

A General Description Of

The NRX Reactor

by

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SYNOPSIS

modified

The NRX Reactor structure, equipment and experimental facilities are described. The purpose of the various components is explained using photographs and diagrams as much as possible.

Dimensions are given so that the reader can visualize the relative sizes of the components.

The report is meant to be an introduction to the NRX Design and Operating Manuals, from which detailed information can be obtained.

It is expected that the report will be of value to trainee NRX Reactor Operations personnel and to those persons who require only a general knowledge of the reactor.

A bibliography of AECL reports pertaining to NRX is given. The report is 80 pages long and contains 20 figures.

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I. INTRODUCTION:

The NRX reactor is a natural-uranium-fueled, heavy-water-moderated reactor with light-water cooling on the fuel rods. The reactor went critical on July 22, 1947 and for many years was the largest high-flux experimental reactor in the world.

NRX was designed as a plutonium-production and experimental reactor at a power of 20 MW. With modifications in 1950 and 1952 the maximum power was increased to 40 MW. In 1961 the operating power was increased to 42 MW. The main work of the reactor at the present time is the testing of fuel elements for power reactors, the irradiation testing of materials to be used in power-reactor construction and the production of radioactive isotopes.

This report is a general description of the reactor, using diagrams to give the purpose of the reactor equipment. Detailed information can be obtained from the NRX Design and Operating Manuals and from the NRX Reactor Branch Handbook, IOI-225.

A short bibliography of AECL reports on NRX is given at the end.

II. REACTOR STRUCTURE

A. Introduction

NRX is housed in a steel-reinforced brick building 113 feet by 145 feet by 90 feet high. The reactor hall is serviced by a crane of 25-ton capacity.

Fig. 1 is a view of the reactor taken from the north-west corner of the reactor hall. Some of the components to be described later are labelled on the photograph.

Fig. 2 is a cutaway drawing of the reactor viewed from the south-west.

Elevation cross-section views are shown in Fig. 3. Note that this diagram is half a north-south elevation and half an east-west elevation.

Fig. 4 is a plan view of the reactor structure. The section is taken at approximately the mid-elevation of the reactor vessel.

For a detailed description of the reactor structure the reader should see the NRX Design Manual A-34. The dimensions and drawing numbers of the structure are given concisely in the NRX Reactor Handbook, IOI-225.

B. Reactor Components

1. Calandria

The heart of the NRX Reactor is the reactor vessel or calandria. The calandria is a cylindrical aluminum tank 8 feet 9 inches in diameter and 10 feet 6 inches high. The heavy-water moderator is contained in the calandria.

The calandria is shaped much like a water boiler in that it has an outer cylindrical shell and inner tubes extending between thick end plates.

The outer shell is made of 1/4-inch-thick aluminum sheet. The upper six inches of the shell is formed from a corrugated piece of 1/4-inch-thick sheet aluminum welded to the main shell. This corrugated section forms an expansion joint to take up thermal stresses in the calandria caused by variations in the heavy-water temperature during reactor operation. The expansion joint can be seen near the top of the calandria in Fig. 2.

Welded to the top and bottom of the calandria shell are aluminum tube sheets three inches thick which form the end plates of the vessel. The 198 calandria tubes are fastened to the top and bottom tube sheets by rolled joints. The calandria tubes are 1/16-inch-thick aluminum cylinders 2-1/4 inches inside diameter. The central thimble, a 1/4-inch-thick aluminum tube of 5-1/2 inches inside diameter, is in the center of the vessel as shown in Fig. 3. The tubes are arranged in a hexagonal lattice 6-13/16 inches between centers.

Cooling-water passages are cut between the rows of holes in the tube sheets. There are sixteen passages in the top tube sheet and eight passages in the bottom tube sheet, the latter being partially cooled by the heavy water. Helium gas flows through the tube sheets around each calandria tube rolled joint. This gas, under slight pressure, prevents the outleakage of heavy water or heavy-water vapour from the calandria.

The calandria is situated in the reactor structure as shown in Fig. 2 and 3 such that the bottom of the calandria is approximately two feet above the main floor of the reactor hall. The boundary between the light and dark paint on the outer reactor concrete shield, as shown in Fig. 1, marks the mid-plane of the calandria.

2. Graphite Reflector and Thermal Columns

Surrounding the side of the calandria is the graphite reflector. As seen in Fig. 2, 3 and 4 the reflector is made in two sections called the inner and outer reflector which are annular rings made up of graphite blocks. The inner reflector is 9 inches thick and is situated with its inner face 1-1/2 inches from the calandria. An annular gap 2-1/2 inches wide called the J-rod annulus exists between the inner and outer reflector. The outer reflector is two feet thick. The reflector is cooled by air flowing down the J-rod annulus.

The reflector reduces the neutron leakage from the reactor by returning to the calandria many neutrons that would otherwise be lost. It has been estimated that the calandria diameter would have to be increased by three feet for the same critical height if the reflector was not present. This, of course, represents a saving in heavy water and fuel rods.

As seen in Fig. 2, 3 and 4, two extensions of the graphite reflector called the North and South thermal columns extend to the outer face of the reactor block. The thermal columns are terminated at the outer ends by cadmium-lined lead and steel doors. The inner end of the columns are 5 feet 10 inches square, while the outer faces are 6 feet 8 inches square. The thermal columns are a source of thermal neutrons used for research experiments. The reactor-control-system ion chambers are situated at the outer end of the North thermal column.

The graphite in the reactor weighs approximately 58 tons. The composition of the NRX graphite is given in Report No. IOI-119 by J.A. Morrison.

Stored energy in the NRX reflector graphite is not at present a problem because the operating temperatures of the graphite prevent the storage of a dangerous amount of energy. The latest measurements of stored energy in the NRX graphite are given in Report No. AECL-889 by H.B. Hilton and E.A.G. Larson.

3. Side Thermal Shields

As shown in Fig. 2, 3 and 4 there are two concentric cast-iron thermal shields surrounding the graphite reflector. These thermal shields absorb radiation that passes through the graphite reflector and dissipate the resulting heat, thereby protecting the concrete shielding from thermal stresses. Each shield is six inches thick, the inner one being 1-1/2 inches from the outer reflector. A two-inch gap separates the two shields and a two-inch gap separates the outer shield and the concrete biological shield. The

total weight of the cast iron in the shields is 155 tons.

These thermal shields are cooled by the reactor cooling air which flows up each side of the outer shield before entering the J-rod annulus.

4. Lower Thermal Shields

There are five lower thermal shields beneath the calandria which absorb most of the radiation passing out of the calandria through the bottom tube sheet. A four-inch-thick steel plate supports the shields which are stacked one upon the other as shown in Fig. 2 and 3.

The shield immediately beneath the bottom tube sheet is the stainless steel lower auxiliary thermal shield. This shield is made of a 1-1/2-inch-thick plate with water channels cut into it and a 1/2-inch-thick cover plate welded over the channels. Installed during the reactor rehabilitation after the 1952 accident, the added cooling capacity provided by the shield enabled the maximum reactor power to be increased from 30 MW to 40 MW. The shield is cooled by water from the main reactor coolant supply reduced in pressure to 30 lb/in² and dissipates approximately 30 kW during reactor operation at 42 MW.

Four water and steel thermal shields are below the lower auxiliary thermal shield. Each shield is 12 inches thick made up of a four and a two-inch-thick steel plate and six inches of water in the form of a circular sandwich weighing approximately 20 tons. The water slows down the fast neutrons which are then captured by the steel. The shields are cooled by a common closed water system which circulates the water through each shield in turn and also through two similar steel and water shields above the calandria. The thermal-shield-recirculation system dissipates approximately 80 kW during reactor operation at 42 MW.

