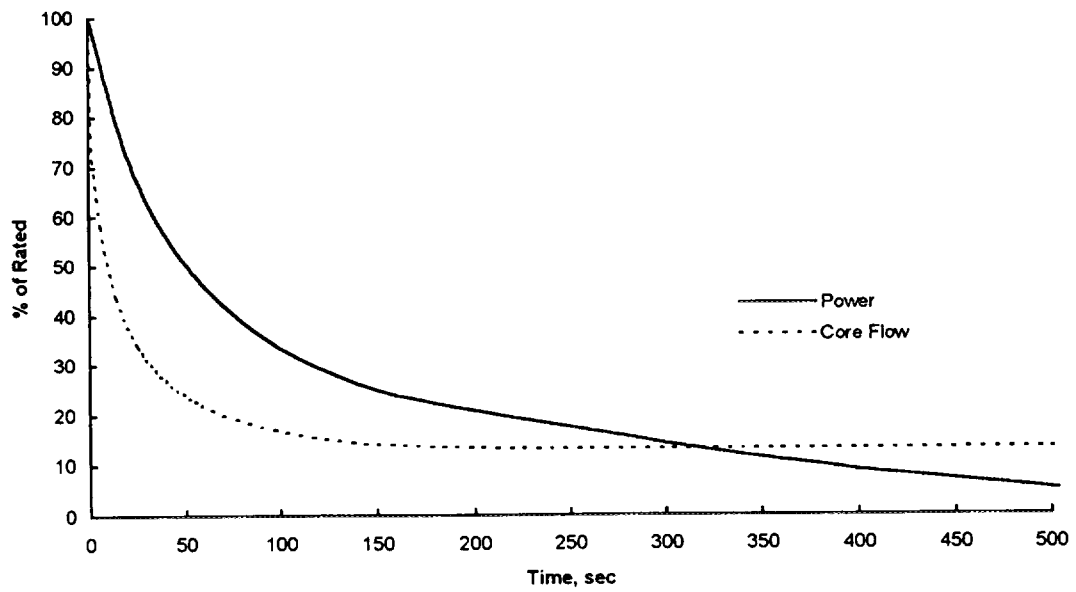Figure 8. Power and Flow During ULOF for Pu Core

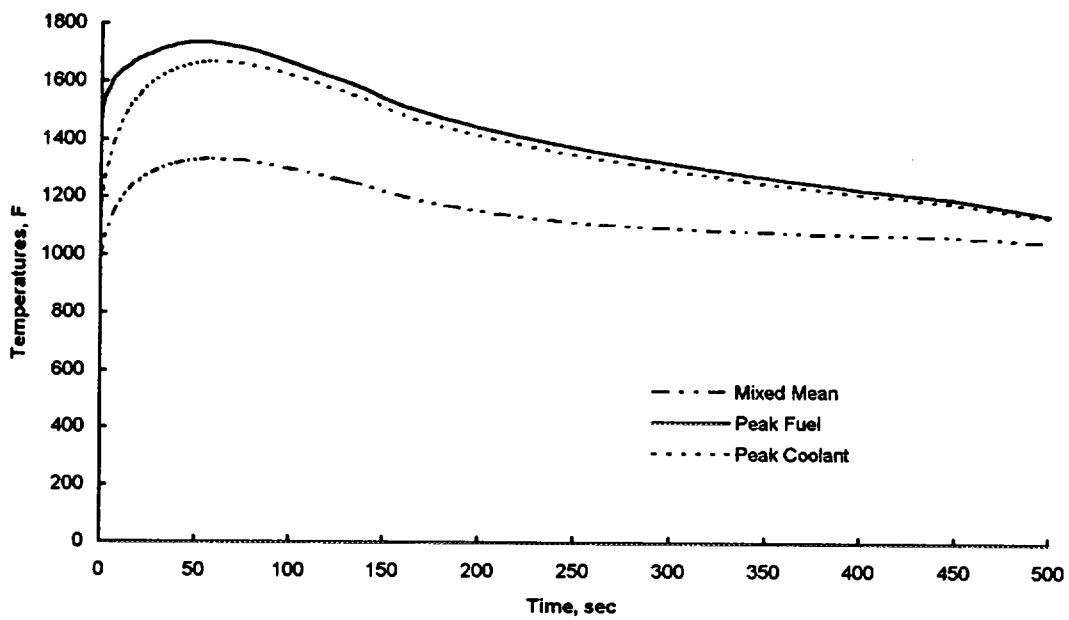Figure 9. Power and Flow During ULOF for U Core
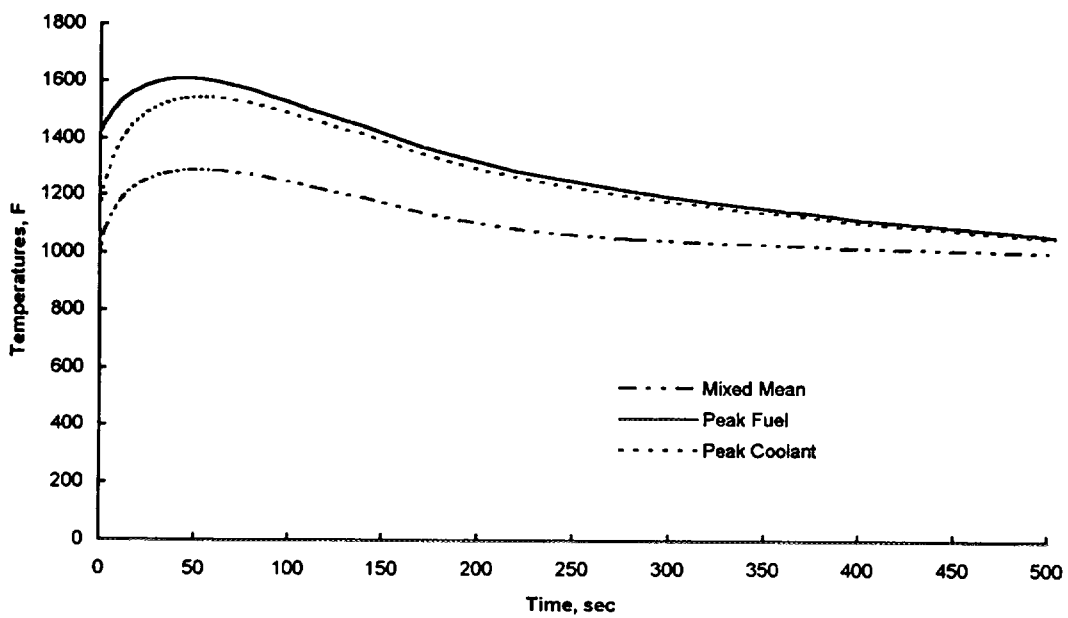
Figure 10. Temperatures During ULOF for Pu Core
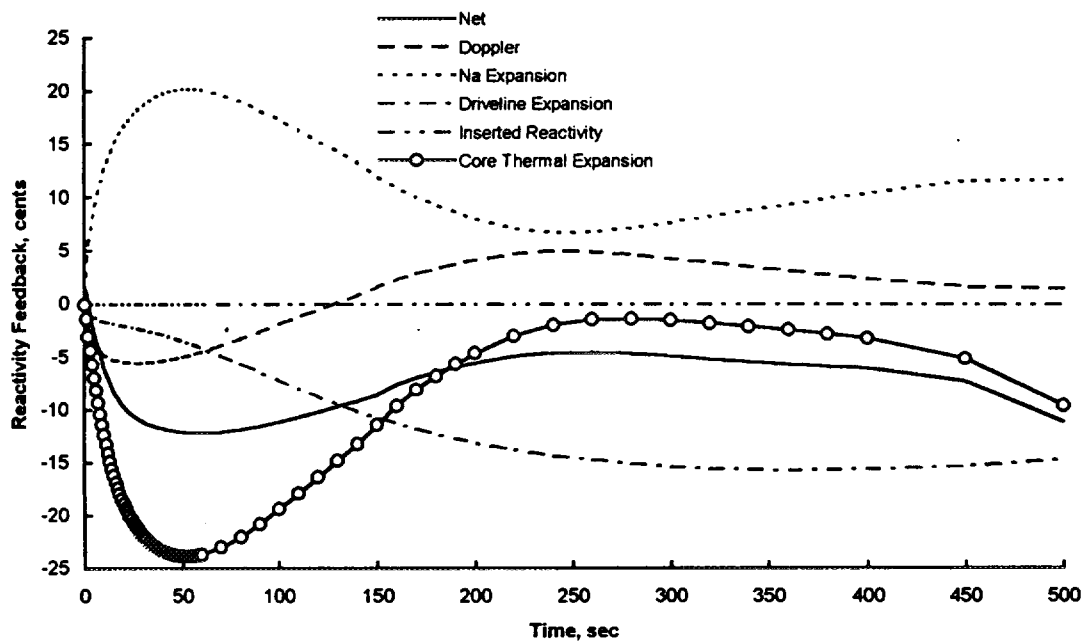

Figure 11. Temperatures During ULOF for U Core

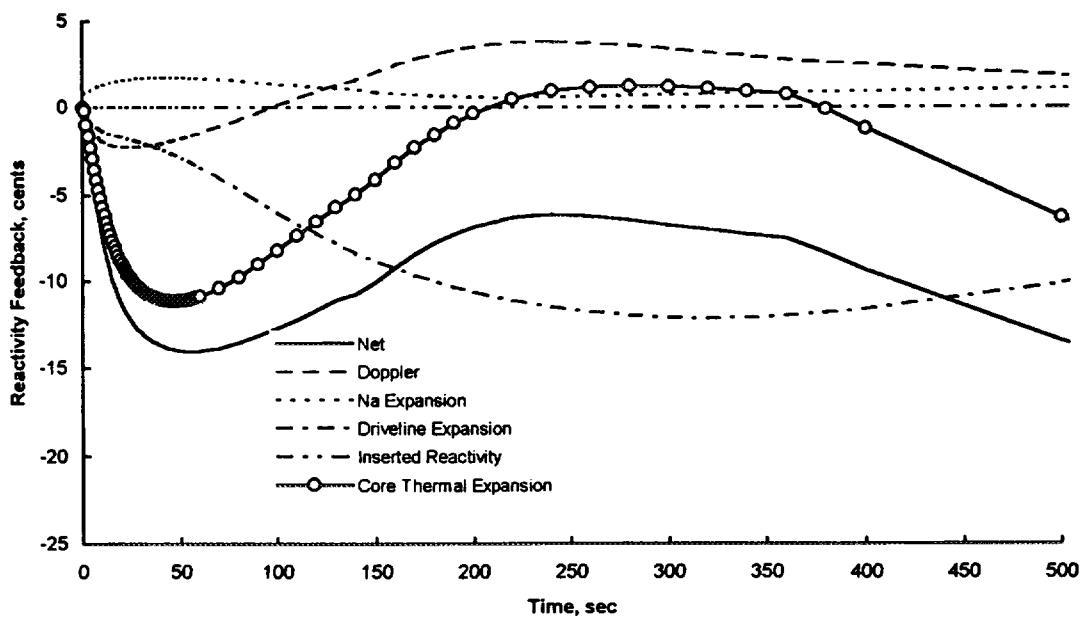Figure 12. Feedbacks During ULOF for Pu Core

Figure 13. Feedbacks During ULOF for U Core

209

Although the sodium expansion reactivity is much larger for U core and core thermal expansion reactivity is more negative for the Pu core, net reactivity is always negative for both Pu and U cores.

The Pu and U cores which have been analyzed do not have GEMs. Adoption of GEMs in the core would increase the safety margin for the loss of primary flow events by the rapid introduction of a large negative GEM worth with the primary flow coastdown.