5. Upper Thermal Shields

There are three thermal shields above the calandria. The inner sections of these shields can be removed, as shown in Fig. 3, in order to replace the calandria.

The two top inner shields are twelve-inch-thick steel and water sandwiches similar to the lower thermal shields discussed in Section II-B-4 above. The inner shield immediately above the calandria was a steel and water shield but, during the rehabilitation following the 1952 accident, the original shield was replaced with an aluminum and water shield because the extremely active steel and water shield made the disassembly of the reactor quite difficult.

The outer upper thermal shields are ring-shaped steel and water sandwiches which support the inner thermal shields.

The upper steel and water thermal shields are cooled by the thermal-shield-recirculation system as mentioned in section II-B-4 above. The aluminum shield is cooled by the high-pressure water reduced to 30 lb/in². This water is used since the chlorine in the low-pressure water system would corrode the aluminum shield. The aluminum shield dissipates 60 kW during reactor operation at 42 MW.

6. Master Plate

The master plate, shown in Fig. 2, serves to position and support all the fuel rods in the reactor. It is a four-inch-thick steel plate, 13 feet 4 inches in diameter with a top layer of stainless steel 0.012 inches thick protecting it from corrosion. Chamfered shoulders around each lattice and J-rod annulus position support the rods by shoulders on the outer sheath of the rods.

7. Biological Shield

The Biological Shield is the eight-foot-thick concrete shield that surrounds the reactor. The outside of it is seen in Fig. 1 while Fig. 2, 3 and 4 show it in cross-section.

Built of ordinary concrete, this shield reduces the radiation from the reactor to well below the biological tolerance level. This low radiation level is required so that research experiments operating around the reactor will not be affected by background radiation from the reactor.

As seen in Fig. 2, the main biological shield forms the support for the whole reactor structure. It is pierced by many experimental and instrument holes as well as the thermal columns. The holes through this shield will be discussed later under "Experimental Facilities".

Three "rooms" are built into the top of the biological shield. One small room, four feet square, houses the Sheldon fan which ventilates the upper header room. The "two" recombination rooms each six feet square and eight feet deep have been made into one room. The recombination rooms were built to contain equipment to recombine the radiolytic decomposition products such as H₂ and D₂ collected from the heavy water by the helium system. The decomposition rate turned out to be much lower than expected so that only a small recombination system was needed, and most of the space in the rooms is given over to experimental use.

Pipe chases in the main biological shield connect the upper and lower header rooms, which, as seen in Fig. 3, are the spaces where the top and bottom of the fuel rods are connected to the main coolant system.

There are four removable biological shields above the upper thermal shields. These concrete-filled steel trays each weigh approximately 18 tons. The steel is one inch thick and the concrete 16-1/2 inches thick. These shields can be removed when it is necessary to replace the calandria.

8. Revolving Floor

As shown in Fig. 1, 2 and 3 this plate forms the top of the reactor structure. It is a shield against radiation from the shut-off rods or any experimental rods that allow radiation to stream upwards from the reactor core.

The revolving floor is a sixteen-inch-thick water and steel sandwich made in two sections. The fixed outer section is made of two eight-inch-thick rings. They form a step which supports the movable inner section.

Nylon balls in a race grooved into the inner and outer section of the floor support the movable shield and allow it to rotate easily.

Two circular manhole covers three feet in diameter made of steel and masonite layers are set into the revolving floor. A 6-1/2 inch hole in each manhole can, by using an indexing system and rotating the inner section and the manhole cover, be set up over any reactor lattice position. Rods are then removed through these holes which are normally closed by steel plugs.

Holes of 2-1/2 inches diameter normally closed by steel plugs, are cut through the fixed deck plate and lead to the J-rod annulus. This ring of holes is numbered 0 to 99. Two 7-1/2-inch-diameter holes have been cut into the positions formerly occupied by J-rod holes 1-2 and 3-4. These large holes can be used for installing bulky equipment for research experiments. The use made of the J-rod annulus will be discussed later in Section I-C-7.

C. EXPERIMENTAL FACILITIES

This section will describe those experimental facilities of NRX that are a part of the structure of the reactor. Experimental facilities that have been developed for irradiation in the reactor core itself will be discussed later.

Detailed descriptions of the experimental facilities are given in report No. CRE-400J by E.J. Wiggins and in the NRX Design Manual No. A-34.

1. Self-Serve Units

The self-serve units were designed for the irradiation of small samples of materials in a relatively high neutron flux. The samples can be irradiated in a neutron flux of from 0.3×10^{13} neutrons/cm²/sec to 1.7×10^{13} neutrons/cm²/sec when the reactor is operating at 40 Megawatts. These flux figures are taken from Report No. CRDC-730 by R.E. Jervis.

The self-serve units are situated at the west side of the reactor block as shown in Fig. 1 and 2. Fig. 4 shows a plan view of the self-serve holes. Fig. 5 is a cross-section of a single self-serve unit.

There were originally eighteen of these units with a total capacity of 60 capsules but, as seen in Fig. 1, one bank of three units has been removed and the holes can be used for research experiments. Three self-serve units extend through the inner reflector as shown in Fig. 5 and contain five samples each. The innermost sample is 4 inches from the calandria wall as shown in Fig. 5. The rest of the units end at the J-rod annulus and contain only three samples each.

The main size limitation on a self-serve irradiation sample is that it fit into a 2-1/4-inch-diameter sphere. Usually the sample is placed in a standard capsule of super pure aluminum 1-3/4 inches high, 7/8 inches outside diameter with a 1/16 inch thick wall. The cover is cold welded to the capsule by a hydraulic press. The capsule is held in the aluminum ball by a cap fastened with aluminum pins. A diagram of a capsule is shown in Fig. 19(c).

In order to install a sample the plug A in Fig. 5 is slid out by the handwheel G until the desired receptacle D is in position. The ball containing the sample is rolled down the inlet pipe. The sample plug is then returned to its original position. The sample is removed by reversing the procedure. The sample plug is rotated through 180° and the sample rolls into the pile sample flask U.

The pile flask, containing the sample, is then taken to a shielded device called the self-serve extractor. The ball is rolled out of the flask. The sample is removed from the ball and put into a lead flask of from 2 inches to 4-1/2 inches thickness depending on how radioactive the sample is. An extremely active sample may be removed from the pile flask and placed in a six or ten inch thick lead flask under water. All self-serve sample capsules are weighted so as to sink in water in the event that the latter procedure is necessary.

Suitable interlocks are arranged so that, for example, two samples cannot be installed in one position and no ball can be removed without a pile flask being in position.

The duration of irradiations in the self-serve facility vary, in general, from a few minutes to one or two weeks. Other facilities are used if longer or shorter irradiation times are desired. Typical samples are gold and sodium and various miscellaneous research targets.

The samples are cooled by air drawn into the reactor through the self-serve inlet and outlet holes by the reactor air system. Samples are limited to 30 watt heat output, calculated assuming they will be in the maximum flux. No materials will be installed that decompose under irradiation, as excessive gas pressure may rupture the capsule.

2. Experimental Holes

There are three holes of twelve inches internal diameter and twelve holes of four inches internal diameter that pierce the shielding radially from the outer face of the reactor to the calandria wall. These holes are situated on the eastern face of the reactor as shown in Fig. 3 and 4.

Each hole has a cast iron inner gate located at the inner face of the biological shield which is operated from the outer face of the reactor. An outer lead gate in a cast iron housing is located at the reactor face.