Unprotected Loss of Intermediate Flow

This event starts with a sudden loss of the normal heat sink by a stoppage of the intermediate sodium flow. The primary pumps are assumed to continue operating at rated conditions until tripped by the TSS. Although this event would normally be terminated by a scram due to high primary cold leg temperature, it is assumed that the RPS fails to detect the over-temperature. Current analysis of ULOHS event is performed without RVACS model, which limits the analysis to early stages of the event.

The core power decreases due to the negative reactivity feedback of the core caused by the increase in core temperatures for both Pu and U cores.

Peak coolant and fuel temperatures maintain steady-state temperatures initially, and then decrease slightly due to the reduced core power. The core temperatures increase rapidly to peak temperatures with the trip of primary pumps by the TSS.

Most dominant reactivity feedback effects are due to sodium expansion and core thermal expansion. Net reactivity is always negative due to the larger contribution from the core thermal expansion.

It should be noted that current analyses mainly focus on reactivity feedback effects at an early stage of the accident. The analysis of long-term performance is necessary with the model for RVACS heat removal.

6. CONCLUSIONS

The safety design of KALIMER emphasizes accident prevention by using passive and natural processes, which can be accomplished by the utilization of inherent safety features for the accommodation of unprotected ATWS events without operator action, and without the support of active shutdown, shutdown heat removal, or any automatic system. Low probability and amount of accidental radiation releases for KALIMER beyond the limits of the site boundary eliminates the need for detailed offsite evacuation plan.

Passive safety features of the KALIMER design include the RVACS and the ACS for the assurance of safety-grade and normal decay heat removal, respectively. KALIMER core is fueled with metallic fuel which has enhanced safety characteristics

210

with negative feedback effects, and the detailed core design will be largely governed by passive safety and reactivity control issues. Features which are now considered include the SASS for passive reactor shutdown, rod stops which limit the reactivity addition during rod withdrawal event, and GEMs for the rapid introduction of negative reactivities during loss of flow event. Seismic isolation of the reactor and steam generator would also increase safety margins.

Improvement of the KALIMER design and assurance of the enhanced safety can be achieved by the preliminary evaluation of ATWS performance of KALIMER core options from the initial concept study phase. Results show that the temperature limits are met with margins for Pu and U cores whose performance would improve with a core design optimization and the introduction of passive features such as RVACS and GEMs.

ACKNOWLEDGEMENTS

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ELEVATION CONFIGURATION OF KALIMER INTERMEDIATE SYSTEM

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Abstract

KALIMER the first LMR in Korea is being developed in KAERI and the features related to configuring the elevation of the intermediate system for KALIMER are investigated. The relationship between elevation and natural circulation was analyzed for decay heat removal capability assurance. The capacities of the dedicated decay heat removal systems in KALIMER, PCVCS and SGBCS are evaluated. Based on the investigation, the idea on the role of each decay heat removal system of KALIMER is set up and discussion on mitigation of a sodium accident was made with regard to the intermediate system elevation configuration.

1. Introduction

Currently 16 nuclear power plants are under operation or construction in Korea and they account for almost a half of the electricity generation in Korea. The level of the nuclear portion in the electricity generation is expected to continue in future, too. The situation necessitates development of a nuclear reactor which can produce electricity more economically with less nuclear waste. A liquid metal reactor (LMR) can be a highly promising candidate of the solution to that necessity since a LMR makes it possible to utilize the energy resources of spent fuel from a PWR on a commercial scale. 100 % of the uranium consumed in Korea is imported. The utilization of spent fuel does not only help in lowering the fuel cost but also the spent fuel storage load from the PWR plants can be significantly reduced. This is an especially important feature in Korea, where it is highly difficult to find sites suitable for nuclear waste storage because of the high population density of the nation. For the plant economy, a LMR plant has a positive promising feature, too. Unfortunately LMR plants of past and current operation unfortunately was /is not economically competitive against other alternatives.

Based on this situation in Korea, the Korean government has launched a Liquid metal reactor development program as one of its long research plan activities. The top tier targets of the program are to develop a basic design by 2001 and construct a prototype or demonstration reactor by 2011. Recently a reference reactor design concept was formulated as a starting point for the rector concept development and the reactor to be developed was named KALIMER (Korea Advanced Liquid Metal Reactor). Some of the main features relevant to this study are as follows. A Pool base primary system, RVACS(Reactor Vessel Auxiliary Cooling System), SGACS(Steam Generator Auxiliary Cooling System), configuration of two IHX's and two steam generators . The thermal capacity of the core is 900 MWth.

For refining and specifying the details of the KALIMER systems, basic features of its systems need to be studied. In general arrangement of equipment, the horizontal configuration is mainly done by space consideration but the vertical configuration, that is, elevation configuration requires additional consideration . Elevation configuration is related to plant performance and safety since it gives effects on natural circulation and on mitigation at sodium water reaction and piping rupture events. Work is currently being done on the elevation configuration and it covers analysis on the characteristics of decay heat removal devices and natural circulation effects on decay heat removal. Based on the analyzed characteristics, decay heat removal schemes for various plant modes are devised. For the analysis, a fast running system analysis code LSYS[1] is being developed. This paper describes the interim results of the work at KAERI and deals with the foresaid subjects and practical relationship between intermediate system elevation configuration and natural circulation. Also some consideration was made on mitigation of a sodium water interaction accident with regard to the elevation configuration.

2. Elevation configuration and Decay Heat Removal

a. Natural Circulation

KALIMER has three methods under consideration for decay heat removal. One is the normal method using feedwater and steam systems. The others are RVACS and SGACS. The normal and SGACS methods utilizes the steam generator as the heat exchanging device and the decay heat from the core needs to be transported to a steam generator even at a loss of the coolant pumps in the primary and intermediate systems. The heat transport is achieved by the natural circulation process and the circulation is a function of elevation difference. To find the practical meaning of the natural circulation and to set up the basic strategy on decay heat removal scheme, the qualitative relation between them are analyzed using quilibrium state relationships and typical LMR system data.

At equilibrium, the flow rate and heat removal are related by Eq.(1)

$$\dot{m}C_p\Delta T = Q \qquad (1)$$

And the flow rate, system flow resistance, and pressure drop can be described by Eq.(2)

$$\Delta p = \frac{k}{2}\rho v^2 \qquad (2)$$

In these equations, $\dot{m}$, C_p, and k epresent respectively system mass flow rate, specific heat, and system pressure loss coefficient.

The system pressure drop needs to be balanced with the head developed by the density difference. The head is expressed as Eq.(3) and the density change on fluid temperature change is modeled as Eq.(4)

$$\Delta p = -(\rho g h)_H + (\rho g h)_L \qquad (3)$$

$$\rho = \rho_o + a(T - T_o) \qquad (4)$$

Here, the subscripts H and L respectively signify the property of a high and a low temperature path. Rearranging the equations brings out the relation Eq.(5)

$$h = -\left(\frac{k}{2A^2}\right)\frac{1}{\rho}\frac{Q^2}{C_p^2}\frac{1}{\Delta T^3}\frac{1}{a}\frac{1}{g} \qquad (5)$$

The parameter group in the parentheses of Eq.(5) represents the system flow resistance characteristics. From Eq.(5) it is deduced that the required elevation difference is proportional to the square of core power, that is, core decay heat in this paper. Plugging system data into Eq.(5) produces the approximate minimum elevation difference between the density centers of a hot and a cold path. When calculation is made for the time of 1 % core decay power, the required elevation difference turns out to be a value in the range of a couple of centimeters to a couple of 10 centimeters for a typical LMR plant including KALIMER.

Figure 1 shows the trend of the core power and primary and secondary system flow rates of a preliminary KALIMER design at the event of loss of power supply to coolant pumps. In the event analysis, the pumps were modeled as having the flow halving time of 5 seconds. The flow rate during the initial phase is formed by the pump inertia and then natural circulation becomes dominant. In Fig. 1 the switch over is made at around 50 seconds . The corresponding core power is about 4% and the required minimum elevation difference becomes a value in the range of a meter. In an actual case the requirement is much less severe especially for a pool type reactor since a pool type reactor such as KALIMER has large thermal inertia. The elevation effects will be discussed further in a later section.

b. RVACS

RVACS (Reactor Vessel Auxiliary Cooling System) removes the residual heat in the system by the natural circulation of air around the containment vessel which surrounds the reactor vessel. The air flows through the gap channel between the containment vessel and air collector cylinder which is located outside the containment vessel. The working principle is same as that of RVACS in PRISM[2]. To define the function of RVACS in KALIMER, the characteristics of RVACS performance were analyzed by a routine of LSYS. A reference design condition was set and effects of functional parameter changes to RVACS heat removal capability was checked. Fig. 2, 3 and 4 are for geometry parameter effects. There is an optimum gap spacing of the air flow passage because of the two competing effects of the buoyancy head buildup and the head loss by friction. The optimum gap was in the order of 30 cm. Fig. 5 and 6 are for property effects. Very little effect was found for the vessel surface emissivity. Fig. 7 summarizes the investigation results. The heat removal capacity is expected to be 3 to 4 MW for KALIMER.
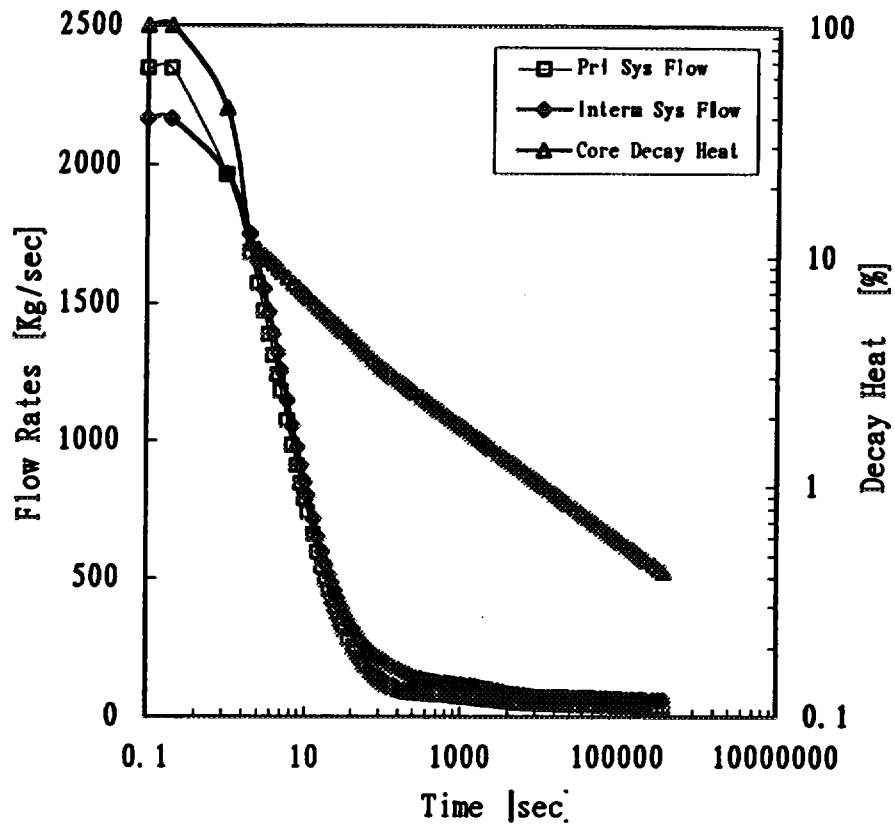
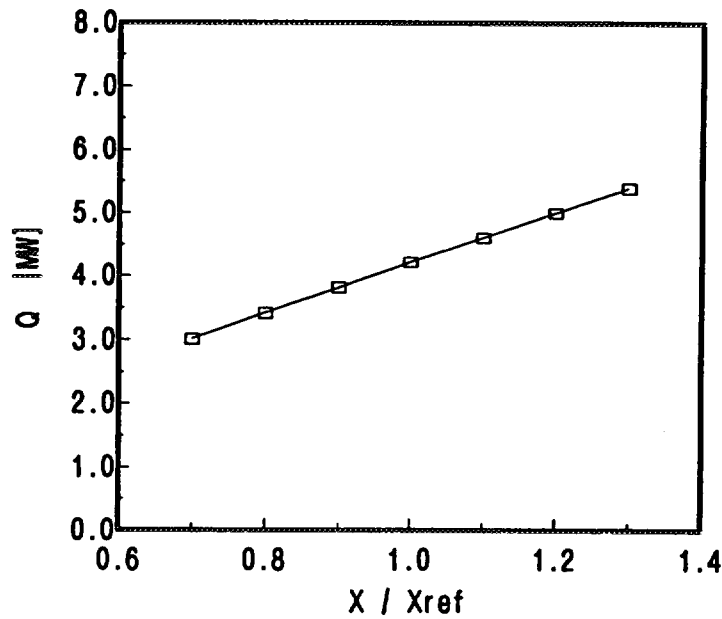Fig. 1   System Flow Rate Coastdown and
Natural Circulation Flow Rate