The holes are lined with aluminum tubing through the graphite reflector and with steel tubing through the side thermal shields and biological shield. The holes are stepped so that escaping neutrons will not have a straight path to follow. When not in use the holes are closed with plugs made of steel, graphite and concrete. The plugs can be removed and experimental equipment installed in the holes using one of two horizontal lead flasks that fit against the reactor block. There is one flask for the twelve-inch holes and one for the four-inch holes. The smaller flask (shown in Fig. 1 and 2) is equipped with a vertical sliding door and an extractor head operated by a full length feed screw. The flask can be positioned at the reactor face by using the reactor hall crane.

Samples and plugs in the experimental holes are cooled by air drawn in past the plugs by the reactor air system. Samples with high heat outputs may have to be provided with their own cooling systems.

The holes are used for nuclear physics experiments and as material irradiation facilities.

3. Pneumatic Holes

At the north eastern corner of the reactor near the north thermal column are two pneumatic holes. They are located 2-3/4 inches above and below the calandria mid elevation and extend tangential to the calandria to the inner face of the graphite reflector as shown in Fig. 4. Originally designed for the irradiation of short lived materials that would be installed and removed by air pressure, the holes have never been used for that purpose. At present a graphite plug is installed in the upper hole, P1, containing six thermocouples that are measuring the temperature of the inner-reflector graphite.

4. Instrument Holes

Sixteen instrument holes of various sizes pierce the reactor structure at various elevations. Some of these are tangential to the calandria and some are blind holes. Some have ion chambers in them for measuring the neutron flux of the reactor. Others have been used for irradiation facilities at various times, notably hole I-2 shown in Fig. 4 which during 1958 contained the horizontal organic cooled loop test section to be discussed in section VI-A-2(d).

5. Miscellaneous Holes

Four 2-1/2-inch-diameter holes for measuring temperature and pressure, originally intended to be used for checking the performance of the reactor cooling air system, are available for experimental purposes if required.

There are eight one-inch-diameter holes through the upper shielding leading to the calandria. These holes were originally intended to give access to experimental tubes in the calandria between adjacent fuel rods. In order to simplify calandria construction these tubes were omitted. Two of the holes are used for thermocouple leads but because of the restricted space in the upper header area the rest of the holes have never been used.

A velocity-selector hole, 5-1/4 inches in diameter, is situated over each thermal column as shown in Fig. 3. These holes were designed to provide vertical beams of neutrons for use with mechanical neutron choppers. They have never been used for experiments and are closed by steel plugs.

6. Thermal Column

The structure of the thermal columns was described in Section B-2 and is shown in Fig. 2, 3 and 4.

The main doors on the outer edge of the thermal columns have a small door set into them which can be raised independently so that neutrons can stream out a 16-inch-square hole.

The section of the thermal column from the side thermal shields to the outer face of the biological shield is movable. This movable section is divided into three 30-inch-long tiers, each on shallow four-wheeled trucks. They can be removed from the reactor so that equipment can be set inside the column.

Various portions of the removable sections may be taken out. Square or round openings may be obtained depending on the neutron beam required for the various experiments.

A central plug, 4-1/2 inches in diameter, which leads directly to the calandria can also be removed if necessary.

7. J-Rod Annulus

The J-rod Annulus is the annular gap between the inner and outer reflector mentioned in Section II-B-2. This facility was designed for the irradiation of thorium in the form of thorium metal or thorium oxycarbonate rods. During the early days of the Project, thorium was called J-metal for security reasons. This facility thus came to be called the J-rod annulus.

The 2-1/2-inch-wide annulus has 84 inlet holes which could be used for irradiations. Fourteen of these are presently blocked by the plugs in the horizontal experimental holes.

Thorium rods were irradiated until 1957 when they were replaced with cobalt slug rods which are sometimes called cobalt J-rods.

There are, at the present time, 65 cobalt slug rods in the J-rod annulus. Each rod contains 3500 grams of cobalt metal in the form of slugs 1 inch long and 1/4 inches in diameter sheathed in 1/16 inch thick aluminum. Five hundred and four slugs in 72 layers of seven slugs each make up the rod. The slugs are held in circular aluminum trays which are stacked on a 7-1/2-foot-long, 5/16-inch diameter aluminum rod which has a large threaded nut on each end. A tray at each end of a layer holds the slugs in a circular array, 1-1/2 inches in diameter. An outer aluminum tube 1-5/8-inches in outside diameter and 11 feet long acts as an

outer sheath. The main reactor cooling air flows by the rods through the J-rod annulus at 16,000 ft³/min. The slugs are irradiated for three to four years to an average activity of about 5 curies/gram.

III. FLOW SYSTEMS

A. Introduction

There are four main flow systems connected with the NRX Reactor. These are: the heavy-water moderator system; the helium system; the light-water coolant system and the cooling air system. Although the systems are interconnected they will be discussed separately for the sake of clarity.

B. The Heavy Water System

1. Purpose

The NRX Reactor uses heavy water as a moderator. The heavy water moderates or slows down the fast (high-energy) neutrons produced in fission. The cross-section for thermal (low-energy) neutron fission of U-235 is much higher than the cross-section for fission by fast neutrons. The moderator thus sustains the chain reaction by slowing down the neutrons from a velocity of the order of 10,000 miles per second to about one mile per second.

The heavy-water system normally contains about 19-1/2 tons of heavy water which at the current price of \$28.00 per pound is worth approximately \$1.1 million.

The term "polymer" has been used for heavy water. During the early days of the Project the term "heavy water" was classified. The use of "polymer" has now essentially been discontinued.

A simplified schematic diagram of the heavy-water system is shown in Fig. 6. As can be seen from this figure, the essential parts of the system are the calandria, three storage tanks, two coolers, two supply pumps and two circulating pumps. A detailed description of the system can be found in the NRX Design Manual IOI-47, Section A-5.

2. Calandria

The calandria has been discussed previously in the section on Reactor Structure. During normal reactor operation the calandria is nearly full of heavy water (as shown in Fig. 2). As seen in Fig. 6, the calandria is fitted with nine lines which are used for filling, draining and circulating the heavy water. These lines

are made of aluminum tubes 2-1/4 inches in internal diameter. They terminate at different levels in the calandria in order that maximum circulation of the heavy water is obtained. Except for the aluminum calandria and these aluminum lines the heavy-water system is made of stainless steel.

3. Heavy Water Storage Tanks

The three stainless steel tanks for storing heavy water are located under the reactor hall floor at the north western side of the reactor as shown in Fig. 2.

No. 1 Storage Tank, made of 3/16 inch thick stainless steel, is 7-1/2 feet in diameter, fourteen feet long and has a capacity of 3670 Imperial gallons. This tank is normally empty and is used only as an emergency storage tank. If, for example, the heavy water in the calandria became downgraded with light water through some accident, No. 1 storage tank would be used to store the downgraded heavy water until it could be purified. No. 1 storage tank is filled by opening the manually operated three way valve V-3160 shown in Fig. 6 and can hold the complete reactor charge of heavy water.

No. 3 Storage Tank, made of 3/16 inch thick stainless steel, is 8 feet in diameter, 6 feet 7-3/8 inches high, and has a capacity of 1729 Imperial gallons. This tank is also called the dump tank. As will be discussed later, when the reactor is shut-down the heavy water level in the calandria is lowered to 140 cm (approximately half full). The water is put into No. 3 Storage Tank. The amount of heavy water kept in the system is just sufficient to fill No. 3 tank, the piping between the tank and the calandria, and the calandria to at most 140 cm.