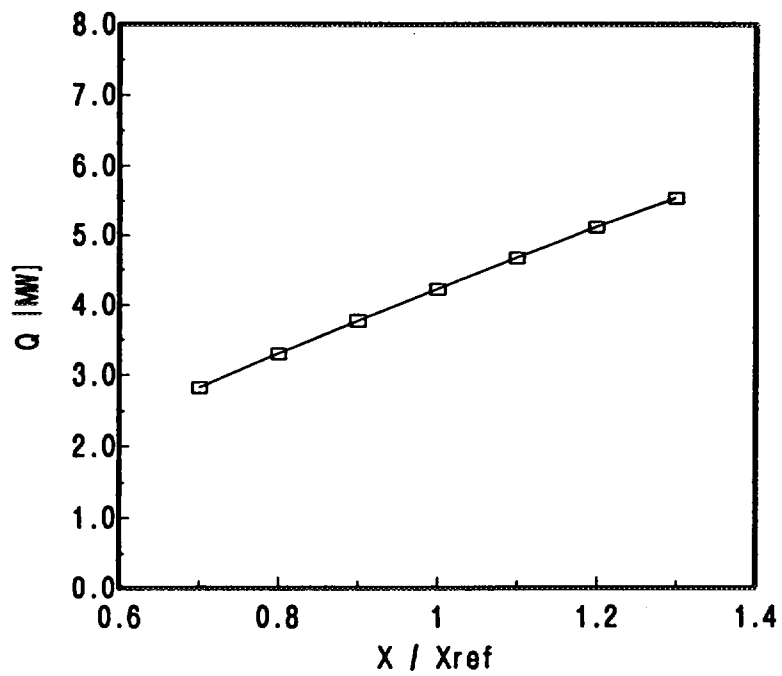Fig. 2   Effects of Reactor Vessel Diameter
on RVACS Capability

Fig. 3 Effects of Reactor Vessel Length
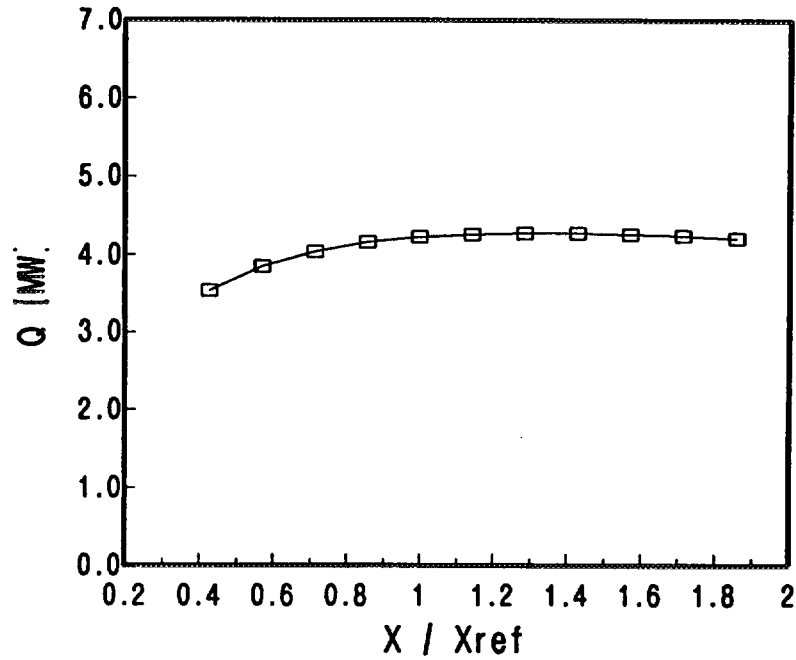on RVACS Capability

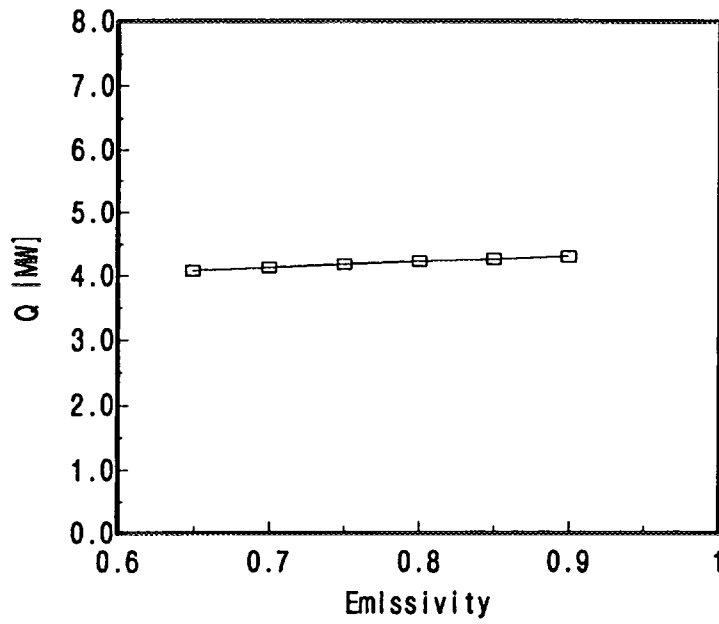Fig. 4 Effects of RVACS Channel Gap Size
on RVACS Capability

Fig. 5 Effexts of Emissivity of Reactor Vessel
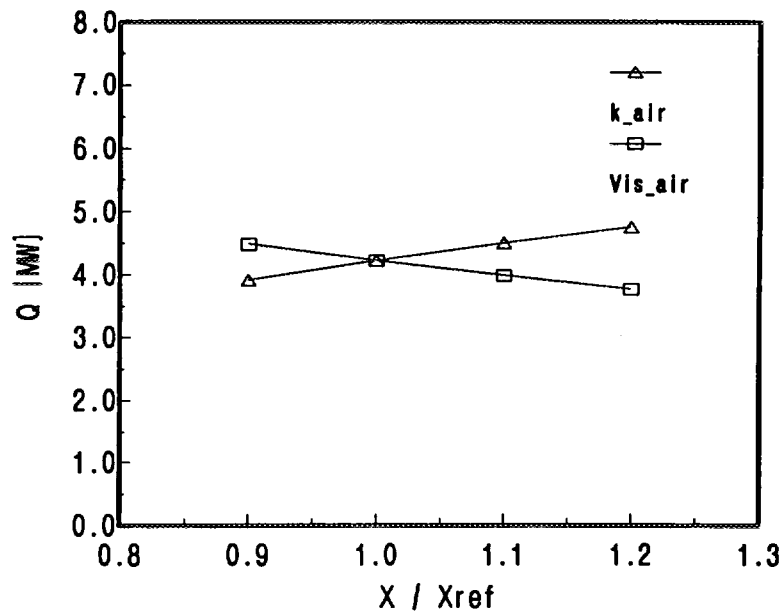
and Containment Vessel on RVACS Capability

Fig. 6 Effect of Air Viscosity and Conductivity
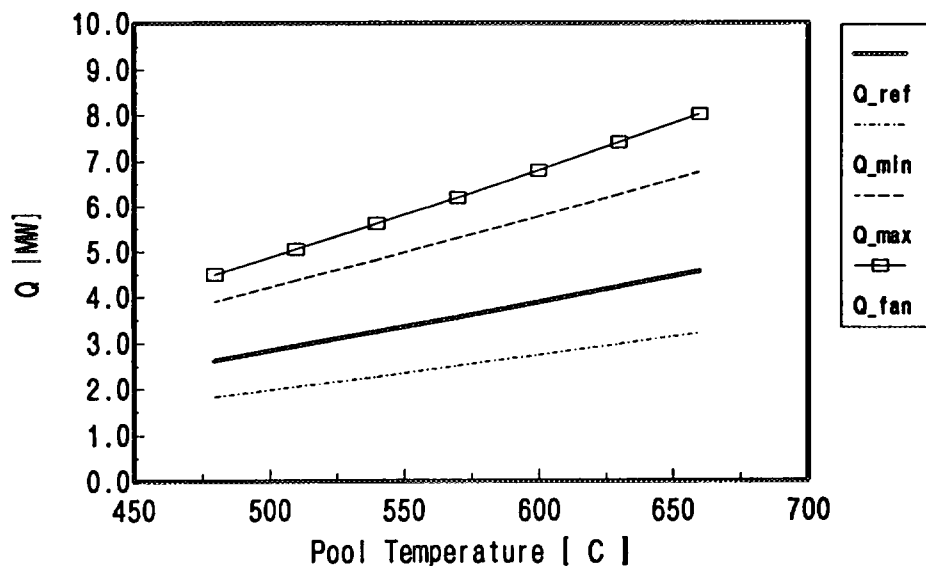on RVACS Capability

Fig. 7 Heat Removal Capability of RVACS

c. SGACS

SGACS (Steam Generator Auxiliary Cooling System) is also an air cooling system but it works on the shell of a steam generator. Its physics is similar to that of RVACS. In KALIMER, the hot intermediate system sodium is flows through the shell side in the steam generator and this system utilizes the hot temperature of the SG shell. To define its role in KALIMER, an analysis similar to the work for RVACS was made to SGACS and the results are shown in the figures from Fig. 8 to Fig. 13. The summary results in Fig. 13 show the total heat removal capability of SGACS will be about 1.0 MW to 2.0 MW for KALIMER.