No. 2 Storage Tank, made of 3/16 inch thick stainless steel, is 7-1/2 feet in diameter, 8 feet 9 inches long, and has a capacity of 2180 Imperial gallons. This storage tank is normally empty. However, when it is desired to drain all the heavy water from the calandria, Valve No. V-3159, shown in Fig. 6, is opened thereby putting all the heavy water into No. 2 and No. 3 storage tanks. When the reactor is started up following a complete draining of the heavy water, No. 2 storage tank is emptied first.

4. Heavy Water Coolers

On passing through the calandria, the heavy water is heated mainly by gamma rays captured in the water and in the calandria structure. It is desirable that the heavy-water moderator be as cool as possible because the cooler it is, the more dense it is and, therefore, a more effective moderator. With a complete reactor loading of natural-uranium rods, about 5% of the fission heat appears in the heavy water.

At 40 MW reactor power this is 2 MW of heat in the moderator. At the time of writing (July, 1961) approximately 2.38 MW or 5.7% of the present reactor power of 42 MW appears in the moderator. The booster rods of enriched uranium aluminum alloy in the reactor do not absorb as many gamma rays as the natural uranium rods and this extra heat appears in the moderator.

The original design of the NRX reactor provided for one heavy-water cooler. This cooler, (now called cooler No. 2) made completely of stainless steel, is a baffled, one-pass exchanger with a floating head. It has 187 tubes $3/8$ inches outside diameter and 135 inches long. The light water is on the shell side and the heavy water flows through the tubes. The outside of the cooler is about 16 inches in diameter and 14 feet long. It is located under the reactor hall floor with the heavy-water storage tanks as shown in Fig. 2.

By 1949, the reactor, which had been designed for 20 MW output, was operating as high as 26 MW. In order to be able to operate at 30 MW during the summer months it was decided to install a second heavy-water cooler. The limit (at that time) of 100°F on the calandria outlet temperature of the heavy water would have been exceeded during the summer when the cooling water from the river approaches 70°F.

In May, 1950 the second cooler, now called cooler No. 1 since the heavy water flows through it first, was connected in series with the original cooler. The new cooler is identical to the older one except that it has 379 tubes each 147 inches long. It enabled the reactor to operate at 30 MW all year while keeping the heavy water temperature below 100°F, the limit considered safe at that time.

With the two coolers in series the calandria inlet heavy water temperature is controlled when possible at 60°F by varying the light water coolant flow through the coolers. During the summer months control of the inlet temperature at 60°F is not possible because of insufficient cooling capacity of the coolers, and the inlet temperature rises to 78°F. The calandria outlet temperature varies from 110°F to 135°F during the year. In the light of better knowledge of the corrosion of aluminum by heavy water the temperature is allowed to exceed 100°F. The heavy-water flow through the calandria is normally 215 Igpm (gal(UK)/min.)

During the winter months the river water temperature falls to about 34°F. Since heavy water freezes at 39°F the cooling water to the heavy-water coolers is turned off whenever the reactor is shut down during the winter.

5. Heavy Water Pumps

As shown in Fig. 6 there are four heavy-water pumps. These pumps are all stainless steel centrifugal pumps with Crane type 1 mechanical shaft seals with ceramic seats and graphite washers.

The two supply pumps each have a rated capacity of 55 Igpm, at a head of 70 feet and are driven by 3 hp motors at 1750 rev/min.

The two circulating pumps each have a rated capacity of 310 Igpm, at a head of 185 feet, and are driven by 25 hp motors at 1750 rev/min.

As shown in Fig. 2 the pumps are under the main reactor hall floor adjacent to the heavy water storage tanks and coolers.

6. Weir Box

Until May, 1958, the heavy-water depth in the calandria (and thus the reactivity of the reactor) was controlled by pumping heavy water into the calandria and out through a weir box as shown in Fig. 6. The weir box was used as a coarse reactivity control with a control rod being used to compensate for small reactivity changes.

The weir box, made of 1/8-inch-thick stainless steel, is seven inches in diameter and two feet long. The two-inch-diameter inlet and outlet lines are of flexible corrugated stainless steel. The weir box is hung from a sprocket by a counter-balanced roller chain, and the sprocket driven by an amplidyne which is controlled from the reactor control console by a selsyn unit. Electrical probes in the box indicate when proper flow is being maintained. The box operates over a range of seven feet corresponding to from three feet to ten feet of heavy water in the calandria. A head of 3-1/2 cm of water is required to maintain flow over the weir.

Since May, 1958 the weir box has been used only as a precise instrument for measuring the amount of heavy water in the calandria when the heavy water inventory is taken during shutdown. The level of the heavy water is controlled during reactor operation by air-operated flow-control valves as will be discussed later. The weir box is positioned at the top of its range during reactor operation where it prevents the calandria from being over filled. If the heavy water flows into the weir box at any time a level probe changes the heavy-water pumps to the 5 Igpm filling rate. The heavy water will then flow back to the storage tank and the calandria cannot be filled beyond about 307 cm.

7. Ion-Exchange Columns

As shown in Fig.6 the heavy-water system is provided with two ion exchange and purification filter units. They are located in the water monitor pit north of the heavy-water storage tank rooms.

Air in the heavy water results in the formation of nitric acid during irradiation. The acid lowers the pH of the water causing corrosion of the aluminum calandria. The ion exchange columns remove metallic ions from the water and maintain the pH at 5.4 to 6. The effectiveness of the ion exchange columns is checked by measuring the conductivity of the water before and after the columns. The conductivity of the water is normally about 0.2×10^{-6} mhos.

The ion exchange column consists of a removable stainless steel resin can, a stainless steel filter cartridge and a steel lead-filled shielding jacket. The resin can is about 34 inches long and four inches in diameter. Stainless steel screens at either end of the resin-can hold the resin in place. Any resin or corrosion products which pass through the screen are filtered out by the filter cartridge which is packed with glass wool.

One of the two parallel columns is used at a time with the other being on standby. Approximately 2-1/2 Igpm of heavy water flow downwards through the column. The mixed bed H-OH Rohn and Haas Amberlite XE-150 resin normally has to be replaced about every six months.

Heavy water is recovered from a spent resin bed using the heavy-water recovery unit. The heavy water is boiled off the resin with steam and the vapour is collected in a cold trap.

8. Heavy-Water Salvage System

There is usually some water leakage from rods in the reactor. This water and any spilled during rod changes flows into the basement sump. The water is periodically pumped into the reactor light-water effluent line after being analysed to make certain no heavy water is present.

If a large heavy-water leak developed, for example, from a calandria tube rupture, heavy water would flow into the sump. This water would be pumped into the salvage system which has two tanks with capacities of 800 Imperial gallons and 1100 Imperial gallons. The water thus salvaged would be purified if necessary and returned to the reactor.

9. Reactor Control With the Heavy-Water System

Originally NRX had four control rods that compensated for fine reactivity changes while the weir box was used as a coarse control of reactivity. In May, 1958 the last control rod was removed from NRX and the control of the reactor is now by automatic variation of the heavy-water level in the calandria.

In conjunction with the reactor control system the calandria may be filled with heavy water by a circulating pump at the rate of 215 Igpm or a supply pump at 50 Igpm or 5 Igpm.

The heavy water from the supply pump goes through the loop-filling line which has a section situated higher than the top of the calandria. Thus, if the supply pump failed, heavy water will not drain from the calandria through the pump to the storage tank. The loop filling line is shown in Fig. 6.

Since during reactor start-up or operation heavy water must be circulated through the calandria and heat exchangers, the circulating pump is always operating irrespective of the calandria filling rate.