d. Overall consideration

The effect of the intermediate system elevation configuration to decay heat removal was checked for a reactor trip event using LSYS. Power supply loss to coolant pumps and reactor trip are made. The primary and intermediate system flows coast down during the initial period and then natural circulation carries out the required heat transfer from the core to the steam generators. In this analysis the feed water flow rate is controlled so that the steam quality from the steam generator is in a specified control band. The analysis was made for two different elevation configurations in the intermediate system. In one case, the geometric center of the steam generator was 1.0m above the geometric center of IHX. In the other case the center difference was 3.5m. The results are shown in Fig. 14. In both cases the core is well cools down as expected from the previous simple analysis. It means the requirement to the elevation configuration is not serious for removing decay heat by natural circulation.
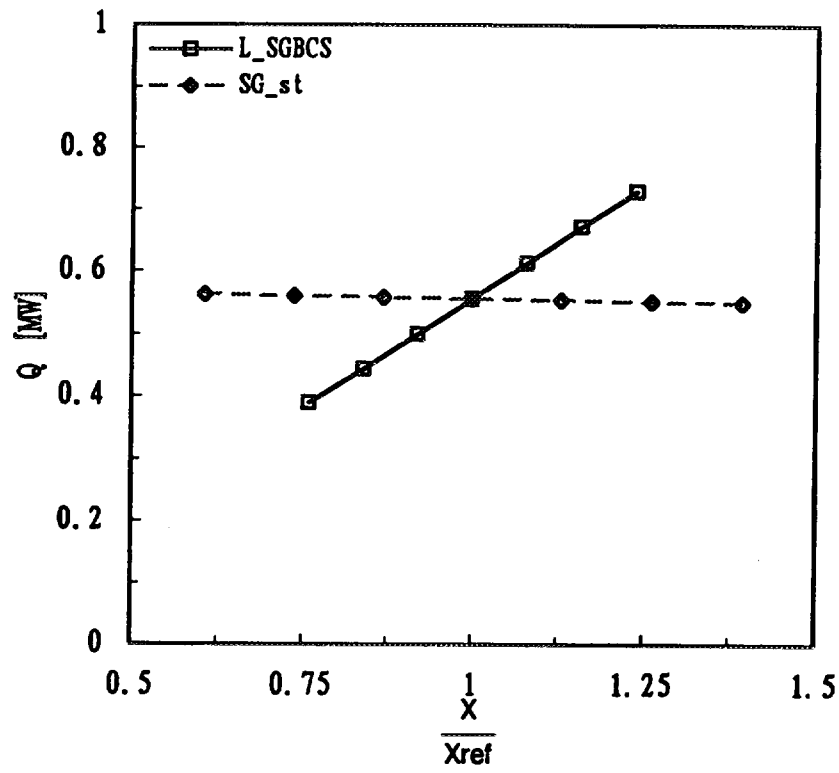
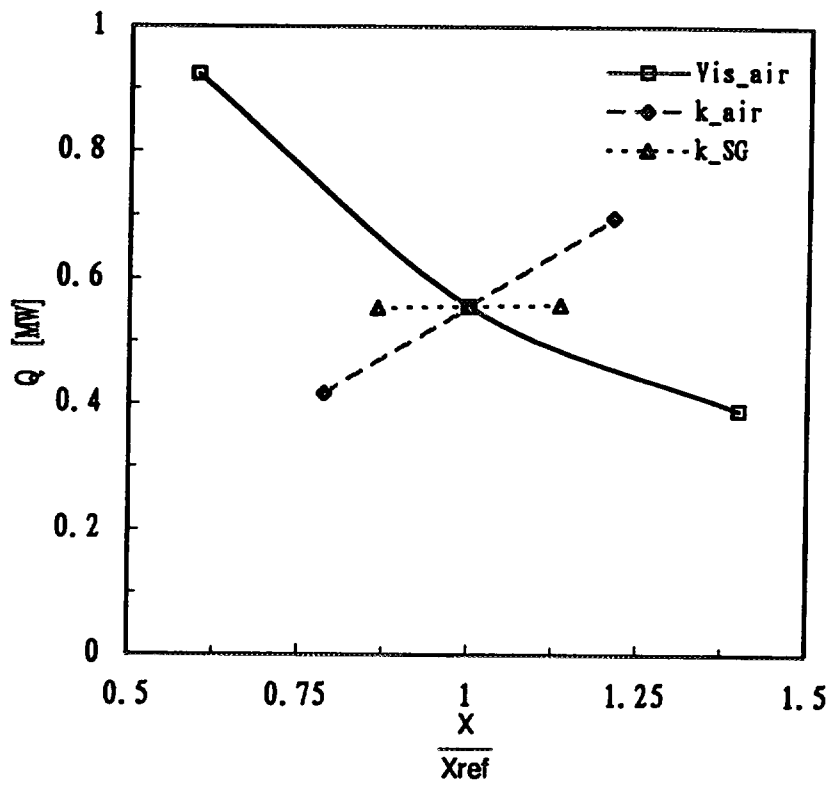Fig. 8 Effects of ACS Length and SG Shell Thickness

Fig. 9 Effects of Air Viscosity and Conductivity, and SG Conductivity
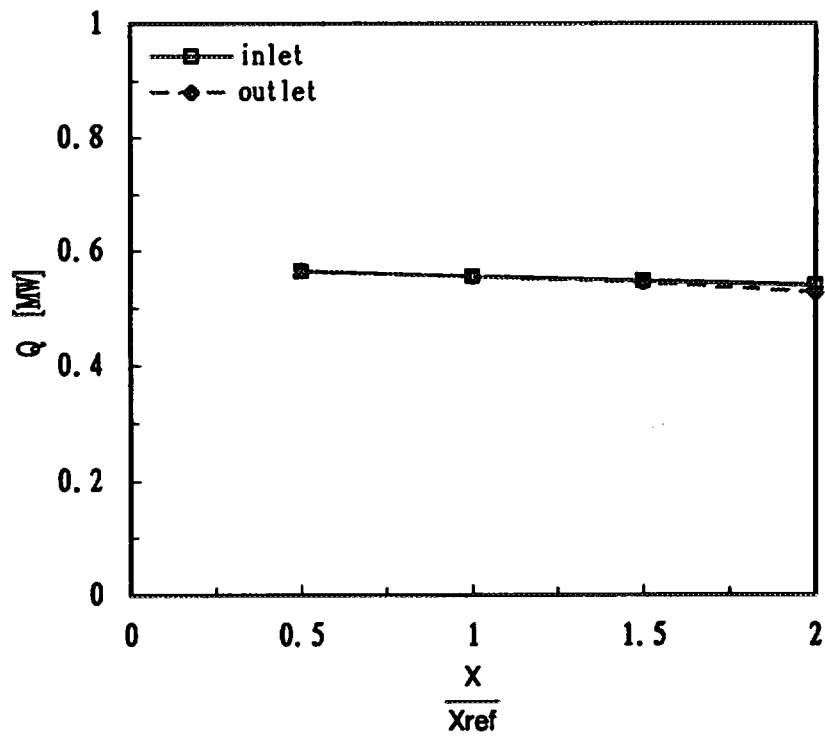
Fig. 10　Effects of Inlet and Outlet Form
Losses
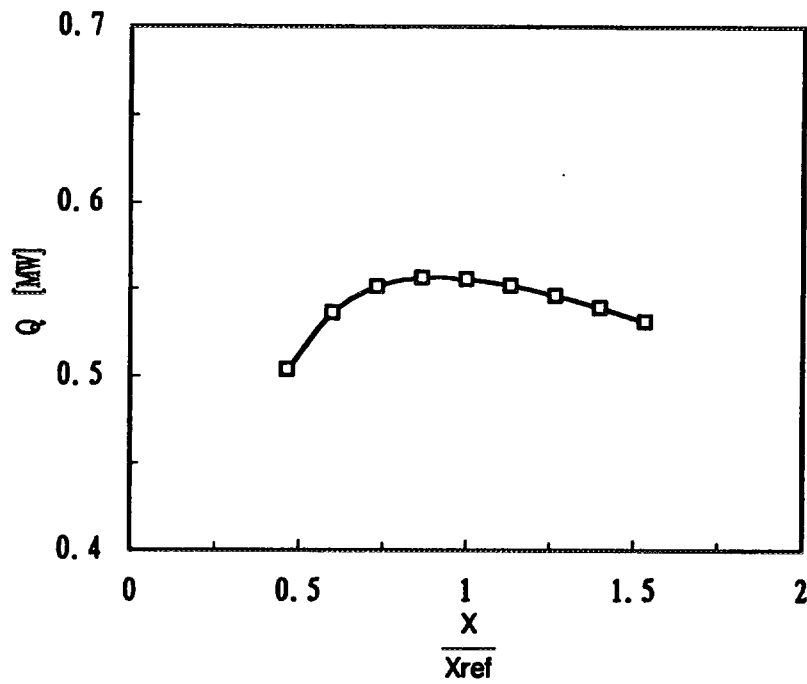
Fig. 11　Effects of ACS Channel Gap
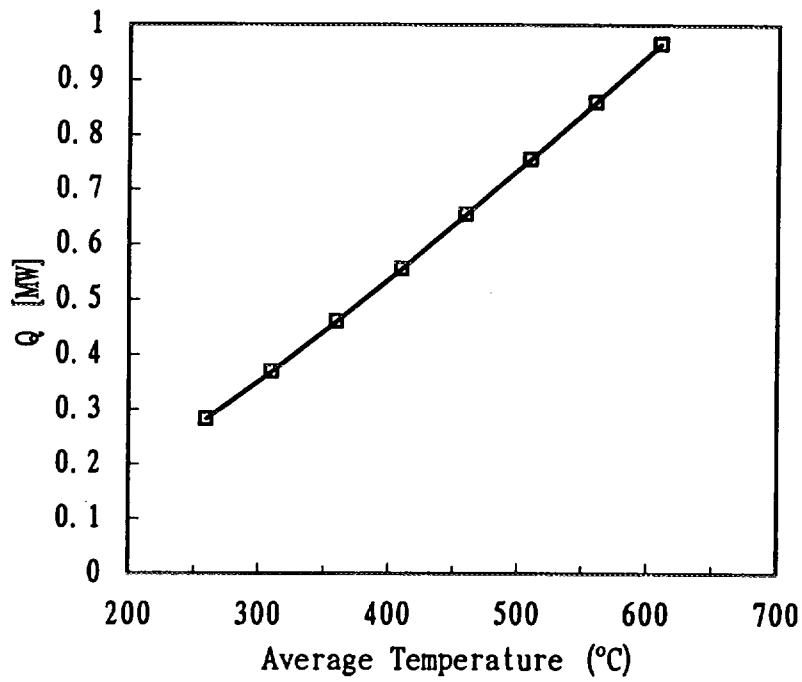
Fig. 12 Effects of SG Sodium Average
Temperature
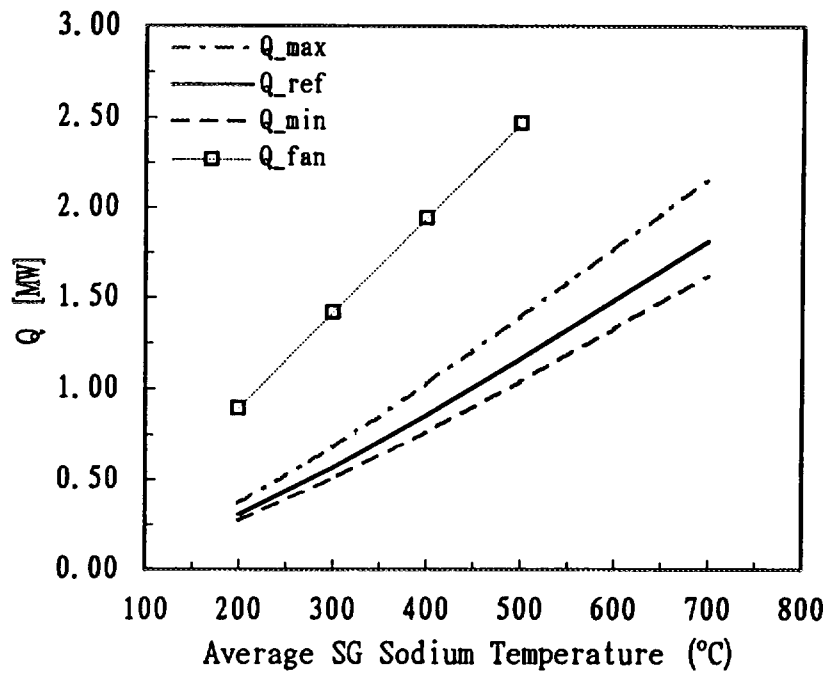
Fig. 13 Heat Removal Capability of ACS Cooling
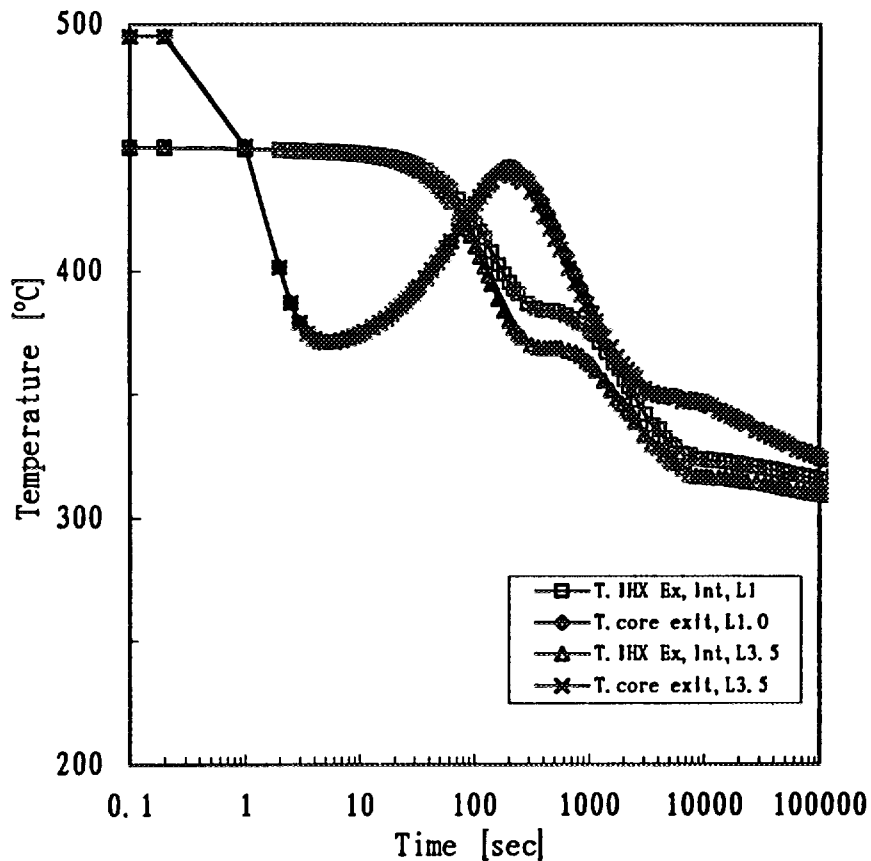
Fig. 14 Core Temperature Trends at two different
Intermediate Elevation Configurations

Of the two dedicated decay heat removal systems considered in KALIMER, SGACS has a very low capacity of 0.1% to 0.2% of the normal core power when a fan is not used. The capacity is equivalent to the decay heat at the time more than a week after shutdown. When a fan is used, the capacity can become about 0.4% of the normal core power, which corresponds to the decay power at a couple of days after reactor shutdown. With fan operation, it can be used for maintenance of the steam or feedwater system.

RVACS has much larger heat removal capability compared to SGACS and its capacity can be 0.3 % to 0.5% of the normal core power. Its heat removal capacity   becomes larger than the core power one to three days after reactor shutdown..

Decay heat removal can be classified into three phases excluding the normal decay heat removal where the coolant pumps and the feedwater/steam system are used. The first phase is the decay heat removal by   the coolant coast down and steam discharge. The second one is the   next phase of the flow coast down. The third one is for long term cooling. Based on the analysis results, the following strategy is being considered for decay heat removal of KALIMER. Because of the highly reliable operation

223

feature of RVACS, RVACS work as a safety grade system and the decay heat removal by the natural circulation is used as a non-safety grade supporting scheme. For the maintenance work on the steam or feedwater system, SGACS be used.

3. View points other than decay heat removal

In configuring the intermediate system elevation of KALIMER, consideration is being also made on the aspects other than decay heat removal. They are on the mitigation and reduction of the possibility of sodium related accidents. To reduce the possibility and amount of sodium discharge, the intermediate system piping length needs to be minimized. However, mitigation of a sodium-water interaction event requires a certain length of the hot and cold legs of the intermediate system to be installed upward. This length can vary substantially depending of the internal structure of the steam generator. A vertical length of 10m to 20m may be required for KALIMER.

Another point considered is on the construction cost related to the vertical location of the steam generators. When the steam generator is located at a relatively low elevation in the containment building, there may not be a sufficient space for the drain tank required for draining sodium from a steam generator at an event of steam generator tube break and the base of the containment building needs to be digged further below, which is generally considered expensive.

4. Conclusion

KALIMER is being developed in KAERI. As one of the activities for the development work, the features on the elevation configuration of the KALIMER intermediate system are being investigated and also a fast running system code LSYS is being developed. The results of the work are ı 1) The elevation difference between IHX and SG is not severely required for decay heat removal. 2) he expected range of the heat removal capacity of PCVCS and SGBCS has been analyzed. 3) A skeleton of the basic strategy for decay heat removal of KALIMER has been set up.

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STUDY AND CHOICE OF MAIN CHARACTERISTICS OF FAST REACTOR - EFFECTIVE MINOR ACTINIDE BURNER

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Abstract

This paper presents the principal design and performance data of advanced fast power reactor core for plutonium and actinides burning. Some information concerning the Russian programme of plutonium utilization are also presented.

INTRODUCTION

Basic feature of fast reactors-ability to transfer nuclear power into self-supplying regime with nuclear fuel -has been scientifically justified and technically verified at demonstration NPPs and nuclear fuel cycle enterprises.

But at the present stage of nuclear power development, when sufficient uranium reserves exist, this feature of fast reactors is not topical and could be called for in some decades.

At the same time the basic features of fast reactors and the experience gained in designing and operation of NPPs with a fast reactors allow to state that NPPs of this type can be effectively used for combined and safe solution of power and ecological problems using accumulated plutonium and decreasing quantity of radiotoxical products of nuclear fuel cycle -burning of minor actinides.

When designing nuclear power installation with above mentioned functions the following should be provided :
- competitiveness with light water and other reactor types in electricity cost,
- high safety level without evacuation of population from adjacent to the NPP installed power utilization (with maximum use of proved technical solutions).

Below some ideas on choosing of major parameters of fast reactor - effective actinides burner are presented

1. UNIT POWER

Reactor power choosing is one the crucial issues of designing.

Unbiased decision of this issue can be obtained only by combined accounting for the results of technical-economical optimization of the basic technical solutions, the design and operation experience of operating NPPs, the applicability of the NPP being developed for the power network.

Power range, in which the practical realization of a power unit with the above mentioned target is possible, is quite wide - from 200 MW(e) to 1600 MW(e).

In this case the following factors are accounted for.

The most important factor is reactor self-protection, which is characterized by the ability to prevent destruction or melting of the core in severe beyond design accidents due to inherent safety features (SVRE≤0), emergency cooling under conditions of sodium boiling) and using of passive means of action on reactivity. The analysis shows that for traditional cores with the power increase, all other factors being the same, the attainment of equal level of self-protection is more difficult (Fig.1). The question whether this tendency would be conserved foe special cores require a more detailed analysis.
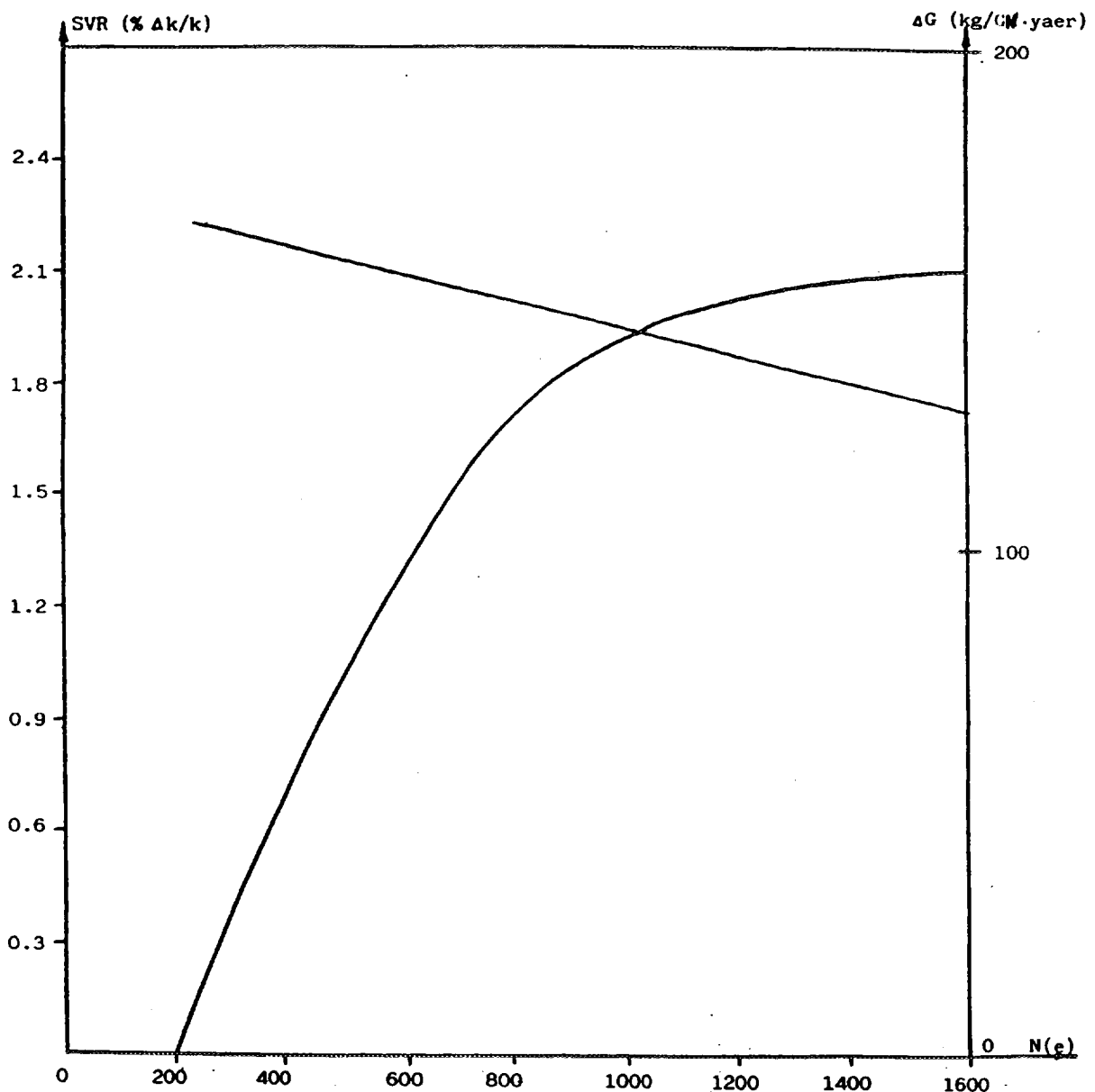

Fig.1. Dependence of SVR and plutonium burning efficiency on reactor power

Reactor power and core volume increase, other parameters being the same, is accompanied by decrease of the neutron leakage and the spectra softening. As a result fuel enrichment and actinide burning effectiveness decrease (Fig.1). Assessments show that increase of reactor power from 800 MW(e) to 1600 MW(e) with the same approach to the core design (decreasing of fuel volume fraction and use of absorbing assemblies) would lead to decrease of plutonium specific burning out effectiveness by 15 %. Average fission cross section of threshold nuclides (Pu-240, Np, Am etc.) also decrease by 10-15%.

2. CHOOSING OF CORE PARAMETERS

2.1. Possible ways of realization of reactor core.

Analysis of fast reactor possibilities for effective actinide burning shows that oxide fuel wholly corresponds to this goal. Oxide fuel has the least breeding among others, more dense fuels types, have been much studied and mastered. More effective for burning fuel -without uranium-238 with inert matrix - still requires long-term and comprehensive studies and can not be laid to the basis of the project.