As can be seen from Fig. 6 the circulating pumps take water from storage tank No. 3 through valve No. V-3079 and pump it through the coolers to the calandria. The heavy water leaves the calandria through dump valves and control valves, returning to the pumps via the three way valve V-3160. No. 3 storage tank "rides" on the line acting as a surge tank.

There are three control valves (marked "CV" on the diagram) which discharge water to the dump tank at a rate demanded by the control system. For steady-power operation, the discharge rate is equal to the filling rate and the calandria level remains constant. Any reactivity changes in the reactor are automatically compensated for by changes in the control valve position which varies the heavy-water height.

The control valves are three-inch stainless steel double-plug globe valves with a bellows stem seal. They are closed by compressed air and opened by a spring and are, therefore, fail safe. On a reactor trip the valves open completely and thus act as dump valves. During steady operation they are approximately 2/3 closed.

There are four dump valves (marked DV on Fig. 6) which are closed during normal reactor operation, three open fully on a reactor trip allowing water to flow quickly to No. 3 storage tank. The dump valve on the inlet line to the weir box is controlled by a manual switch from the control room. This valve is normally left closed because the differential pressure cell which measures the height of the water in the calandria is attached to this line.

The dump valves are three-inch stainless steel plug globe valves with a bellows stem-seal and are closed by compressed air and opened by a spring.

The initial heavy-water dumping rate with six dump and control valves open is specified at 800 Igpm at a heavy-water height of 270 cm. The dump valves are wired in two separate banks of three valves each chosen so that if one bank failed to open, the second bank will provide a dump rate 65% of that of a full dump on both banks.

10. Heavy-Water Losses and Purity Control

Heavy-water losses must be kept to a minimum. Heavy water is expensive and irradiated heavy water contains tritium which is very toxic.

To ensure that heavy-water leaks at flanges, valves and other fittings can be detected quickly, heavy-water drip-tray detectors called "Beetles" have been installed wherever these leaks might occur. Any water dripping into the tray collects in a small depression at the bottom and makes contact with a probe. An alarm rings in the control room indicating a water leakage which is immediately investigated by the operating personnel.

Leakage of heavy-water vapour into the reactor cooling air system is detected by the "cold finger" sampling apparatus. Daily samples of the air in the exhaust duct are taken through a liquid nitrogen cooled trap. The frozen water vapour collected is analysed for heavy-water content. If the heavy-water concentration is greater than 0.017% which is the heavy water content of normal water, the air in various sections of the reactor building is sampled in order to find the heavy-water leak.

Heavy water for the reactor must meet the following specifications:

1. Isotopic Purity 99.80 weight percent (minimum)
2. Minimum pH 5.8
3. Ammonia 3 ppm
4. Chloride 5 ppm
5. Boron and Cadmium - Nil .

Routine weekly samples of the heavy water in the reactor are analysed. At the discretion of the shift supervisor additional samples may be taken at any time. The water samples are taken by inserting the needle of a laboratory syringe through a gum rubber diaphragm into the heavy-water stream at each of three sampling stations.

The isotopic purity of the heavy-water is continuously monitored by the Tri-Non Analyser which has a flow of 30 cm³/min. through it. This instrument measures the absorption of a particular wave length of infra-red light by the water. This light is preferentially absorbed in light water, and the instrument rings an alarm in the control room if the isotopic purity of the heavy-water drops below 99.80 percent.

The average heavy-water operating loss from 1955 to 1960 is about 25 pounds per month. This loss rate includes loss due to leakage, spills and samplings that could not be recovered.

C. Helium System

1. Purpose

Helium is used to vent the heavy-water system and to equalize the pressure above the heavy-water throughout the system as shown in Fig. 7. The helium also collects the radiolytic gases or decomposition products of the heavy-water and carries them through a recombination system.

The heavy water system and the helium system together form a closed system. Air cannot be used in the system since nitric acid would be formed under irradiation which would cause corrosion in the calandria and would lower the pH of the heavy-water. Helium is used since it is an inert gas and has a very low cross-section for neutron capture.

A detailed description of the helium system is given in the NRX Design Manual IOI-47, Section A-4.

2. Gasholder

The helium gas holder is in a room adjacent to the reactor hall along the south wall of the NRX Building.

The gasholder outer shell is 13 feet, 4 inches in diameter and is made of 1/4-inch-thick stainless steel plate. That part of the gasholder not in contact with helium is made of mild steel while the rest of holder including all piping is made of stainless steel.

The maximum capacity of the holder is 583 cubic feet of which at least 40 cubic feet is left for expansion. There is an oil seal between the two sections. The outer

adjustable section is guided by four vertical rails and is counterweighted with 3600 pounds of lead which gives a pressure of 12 inches of water to the helium system.

As can be seen in Fig. 7 the gasholder is connected to the calandria and the three heavy-water storage tanks by two-inch-diameter stainless steel lines. The adsorber system is connected across the two lines from the gasholder.

The various sections of the helium system are always valved in to allow free flow from one to the other. The adsorber and recombination systems are fed by separate helium blowers from the main gas lines.

3. The Recombination System

As mentioned in section I-B-2 the recombination system is contained in the recombination rooms near the top of the reactor block.

This system is composed of two flame arrestors, a catalyst bed and a helium blower.

The helium blower circulates 10 ft³/min of helium through the recombination system. The present blowers are made of Electrolux vacuum cleaner parts having a 1/10 hp motor run on 400 cycles/sec at 11,000 rev/min. At the time of writing, these blowers are being replaced by 400 cycles/sec motors built by the Rotron Manufacturing Company, Inc. The blowers are inside an aluminum casing 7-1/2 inches in diameter and twenty inches long.

The catalyst chamber is made of 1/4-inch-thick aluminum plate. It is 7 inches in diameter and 13 inches long, the catalyst being held in position by 1/8-inch-thick perforated plates covered by a stainless steel wire mesh cloth. The catalyst, which is deuterized before installation, consists of palladium/alumina pellets approximately 1/8 inches in diameter placed in the chamber to a depth of 4-1/2 inches making a 9.6-pound charge.

The two flame arrestors are three-inch "Protectoseal" pipe-line arrestors. Each comprises a cast stainless steel casing with twenty-five aluminum grid plates.

Thermocouple probes are installed in the gas line before and after the catalyst chamber. These serve to monitor the effectiveness of the catalyst. During normal reactor operation the temperature rise is about 10°F to 20°F, while on a reactor trip it rises to 40° to 50°F because of the gases circulated through it when the heavy-water is dumped.

A pressure switch monitors the flow through the catalyst actuating an alarm when the flow drops to zero and giving a trip signal which will shut-down the reactor in four hours unless the situation is corrected.

4. The Adsorption System

The adsorption system, commonly called the "sorber system", is used to remove nitrogen and any excess oxygen from the helium. Nitrogen enters the heavy-water/helium system whenever a pump or some piping is changed and possibly by diffusion of air into the system through the helium leaks that are present.

The nitrogen is removed by passing the helium gas through beds of activated charcoal cooled by liquid nitrogen.

The main components of the sorber system are a catalyst unit, two freezer driers, the freezer-dryer refrigeration system, a heat exchanger, two charcoal beds and a Kinney vacuum pump.

Helium is drawn from the main gas line at the rate of 2.1 ft³/min as shown in Fig. 7 by a 2 hp blower similar to the one on the recombination system. It passes through a catalyst identical to that on the recombination system which removes most of the oxygen before the gas enters the sorber equipment.

The gas passes through one of two freezer driers which are connected in parallel so that one is operating while the other is being defrosted (reactivated). The freezer drier that removes heavy-water vapour from the helium is cooled by an antifreeze solution which is in turn cooled by a Freon refrigeration unit. The freezer driers are defrosted by two thermostatically controlled chromolux heaters attached to the bottom of each drier.

The helium then passes through a heat exchanger to one of two parallel-connected carbon beds of which one can be used while the other is being reactivated. Each carbon bed contains 10 pounds of activated coconut charcoal. The gas passes down through the charcoal which is cooled by liquid nitrogen in a container fitting around the carbon bed. This container is replaced by a jacket heater when the carbon bed is defrosted. The adsorbed impurities are pumped off by the Kinney vacuum pump. A cold trap cooled by liquid nitrogen between the pump and the charcoal bed traps any heavy-water from the bed and also keeps the vacuum-pump oil from entering the bed. Four pairs of thermocouples measure the temperature of the bed. During reactivation the charcoal bed is heated to 800°F while during operation it is cooled to -321°F by the liquid nitrogen.

The purified helium leaves the beds and passes through the heat exchanger precooling the gas entering the recombination system. The helium then returns to the main system as shown in Fig. 7.

The sorber system operates on the average 30% of the time. Operation is started when the nitrogen content of the helium as reported in gas samples from the main system approaches 0.1%.

5. Helium System to the Tube Sheets

As mentioned previously in section II-B-1 there are helium gas passages cut into the calandria tube sheets connecting the lantern rings around each tube. In the event of a defective seal between a calandria tube and the tube sheet, dry helium would leak into the calandria rather than heavy-water or helium saturated with heavy water leaking outward.

The helium for the top tube sheet is taken from the main helium system through 1/2-inch-diameter stainless steel lines as shown in Fig. 7. The helium is static in that no continuous flow is maintained through the tube sheet.

The helium to the bottom tube sheet has a separate helium cylinder and gas holder supply system. It consists of a standard helium pressure bottle and a gas holder made from a three foot length of 12-inch-diameter pipe. The gas holder is pressurized to seven lbs/in² gauge by means of the helium cylinder once every 24 hours, and is connected to the tube sheet by a 1/2-inch stainless steel line.

6. Helium Losses and Purity Control

Helium losses from the heavy-water helium system average between 400 to 500 standard cubic feet per month. These losses are made up from time to time by adding helium as required.

As mentioned before, the nitrogen content of the helium system is normally kept to less than 0.1% by the operation of the carbon sorber system. In any particular section of the system the deuterium content is not to exceed 5% if the oxygen content exceeds 1%, or 6% if the oxygen content is less than 1%. For the system as a whole the excess deuterium after all the oxygen is used up is not to exceed 1.5%. These specifications guard against the possibility of an explosion in the system.

Samples of helium are taken by attaching an evacuated sample flask to sample valves in the system and collecting the helium. The sample stations are vented to the reactor exhaust-air system to avoid exposure of personnel to tritium.

Samples are taken daily before and after the catalyst in the recombination system and twice a week at the gasholder and in No. 1 and No. 3 heavy-water storage tanks.

D. The Light-Water System

1. Purpose

The light-water system provides the main cooling for the reactor. All reactor components except those cooled by the air system are cooled directly or indirectly by light water.

The light-water system is divided into the low-pressure and the high-pressure systems.

The water in the low-pressure system is at 40 lb/in² pressure. It is the source of supply for the high-pressure water system and provides direct cooling to various components in the reactor building such as the heavy-water coolers as shown in Fig. 8.

The high-pressure water system is kept at approximately 175 lb/in² by a 300-foot-high head tank which rides on the system. The bulk of the water is used for fuel rod cooling in the reactor. Smaller amounts are used for cooling other reactor components and in miscellaneous plant uses as shown in Fig. 9.

A detailed description of the light water system is given in NRX Design Manual IOI-47, Section A-1 and A-2.

2. The Low-Pressure Water System

Approximately 4500 Igpm of water is drawn through a thirty-inch intake line from the Ottawa River by pumps at the AECL power house.

The water enters mixing basins in Building No. 440 flowing to a sedimentation basin where large particulate matter settles out. It is then sent through sand filter beds and into two clearwells which hold 217,000 Imperial gallons of water each. This filtration system was designed for the chemical purification of the water. When the NRX reactor was built it was thought that impurities in the river water might be deposited on the heat-transfer surfaces of the fuel rods and limit reactor operation. It was found that purification of the bulk cooling water was not necessary and only the filtration part of the plant is used.

Water is drawn from the clearwells by the low-pressure pumps. As can be seen in Fig. 8 there are five pumps that can be used to supply low-pressure water. Three of these pumps have capacities of 2200 Igpm against a 45 lb/in² head

being driven by 100 hp electric motors. They are used during normal operation and supply approximately 4200 Igpm. Pump No. 5 is a 720 Igpm pump powered by a 30 hp electric motor. It is used if one of the large pumps fail and also during the summer when cooling water flow to the heavy-water coolers is at maximum. Pump No. 1 is a 1200 Igpm pump driven by a 60 hp steam turbine and is used as an emergency supply pump should the electrical power to the other pumps fail.

Water discharged from the low-pressure pumps enters the low-head tank (Building No. 442) through a twenty-inch-diameter line. The low-head tank is constructed of welded steel plate 66 feet in diameter and 39 feet high. It has a capacity of 800,000 gallons. There are two steam lines in the tank that are thermostatically controlled by the temperature of the water. A twenty-inch-diameter overflow line returns water to No. 2 clearwell. The low-head tank supplies water to the high-pressure pumps, the reactor building low-pressure-water system and to the heavy-water coolers as shown in Fig. 8.

Two strainers remove scale and rust from the water before it enters the reactor building. The strainers are of 30-mesh stainless steel screen in the form of a cylinder 6 inches in diameter and 24 inches long. One strainer is in use at a time, the second being on standby. Pressure-drop measurements across the strainer indicate when it is becoming plugged.

Chlorine is added to the low pressure water that flows through the heavy-water coolers in order to inhibit algae formation. A booster pump rated at 600 Igpm increases the flow of water to the heavy-water coolers to ensure adequate cooling in the summer.

3. The High-Pressure Water System

As mentioned in Section III-D-1 above, the bulk of the water in the high-pressure system is used to cool the fuel rods in the reactor. A simplified schematic flow diagram of the system is shown in Fig. 9.

Approximately 4000 Igpm flows from the low head tank described above to the inlet of the high-pressure pumps through a 20-inch-diameter pipe.

There are three high-pressure pumps, one of which is normally kept on standby duty. The pumps are rated at 1980 Igpm at a 175 lb/in² head. They are driven by 250 hp electric motors.

The pressure head on this water system is provided by the high-head tank (Building No. 444). This 300-foot-high tank also serves as a temporary emergency supply of high-

pressure water should the high-pressure pumps fail. The high-head tank has a capacity of 80,000 Imperial gallons. The tank, built of 1/4-inch-thick steel plate, is about 16 feet high and 29 feet in diameter and stands on six 12-inch "I" beams set on concrete pillars. The inlet pipe is heated by warm water during the winter in order to prevent freezing.

During normal operation about 75 Igpm of water overflows the tank returning to the low-head tank through a 20-inch-diameter line. Should the high-pressure pumps fail, the high-head tank would provide the full high-pressure water flow to the reactor of 3000 Igpm for about twenty minutes. Check-valves on the outlet of the high-pressure pumps prevent backflow through the pumps. When the high-head tank had drained completely, low-pressure water would flow via the 20-inch-diameter-high-pressure pump bypass line from the low-head tank at a maximum rate of 1000 Igpm. However, rather than drain the low-head tank, the steam-driven low-pressure pump No. 1 would be started and would supply low-pressure water at a rate of about 1,000 Igpm. If a prolonged shutdown took place the flow to the reactor could be throttled by the motorized valve between the bottom fuel rod headers and the delay tanks (as shown in Fig. 9) in order to conserve water.