Preliminary investigations of various core types have shown that it is necessary to renounce breeder blanket and to increase the plutonium enrichment up to 45 %. The main problem in search of optimal core configuration consists in searching of the most correct ways of fuel enrichment increasing corresponding to various requirements in reactor physics and safety.

The numerous studies have shown that the plutonium enrichment increasing can be attained by two ways [1] :
- by introduction of absorbing materials into the core,
- by reducing of fuel volume fraction.

Some ways of fast reactor core realization for effective actinide burning can be demonstrated by the example of three core types as applied to the BN-800 reactor project.

a) Core in design dimensions (565 SAs). Enrichment increase is obtained by introducing absorber assemblies on the base of natural boron carbide and some decreasing of fuel volume fraction using fuel pins will less diameter (6.0*0.4 mm). Absorbing assemblies are located uniformly in the centers of 48 modules, consisting of 7 SAs to which subzones of middle (MEZ) and high (HEZ) enrichment are divided (Fig.2). The power level is hold due to core height increase up to 105-110 cm. Maximum plutonium burning effectiveness is characterized by value 342 kg/year (61.1 kg/TW*h).

b) Core is extended one SA row in the radial blanket (655 SAs total, Fig.3). Enrichment increase is obtained only by decrease of fuel volume fraction and introducing of absorbing blankets from natural boron carbide. Maximum effectiveness of plutonium burning at fuel pin diameter 6.0*0.5 mm with increased central hole is 400 kg/year (71.4 kg/TW*h)
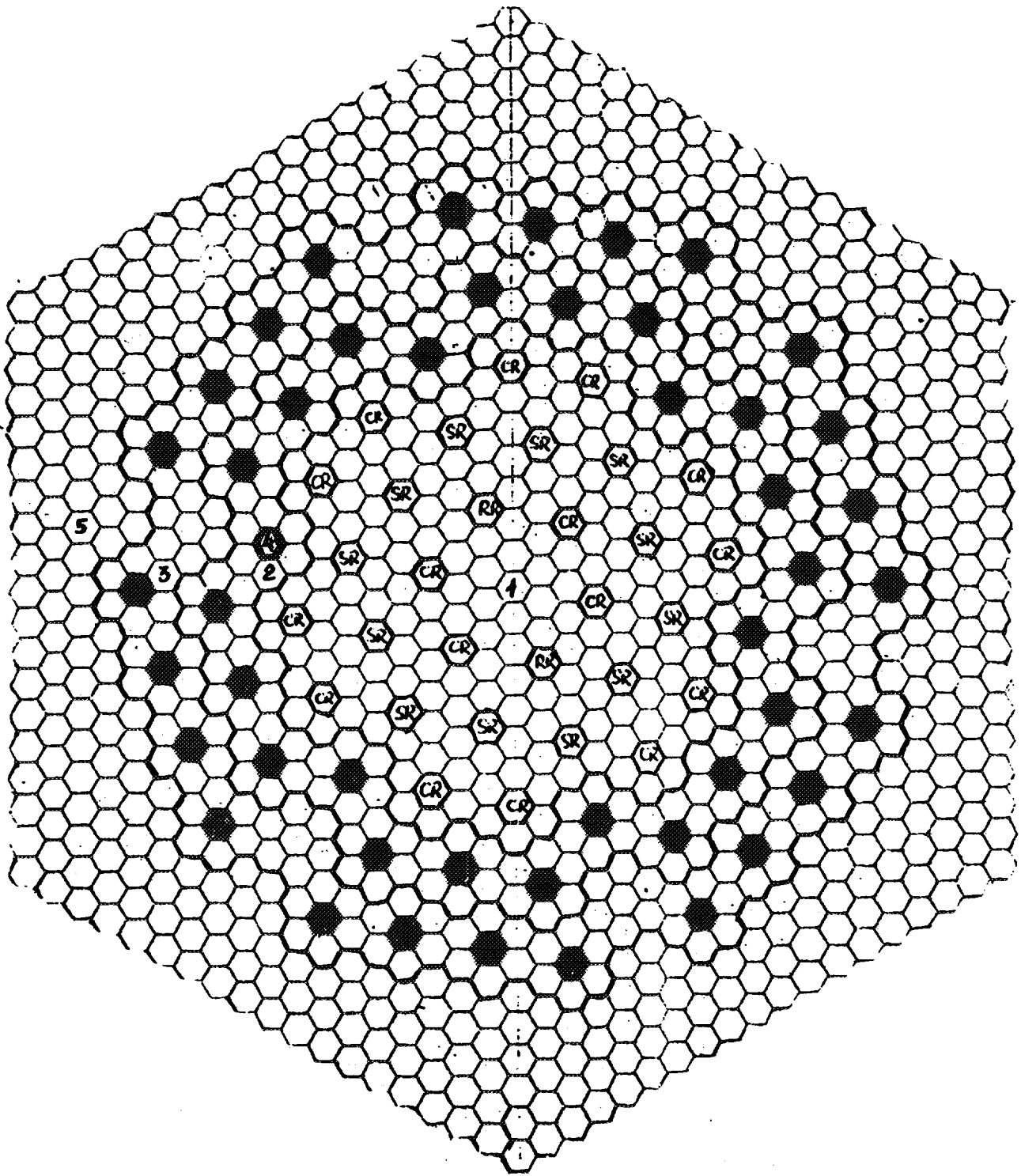
Fig.2 Core Layout with Absorber Subassemblies

1-LEZ SA   5- Steel blanket
2-MEZ SA   SR- safety rod
3-HEZ SA   CR- compensating rod
4-Absorber subassembly   RR- regulating rod
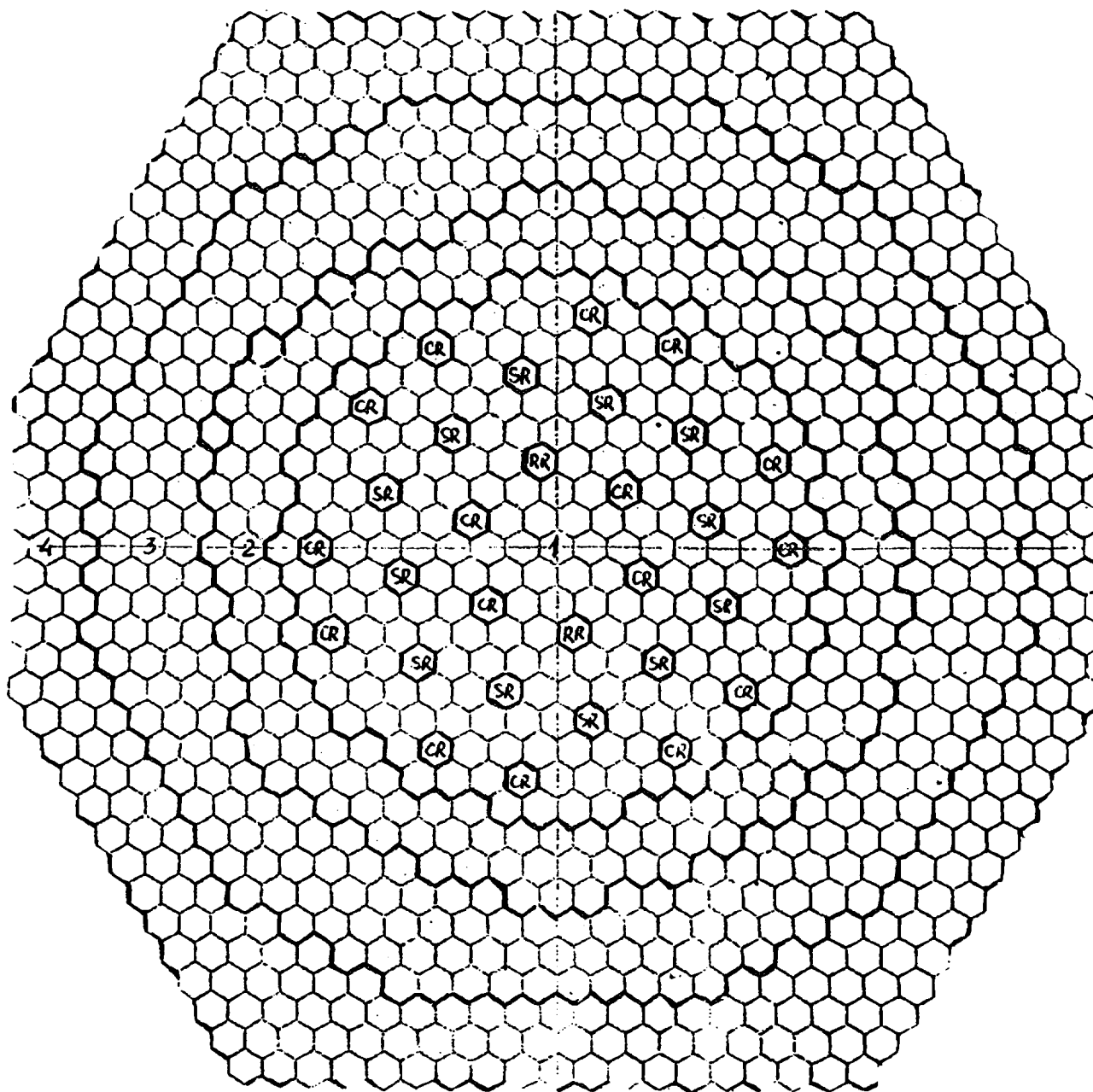
Fig.3 Core Layout with Increased Radius and Fuel Pins Diameter
6.0*0.5 mm

1- LEZ SA
2- MEZ SA   SR- Safety rod
3- HEZ SA   CR- Compensating rod
4- Absorber blanket   RR- Regulating rod

c) The core is extend one as far as outer boundary of the in-reactor storage (IRS) and totals 900 SAs. Maximum effectiveness of plutonium burning is 490 kg/year (87.0 kg/TW*h), but maximum fuel enrichment begins to exceed noticeably the prescribed standard 45 %.

Detailed date on the studies carried out are presented in Tables 1, 2, 3.

2.2 Results analysis

Analysis of the results obtained and possible technical decisions allows the following conclusions.

a) Enrichment increase through introduction of absorbing SAs into the core allows to attain the sufficiently high plutonium burning characteristics. However, such way have some disadvantages, the main of them is substantial non-flattering of power distribution, connected with great amount of absorber sub-assemblies. Some difficulties are connected with the zero sodium void effect achieving because of substantial core height increasing.

The last problem can be solved by using of absorbing blankets instead of steel ones, although such cores require the additional investigations.

b) Enrichment increase by means of fuel volume fraction decreasing give the good plutonium burning characteristics. The disadvantage of such way is core dimensions increasing .

Using the absorbing blankets from boron carbide substantially decreases the SVRE value in the core, opening the additional possibilities for the minor actinide burning.

Using of fuel pins with increased diameter of central hole in the fuel pellets requires to solve a number of technological problems, connected with fuel pellets manufacturing and problems of mechanical and radiation stability of such pellets.

Increasing the core dimensions on 90 sub-assemblies for the BN-800 reactor project will require the re-design of lower collector, but this problem is wholly solved.

c) More significant increase of SA number in the core (up to 900) allows even more to decrease the fuel volume fraction (on 50-70%) and increase effectiveness of plutonium burning up to 87 kg/TW*h.

However, the substantial exceeding of admissible plutonium enrichment limit (45%) in those cores, the necessity of substantial re-design of in-vessel structure make such way of core design not rational.

Proceeding from the above, the most optimal variant from various points of view is the core with 665 SAs and fuel pin with diameter 6.0*0.5 and central hole, which diameter is equal 0.5 of fuel pellet diameter. Such dimensions correspond to the fuel volume fraction $\varepsilon=0.2$.

Table 1

Core versions with absorber introduction

	Fuel pin 6.6*0.4 (ε_{fuel}=0.335)			Fuel pin 6.0*0.4 (ε_{fuel}=0.269)		
Number of SA	48	48	48	48	48	48
Absorber volume fraction, %	0	20	60	0	20	60
Number of fuel SAs in core	517	517	517	517	517	517
Core height, cm	105	110	110	105	110	110
Fuel enrichment by zones						
LEZ	19.4	17.7	19.2	23.2	21.2	23.0
MEZ	22.0	27.2	29.5	26.4	33.1	36.0
HEZ	27.0	35.9	38.0	32.3	41.4	45.0
Core SVRE %Δk/k	+2.10	+1.44	+1.41	+1.84	+1.34	+1.19
Quantity of burned plutonium,						
kg/yr	190	254	276	230	310	342
kg/yr GW(e)	238	318	345	288	388	428
kg/ TW*h	33.9	45.3	49.2	41.1	55.4	61.1

Table 2

Core versions without absorber with fuel pins 6.6*0.4

	Variant				
	1	2	3	4	5
SA number in core	565	565	655	900	900
Fuel pin number in SA	127	127	127	91	91
Fuel volume fraction	0.335	0.335	0.335	0.240	0.160
Core height, cm	100	100	80	85	130
Blanket material	steel	natural boron carbide	natural boron carbide	natural boron carbide	natural boron carbide
Fuel enrichment by zones					
LEZ	17.5	19.6	20.1	26.7	28.4
MEZ	19.7	22.1	22.7	29.7	31.6
HEZ	24.3	27.2	27.9	36.5	38.8
Core SVRE %Δk/k	+2.03	+0.87	+0.41	-0.61	+0.45
Quantity of burned plutonium,					
kg/yr	146	195	209	273	370
kg/yr GW(e)	183	244	261	342	465
kg/ TW*h	26.1	34.6	37.1	48.5	66.0

Table 3

Core versions without absorber with fuel pins 6.0*0.4 mm and 6.0*0.5 mm
and absorbing blankets from natural boron carbide

	Variant					
	1	2	3	4	5	6
SA number in core	655	655	655	900	900	900
Core height, cm	85	85	85	85	130	130
Fuel pin number in SA	127	127	127	91	61	61
Fuel pin diameter*cladding thickness, mm	6.0*0.4	6.0*0.4	6.0*0.5	6.0*0.4	6.0*0.4	6.0*0.5
Relation of central hole diameter to fuel pellet diameter	0.3	0.5	0.5	0.3	0.3	0.5
Fuel volume fraction	0.27	0.22	0.21	0.19	0.13	0.10
Fuel enrichment by zones LEZ	24.8	31.6	34.2	32.1	34.6	37.5
MEZ	28.1	35.8	38.7	36.4	39.2	42.5
HEZ	34.5	44.0	47.6	44.6	48.0	52.0
Core SVRE %Δk/k	-0.7	-1.3	-1.5	-3.0	-1.4	-1.6
Quantity of burned plutonium, kg/yr	332	371	400	412	457	489
kg/yr GW(e)	414	463	500	515	571	611
kg/ TW*h	58.9	66.2	71.4	73.2	81.3	87.2

3. THE MAIN PRINCIPLES OF PLUTONIUM   UTILIZATION PROGRAM IN RUSSIA.

Accumulated stocks of extracted power plutonium together with expected receiving significant amount of weapon-grade plutonium in result of disassembling of nuclear weapon requires the developing of certain strategy of plutonium handling including safety, economic, ecology issues and regime of unspreading.

Conception of MinAtom of plutonium handling is based on following main principles [2]:

- maximum using of accumulated experience of plutonium handling;
- reliable protection against diversion and uncontrolled utilization of plutonium ;
- acceptability with point of view of ecology and in interests of environment;
- possibility of using of existing basis for development of optimal fuel cycle with long-term perspective;

Now it has been considered three possible directions of plutonium utilization :

- using of PO "MAYAK" (radiochemical facility, facility for MOX fuel production, NPP with BN-800 reactor);
- using of existing reactors (BN-600, modern unit of VVER-1000);
- using VVER-1000 reactors planning to construct.

Comparative estimations of above-mentioned directions show that the first direction is most preferable for following reasons:

- in the directions № 1 it is realized the conception of closed nuclear - power center where it is more simple to solve the problem of prevention of plutonium using without sanctioning;
- the using only existing reactors (fast reactor BN-600 and 4 units of thermal reactors VVER-1000) doesn't allow to utilize wholly plutonium which is expected to receive;
- the development of nuclear power engineering in the third direction requires the introduction of the additional power units whose capacity should be in 3 times more than one in variant with fast reactors.

The characteristics of BN-800 fast reactor and reactor for Pu burning ("burner") on the base of BN-800- type reactor from point of view of utilization (transformation into the form of spent fuel) and burning (physical destruction) with simultaneous using of energy potential of plutonium are given in the Table 4.

The table 4 shows that the BN-800 type core modernized for effective plutonium utilization allows to expand significantly the possibilities of this reactor for states where it is necessary both utilization and burning of plutonium as nuclear material.

It is interesting in some extent to consider the possibilities of fast reactor work in the system with VVER-type reactors.

For study of thermal and fast reactors system work feasibility in closed fuel cycle we will consider the simplified scheme of this system given on the Fig. 4.

Plutonium together with minor actinides produced in two VVER is mixed with spent BN-800 reactor fuel, from which the fission products are removed, and comes again into the fast reactor core.

After multi-recycle according to the above-mentioned procedure in the fast reactor core the quasistationary isotopic composition of actinides is formed.

The isotopic compositions of actinides of unloading VVER fuel and quasistationary composition of ones after multi recycle in the fast reactor core are given in the Table 5.

Thus, in the isotopic composition of actinides (plutonium + minor actinides) the minor actinides fraction increases approximately in 1.5 times that corresponds to the content of minor actinides in the fresh fuel of fast reactor on the level of 3-4 %.
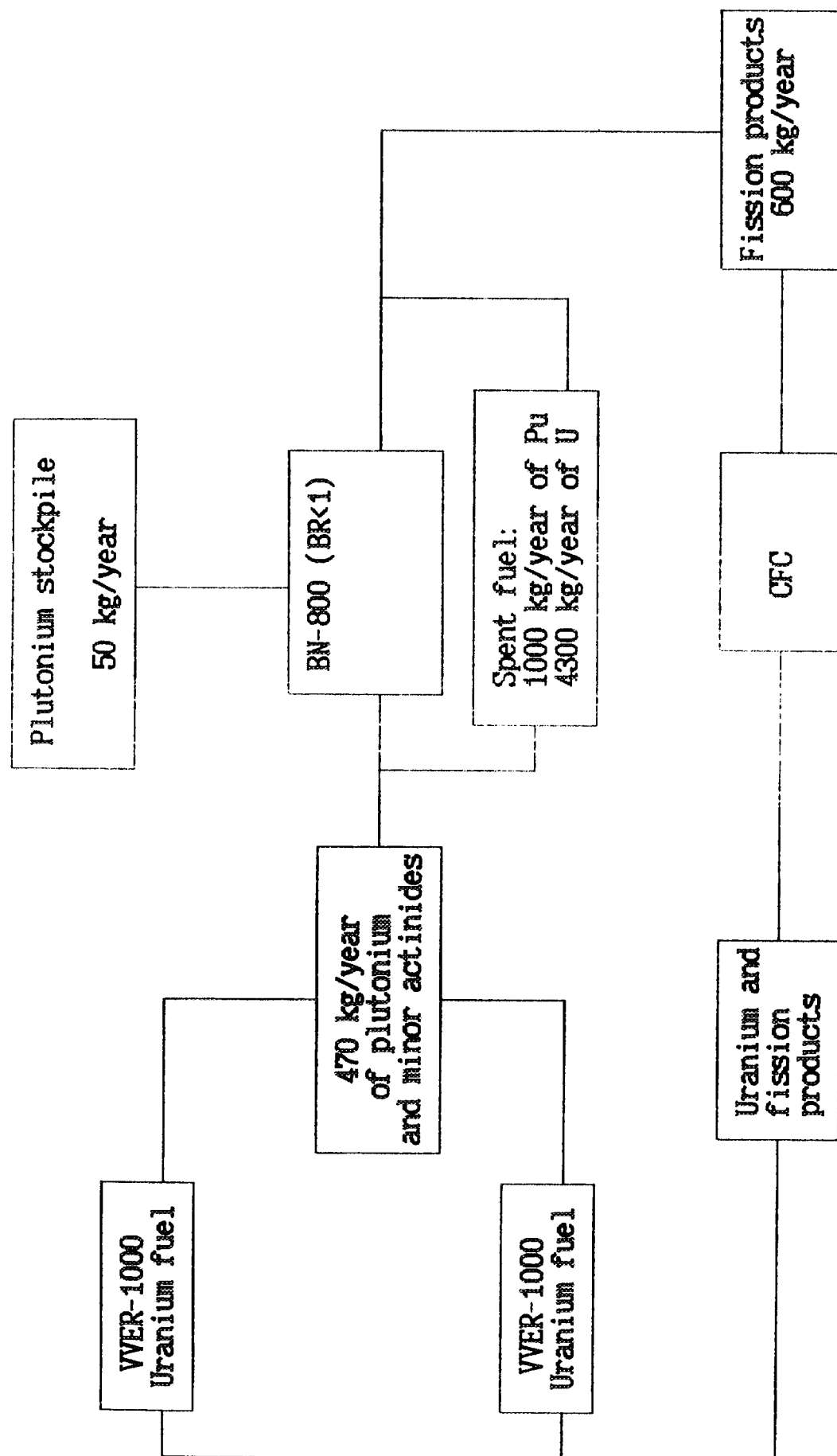
Fig. 4. Scheme of thermal and fast reactors system, working in closed fuel cycle.

The boxes in the figure contain the following text:

- Plutonium stockpile 50 kg/year
- BN-800 (BR<1)
- Spent fuel: 1000 kg/year of Pu 4300 kg/year of U
- Fission products 600 kg/year
- CFC
- 470 kg/year of plutonium and minor actinides
- VVER-1000 Uranium fuel
- VVER-1000 Uranium fuel
- Uranium and fission products

So in this scheme it is taken place the effective plutonium utilization as well as minor actinides. From point of view of the ecology it is obvious the efficiency of nuclear power engineering work in accordance with above-described principle.

Table 4

Characteristics of BN-800 reactor project and BN-800 reactor-burner

Parameter	BN-800	BN-800 'burner'
Pu loading in the reactor, kg/yr	1600	1400
Pu unloading from the reactor, kg/yr	1600	1000
Burnt Pu in the core, kg/yr	600	600
Pu produced in the blanket, kg/yr	150	0
Pu produced in the core, kg/yr	450	200

Table 5

Isotopic composition of actinides in discharged fuel of VVER and in the fuel of BN

Isotope	Isotopic composition of actinides	
	Unloading VVER fuel	Quasistationary composition
Pu-238	1.2	2.4
Pu-239	51.7	45.4
Pu-240	22.6	30.8
Pu-241	11.7	6.3
Pu-242	5.4	6.1
Np-237	3.0	1.6
Am-241	2.7	4.9
Am-242m	-	0.2
Am-243	1.1	1.7
Cm-244	0.4	0.6

CONCLUSION

1. The using of fast reactor in the nuclear power engineering system allows to solve power and ecological problems by means of effective plutonium and minor actinides utilization.

2. Analysis shows that for medium capacity reactors (BN-800 type) it is possible to develop a core with MOX fuel for effective plutonium utilization on the base of using of fuel pin structure with low fuel volume fraction and increased enrichment of plutonium and also using of absorber shield.

It is provided a high safety level (optimal Doppler- effect and close zero or negative value of SVRE).