Normally water flows from the high-pressure pumps through the flow-measuring orifice B-1-F-8 and on through the high-pressure strainers. There are two strainers, one being standby. Each strainer is 20 inches in diameter and 48 inches long made of stainless steel screens with 1/64-inch openings. The condition of the strainers is monitored by measuring the pressure drop through them. Each strainer is provided with check valves and gate valves so that each can be isolated separately. The six-inch emergency line to bypass the high pressure strainers tees into the twenty-inch-diameter-line just past the high-pressure strainers as shown in Fig. 9.

Various lines go off from the main line past the strainers and supply water to the experimental loops, the rod-storage blocks, the vertical travelling fuel-rod flask, the thermal-shield recirculation-system cooler and the auxiliary diesels. These components take off about 800 Igpm between the strainers and the reactor flow-measuring orifice E-1-F-16.

About 3100 Igpm of high-pressure water enters the reactor building through the twenty-inch pipe. Approximately 100 Igpm is taken off for cooling experimental rods, the tube sheets, the stainless steel auxiliary thermal shield and the upper aluminum thermal shield before entering the ten-inch-diameter ring header in the upper header room shown in Fig. 2 and Fig. 9.

The water then flows through the top cross headers and down through the various fuel rods. The pressure at the tops of the rods is about 155 lb/in² gauge. Flow through the various types of rods varies from 25 Igpm to 15 Igpm. The pressure of the water at the bottom of the rods varies from 120 lb/in² to 56 lb/in² depending on the flow through the rod.

The water pressure and hence the flow at the base of each fuel rod is monitored continuously by the E-1-P-19 Budenberg gauge system. The bottom of the rod is sealed by O-rings to the bottom valve assembly. A 12-inch-long riser line containing an orifice 0.390 inches in diameter connects to the bottom cross header. Two 1/4-inch-pressure taps in the bottom header assembly are connected by copper lines to three Budenberg gauges. The three gauges have two movable contacts which are used as upper and lower pressure limits to trip* the reactor if these settings are exceeded. If the limit is exceeded for less than one second an alarm annunciates in the control room. One gauge is normally set six lb/in² above and below the operating pressure while the others are set eight lb/in² above and below. In this way the first channel acts as an alarm while if one of the other channels comes in as well the reactor will trip. The flow through the rod is calculated from the pressure measurement as

$$F_{rod} = (2.35 \pm 0.05) \sqrt{P_{rod}-30}. \text{ Where } F_{rod} \text{ is in Igpm and } P_{rod} \text{ is in lb/in}^2.$$

- * A reactor trip is a fast reactor shutdown initiated automatically by the reactor protection system or manually by the reactor operator. When the NRX reactor trips the six shut-off rods drop into the core and the heavy water drains to the 140 cm level of the calandria.

Each rod lower header assembly has a thermocouple well so that the temperature rise across each rod can be measured.

A small flow of water (0.1 Igpm) is taken from the Budenberg-gauge pressure lines for the gaseous-fission-product monitoring system which detects ruptured rods. Two lines from each rod feed into separate gas strippers. The twenty gas strippers collect the entrained gases which are then sent through separate beta counters. Since the two lines from each rod go to separate monitors, a grid system enables the 190 rod positions to be monitored by twenty counters. Alarms annunciate in the control room on a grid board when the radiation exceeds preset limits. In this case the reactor would probably be shut down and the rod removed.

From the base of the rods the water enters via 3-inch-diameter cross headers one of two 16-inch diameter bottom headers which join to form a twenty inch diameter line extending to the delay tanks as shown in Fig. 9.

The two delay tanks (Buildings Nos. 103 and 104) with capacities of 283,000 Imperial gallons each are identical in construction. They are of steel plate varying from 1/4 inch thick to 3/8 inch thick welded to make a cylindrical tank 48 feet in diameter and 25 feet high. A four-turn spiral baffle in the tanks causes the water to take about an hour to go through each tank. Outflow is over a weir from number one tank to number two and thence to the Ottawa River. The two-hour delay from the reactor to the river allows radioactivity induced in the impurities in the water while flowing through the reactor to decay to a safe level. The NRX reactor contributes an insignificant amount of long lived activity to the River. Report No. AECL 1095 by J.E. Guthrie concluded that the whole AECL Project does not contribute a significant amount of activity to drinking water from the Ottawa River.

In addition to the gaseous-fission-products monitoring system noted above, the effluent high-pressure water system has several monitors which measure radiation in the bulk coolant. There is a delayed-neutron monitor between the reactor and the delay tanks. There are three gamma-ray monitors, one before and after the first delay tank and one after the second delay tank.

Samples of the effluent water as well as of the river flora and fauna are analyzed in order to guard against high radiation levels from the reactor being added to the river.

E. The Ventilation and Air Conditioning System

1. Purpose

The NRX air system maintains an atmosphere of the

proper temperature and humidity in the reactor hall and in the experimental loop frames and the storage blocks. It also serves to carry away radioactive dust and gases generated within the building, discharging these through a stack one mile from the reactor. The ^{dew point} temperature of the air entering the reactor hall is kept low enough to prevent condensation on the cold water pipes.

Air circulation through the reactor itself is primarily for cooling the graphite reflector and the side cast iron thermal shields. During reactor operation at 42 MW about 280 kW of heat is removed by the air system.

A schematic diagram of the air system is shown in Fig. 10 taken from the NRX Reactor Handbook. Report No. IOI-219 by W.N. Selander is a recent review of the air-system temperature conditions.

A detailed description of the air system is given in the NRX Design Manual, IOI-47, Section A-3.

2. Air Treatment in Building 124

The air conditioning equipment in Building No. 124 is designed to supply air at a temperature of 70°F and a dew point temperature 3°F less than the temperature of the water entering the reactor building. This prevents condensation on water pipes in the reactor hall.

Air enters Building 124 through a 5 by 7 foot duct provided with a bird screen, a fly screen, and wooden louvers which protect the main fibre-glass filters. The filters are in 20-inch-square frames containing four one-inch-thick layers making a filter panel 8-1/2-feet wide by ten feet high. These filters are removed during the winter months as condensation and freezing on the fibre-glass wool restricts the air flow.

The duct between the filters and the main blower is 10-1/2 feet high, 6 feet wide and 19 feet long. This length of duct contains the air-conditioning equipment such as heating and cooling coils which treat the air in order to reach the required temperature and humidity conditions.

The air-conditioning blower has a rated capacity of 26,000 ft³/min and is driven by a 25 hp motor.

The air enters the reactor building through a 54-inch-square lagged duct near the roof.

3. Air Circulation in the Reactor Hall

The air enters the reactor hall through the duct from Building No. 124 at temperatures ranging from 54°F in the summer to 72°F in winter. The duct splits into four

smaller ducts each one leading to a thermostatically-controlled, steam-heated Low-Boy heater and fan, after crossing an air gap of about twenty feet.

These heaters are mounted on platforms suspended from the steel work supporting the reactor hall roof. Each unit circulates 11,000 ft³/min while the ducts each supply 4,500 ft³/min. The heaters are capable of raising the air temperature in the reactor hall by two to three degrees Fahrenheit. Ducts from the heaters extend down the walls and discharge air near the floor level.