3. Considered scenarios of plutonium utilization show that fast reactors using in the closed nuclear-power centers system fully in Russia satisfies the requests of effective plutonium using and with simultaneous solution of unspreading problem.

4. Advanced core of BN-800-type reactor from point of view of increasing of plutonium utilization allows in the conditions of thermal and fast reactors to solve the problem of spent fuel radiotoxicity reduction by means of including produced actinides into fuel cycle.

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ADVANCED DESIGN OF THE FAST BREEDER REACTOR IN PNC

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Abstract

The new Long-Term Program for Research, Development and Utilization of Nuclear Energy, issued by the AEC of Japan in June of 1994, indicates that in addition to efforts to safety, reliability and economic efficiency, possibilities regarding technology capable of meeting diverse needs of future society, including reduction of impact on the environment and assurance of nuclear non-proliferation, should be pursued. to widen technological options. Following this policy, PNC has started to make studies on Advanced Nuclear Recycling Technology and a new concept of a fast reactor core with a newly designed fuel has been identified.

The advanced concept is aimed at commercialization of FBRs and features,

(1) Nitride Fuel Core to enhance passive safety

(2) Ductless Fuel Assemblies

(3) Minor Actinide Recycling

It was decided in PNC that a new reactor to test these features should also be designed, in order to demonstrate the new concept.

1. Background

In Japan, it was decided from the beginning of the national peaceful nuclear power development program that plutonium should be recycled after nuclear fuel was drawn out of the power reactor as spent fuel. Since then, the commercial Magnox type reactor and Light Water Reactors have been introduced and in parallel to this, the fast breeder reactor and the nuclear fuel recycling development program has been proceeding. This is planned under the Long-Term Program for Research, Development and Utilization of Nuclear Energy, which is the national program issued by the Atomic Energy Commission of Japan and generally reassessed and newly issued every 5 years.

At present, utilizing recycled spent fuel, the prototype fast breeder reactor Monju is completed and has demonstrated the first power distribution at 40% full power to the local district on the 29th of August 1995. Related programs such as MOX fuel fabrication and spent

fuel reprocessing have both been successfully developed to achieve their original purpose and construction of the first commercial reprocessing plant has just started at Rokkasho, Japan.

However, in the latest national long range program, issued in June, 1994, it was noted in "Research and Development of Advanced Nuclear Fuel Recycling Technology"

— In development and utilization of nuclear energy it is important not only to strive for improvement of safety, reliability and economic efficiency but also to pursue possibilities regarding technology capable of meeting diverse needs of future society, including reduction of impact on the environment and assurance of nuclear non-proliferation, thereby widening technological options.

Following this policy and owing to a few years of preparatory studies, PNC has started to make studies on this Advanced Nuclear Fuel Recycling Technology and from the works done during the Fiscal Year 1994, a new concept of a fast reactor core based on an optionally designed fuel concept has been identified. It has been decided that, added to the experimental reactor JOYO and the prototype power reactor MONJU a new reactor to test these features should be designed at PNC, in order to demonstrate the performance of the new concept. Construction of this plant is expected after authorization by the AEC.

2. Advanced Concept

The target of the advanced reactor design should be to dedicate to achieving economic efficiency throughout the whole nuclear fuel cycle, enhancing new requirements and applying the most advanced technology. In the Japanese FBR program, the plant system is scaled up starting from the experimental fast breeder reactor to the prototype type and then to the demonstration type.

As described in the 1st section, new requirements are items such as decreasing future burdens to the global environment and enforcing stronger (than present) proliferation resistance.

From the studies done by the reactor design section of PNC, nitrides were chosen with the highest priority as the future candidate. This was based on the assumption that nitride fuel fabrication as well as spent fuel reprocessing would be developed by the time 2030s when FBR commercialization is expected. One of the most attractive reasons

of this choice was the behavior of the core during ATWS (anticipated transients without scram) and this was introduced by Hayashi in A Conceptual Design Study of a Large FBR Plant Enhancing Passive Safety, presented at the International Topical Meeting on Sodium Cooled Fast Reactor Safety at Obninsk, Russia October,1994. However, as optional material metal and oxides with innovative recycling technology is also looked into.

In the studies of FY1994 the following surveys and results were attained.

Core Performance

In order to evaluate the performance, the core and the fuel assembly were specified, as a starting point , as follows.

Reference Core

Operating Cycle Length	Approx. 20 months
Core Average Burnup	150 MWd/t
Fuel Core Residence Time	100 months
Maximum Linear Heat Rate	MOX Approx. 300 W/cm for MN Approx. 400 W/cm
Core Configuration	Homogeneous Core
Nitride Core	spectrum-adjusted by ZrH
Core Height	MOX Approx. 100 cm for MN Approx. 80 cm

Reference Fuel

Type	Ductless
Volume Fractions	Vf:VSUS:VNa=47.3:17.5:35.2
Fuel Outer Diameter	8.5mm
Fuel Smear Density	MOX 83.7%TD for MN 80.0%TD
Sub-Assembly Pitch	171mm
Pins/Assembly	271

Calculations and Analyses showed that,

(1) From comparison of size dependency between core sizes of 2.5m, 3.0m and 4.0m it was found out that there was small difference in the Doppler Coefficient and the Sodium Void Reactivity.

(2) When minor actinides are recycled, the effect of contamination by rare earth is significantly large.

(3) In the case of an Unexpected Loss Of Flow, under certain temperature conditions, coolant boiling can be prevented in both the MOX core using GEM and the spectrum-adjusted MN core. Considering the penalty to the plant design, 3% of minor actinides and about half in percentage of contamination by rare earth was tolerable to maintain the situation.

(4)   Additional analysis on Plutonium burning was done to ascertain the flexibility of the core configuration of the ductless fast core. Shuffling the absorber assemblies and the core fuel assemblies high burnup up to 150MWd/t with plutonium enrichment from 40 to 50 % was achieved by calculation.

Mixture of Minor Actinides

The purpose of recycling minor actinides is to economically optimize the nuclear fuel cycle and reduce burdens from radioactive waste, putting in mind stronger resistance to nuclear proliferation. In the studies made, characteristics of the mixed material such as decrease of thermal conductivity and melting points were considered as well as the higher decay heat of spent fuel, which could disturb the fuel handling system. Three different ways to assemble the fuel was compared. One was to homogeneously spread the minor actinides with uranium and plutonium. Another was to make target pins and spread them in a certain percentage to an assembly and the last was to make a target assembly with minor actinide pins.

It was concluded that, with the present outlook on existing recycling technology and with an optimistic view on future partitioning of rare earth, fuel assembly with 7 to 10 % contents of minor actinide pins would be the most effective way of burning. The decay heat becomes higher because of the increase of the recycling minor actinides, however, higher burnup at a range near 150MWd/t did not necessarily result in higher decay heat.

Fuel Assembly

General requirements for the design were to improve the economy of recycling, add safety features, decrease waste and to effectively burn plutonium in necessary situations.

In order to satisfy these requirements, design studies were done to seek high burnup for longer life, low power density and pressure drop at the core, larger allowance for easier fabrication, easier disassembly for reprocessing, minor actinide recycling and removal of duct to decrease the increase of waste.

In the beginning of our studies the ductless fuel type assembly was selected as a most promising candidate.

Through the studies it was recognized that the higher fraction of fuel volume enables decrease of loss of burnup reactivity and gain of breeding ratio. The loss of capacity to precisely distribute coolant flow, enables flexibility to core configuration. The removal of the hexagonal duct as well as the planned simplified coolant entrance structure, decreases a significant amount of material and also removes the effect of damage of the duct from irradiation which results in longer life of the assembly inside the core. Additionally, higher natural circulation to increase safety of the core can be expected because of the lower pressure drop.

However, detailed design of flow distribution, new safety and seismic design is needed and much work is under way to see the feasibility of the design.

Interface with Plant Design

The impact of a new core/fuel design to the plant is mainly featured in 2 areas. One is size and shape of the fuel and the other is its radioactivity. Other features would be control rod configuration, failed fuel detection system etc.

In our preliminary study, areas with possibilities for large changes were identified. The largest impact to the reactor vessel is the core support system and the flow path of the coolant. Higher decay heat affects the heat removal capacity, fuel handling and transportation system and the spent fuel storage system. In the case of minor actinide recycling, the impact to fuel handling of new fuel can also be significant.

Through the overall review, it was recognized that if the decay heat of the spent fuel becomes significantly large the whole recycling system, including the transportation and site location could be affected. However, there seemed to be too many parameters and uncertainties involved and the study only suggested its importance.

3.Conceptual Design of the Advanced Recycle Fast   Test Reactor

In parallel to the design study of the advanced core, design of a new test reactor, tentatively called the **Advanced Recycle Fast   Test Reactor** has started.

The size of the core and expected performance is to be decided from the advanced core concept mentioned before. In order to minimize the plant cost..

Present Features of the Reactor
- Core physics tests of large scale and advanced type fuel
- Advanced fuel performance tests
- Demonstration of advanced recycling system
- Passive safety tests
- Demonstration of flexibility of core configuration
- Test of ductless fuel

Future Schedules
Detailed design of the ductless fuel assembly is going on at the present time and the first phase of the conceptual design of the **Advanced Recycle Fast Test Reactor** has just started. It is expected to be completed at the end of fiscal year 1996, March, 1997.

ABBREVIATIONS

AEC	-	Atomic Energy Commission
ASS	-	Above Structure System
ATWS	-	Anticipated Transients Without Scram
B_4C	-	Boron Carbide (absorber material)
CDA	-	Core Disruptive Accident
CEFR	-	China Experimental Fast Reactor
CRDM	-	Control Rod Drive Mechanism
CSR	-	Control and Safety Rod
CSS	-	Core Support Structure
DFBR	-	Demonstration Fast Breeder Reactor
DMRS	-	Direct Decay Heat Removal System
DPA	-	Displacements Per Atom
DRACS	-	Direct Reactor Auxiliary Cooling System
DSR	-	Diverse Safety Rod
EBR	-	Experimental Breeder Reactor
ED	-	Effective Days
EVTP	-	Ex-Vessel Transfer Position
FA	-	Fuel Subassembly
FBR	-	Fast Breeder Reactor
FBTR	-	Fast Breeder Test Reactor
HA	-	Heavy Atoms
IFTM	-	Inclined Fuel Transfer Mechanism
IHTS	-	Intermediate Heat Transport System
IHX	-	Intermediate Heat Exchanger
IVTP	-	In Vessel Transfer Post
IWGFR	-	International Working Group on Fast Reactors
KAERI	-	Korea Atomic Energy Research Institute
KALIMER	-	Korean Advanced LIquid MEtal Reactor
LMFR	-	Liquid Metal (cooled) Reactor
LRP	-	Large Rotatable Plug
LWR	-	Light Water (cooled and moderated) Reactor
MOX	-	Mixed Oxide (PuO_2, UO_2) Fuel
NPP	-	Nuclear Power Plant
NSSS	-	Nuclear Steam Supply System
PFBR	-	Prototype Fast Breeder Reactor
PNC	-	Power Reactor and Nuclear Fuel Development Corporation
PRISM	-	Power Reactor Innovative Small Modular
PSP	-	Primary Sodium Pump
RCB	-	Reactor Containment Building
REARA	-	Reactor RApides
RVACS	-	Reactor Vessel Auxiliary Cooling System
SG	-	Steam Generator
SGACS	-	Steam Generator Auxiliary Cooling System
SRP	-	Small Rotatable Plug
SVRE	-	Sodium Void Reactivity Effect
UFS	-	Upper Fixed Shield
UIS	-	Upper Internal Structure
ULOF	-	Unprotected Loss Of Flow
ULOHS	-	Unprotected Loss Of Heat Sink
UTOP	-	Unprotected Transient Overpower

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246

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