Air is drawn from the main reactor room into the rod-storage blocks, loop frames, vertical flask and the upper and lower header rooms through openings in the floor and structure. The air is exhausted into a U-shaped collector duct which has fans at each end which also draw air from the reactor room.

These fans, called the North and South Wall Fans, mix the air from the duct and the reactor hall and blow it into two ducts of 3-1/2 feet diameter under the hall floor leading to the reactor itself.

The flow in each of these ducts is measured by the orifices E-3-F-8 and E-3-F-9 shown on Fig. 10. The normal flow rate is 6,200 ft³/min and 7,000 ft³/min respectively.

4. Air Circulation in the Reactor

The upper header room and the recombination rooms are ventilated by the Sheldon fan which blows the air into the south duct as shown in Fig. 10. This fan is driven by a 5 hp motor and has a capacity of 3,000 ft³/min at 70°F. The fan and motor are in a recess near the top of the east face of the reactor structure as mentioned previously in Section II-B-7.

The Buffalo fan draws air from the fast neutron rods, to be discussed later, and discharges it into the outlet of the Sheldon fan. This fan has a rated capacity of 200 ft³/min.

The air from the north and south ducts enters the 2-1/2 foot reactor supply duct which forms a circle of 23 feet outside diameter beneath the reactor structure. The air is drawn up through the reactor between the cast iron side thermal shields and between the outer shield and the inner surface of the biological shield. When the air reaches the lower side of the bottom upper thermal shield it flows across towards the center of the reactor to the J-rod annulus. The air then flows down the J-rod annulus and the graphite reflector to a ring-shaped

collector duct concentric with and directly below the intake header duct mentioned above (See Fig. 3).

The lower header room is vented to this exhaust duct by a one-foot-square dampered opening whenever anyone enters that area.

The duct from the reactor to the filter house, Building No. 101X, varies from a four by five foot concrete duct to a five-foot-diameter duct when it enters the filter house.

The flow, temperature and radiation level of the air is measured as it leaves the reactor hall. The normal outlet flow is about 16,000 ft³/min at a temperature of 135°F.

5. Air Filter System

Building 101 and 101X house the exhaust fans and the effluent air filter system. The air passes through spun-glass filters and 12-inch-thick Cambridge absolute paper filters. The air exhaust fans, called "C" and "D" fans, are driven by 100 hp motors and each have capacities of 18,000 ft³/min. One fan is used as standby and starts automatically should the other fail. A steam-operated fan can be used if necessary during a power failure. There are water seals on either side of the fans to isolate them so that maintenance work can be done.

The NRX effluent-air system is normally connected with the NRU reactor air system which has booster fans blowing the air to a high stack one mile from the reactors.

The activity of the air is monitored as it leaves the stack. In addition eight monitors scattered throughout the project area measure any radioactive fallout from the reactors.

IV. FUEL RODS

1. General

The standard fuel element for the NRX reactor is the natural-uranium metal rod which has in the past, been called an X-rod. This name came about because when the AECL project started in 1944 the word "uranium" was classified. Uranium was thus called "X-metal". The NRX natural uranium metal fuel rods came to be called X-rods.

The standard "clean cold" loading of NRX consists of 192 fresh Type-5 uranium metal rods. Fig. 11 and 12 show the general arrangement of the fuel rods. These diagrams are of a Type 5 uranium metal rod. The development of the uranium metal rod has resulted in six different types of rods being used at various times. Not all of these variations will be discussed here. Report No. ED-65 by J.W. Gosnell discusses all the different types of uranium metal rods that have been installed in NRX up to 1957.

Table No. 1 gives the dimensions, etc. of the uranium metal rods commonly in use in NRX at the present time. Types 6, 6A and 6B are equivalent, from the nuclear standpoint, to types 5, 5A and 5B respectively.

A complete description of the uranium metal rod can be found in the Reactor Handbook IOI-225 and in the NRX Design Manual IOI-47, Section A-12.

2. Components and Rod Assembly

As can be seen in Fig. 11, cooling water enters the rod through the header valve which is bolted to the cross water header. The upper valve section of the rod fits into a cylinder that is part of the header valve. O-rings at the top and bottom of a lantern ring make a water tight seal against the cylinder when they are compressed by an upper threaded ring nut. Water enters through an inner and outer lantern ring and flows down the rod.

The water then flows through the support spider ring shown in Fig. 12 from which the rod inner assembly is suspended. The rod itself is supported from the master plate. The cooling water flows past the fuel, through the aluminum lower rod section and out through a lantern ring and lower valve similar to the top valve assembly.

The top shielding plug, made of two aluminum rods screwed together, serves to prevent radiation from streaming up through the upper reactor shields. The upper rod is 6-1/2 feet long and 1.6 inches in diameter, the lower rod is 3 feet 11 inches long and 1-1/4 inches in diameter.

The uranium metal section, as can be seen in Table No. 1, is 120-1/2 inches long, 1.360 inches in diameter and weighs 120 pounds. The natural-uranium-metal is sheathed in a 0.079-inch-thick aluminum sheath which has three equally spaced 0.057-inch-high longitudinal fins designed to keep the element central in the outer sheath as shown in Fig. 12. The Type 5A, 5B, 6A and 6B rods have 0.040 inch thick inner sheaths. Fins on the 5B and 6B rods are 0.097 inch high since these rods have a 0.110 inch thick water annulus while the other rods have 0.070 inch thick water annuli. The B type rods are composite rods which are used in the Canada India Reactor (CIR) where the rod cooling water operates at a high temperature. They were used for a short time in NRX when a change from Type 5 rods to Type 5A was contemplated in 1956. The B type rods use the inner sheath of a type 5A and the outer sheath of a type 5 as can be seen from Table No. 1.

The type 6 rod has internal threads on the ends of the uranium metal section while the Type 5A and 5B has external threads as shown in Fig. 12. In both cases the aluminum

Item	Solid UO ₂ Rods	Annular UO ₂ Rods	NATURAL URANIUM METAL RODS					
			Type 5	Type 5A	Type 5B	Type 6	Type 6A	Type 6B
<u>Outer Sheath</u>								
Material	65S. Al	65S. Al	1S. Al	1S. Al	1S. Al	996A-F. Al	996A-F. Al	996A-F. Al
O.D. ins.	1.795 ⁶⁶²	1.795	1.740	1.660	1.740	1.740	1.660	1.740
I.D. ins.	1. 662	1.662	1.660	1.580	1.660	1.660	1.580	1.660
Wall ins.	0.065	0.065	0.040	0.040	0.040	0.040	0.040	0.040
<u>Inner Coolant Tube</u>								
Material		57S. Al						
I.D. ins.	1.420	0.500						
O.D. ins.	1.520	0.598						
<u>Inner Sheath</u>								
Material	57S. Al	57S. Al	1S. Al	1S. Al	1S. Al	996A-F. Al	996A-F. Al	996A-F. Al
O.D. ins.	1.510	1.520	1.518	1.440	1.440	1.518	1.440	1.440
I.D. ins.	1.410	1.42	1.360	1.360	1.360	1.360	1.360	1.360
Wall ins.	0.050	0.050	0.079	0.040	0.040	0.079	0.040	0.040
Fin Height ins.	0.057	0.057	0.058	0.057	0.097	0.058	0.057	0.097
Water Annulus	0. 071	0. 071	0.071	0.070	0.110	0.071	0.070	0.110
<u>Uranium Section</u>								
Material	UO ₂	UO ₂	Nat. U	Nat. U	Nat. U	Nat. U	Nat. U	Nat. U
Weight Nat. U. Kg	28	23.1	54	54	54	54	54	54
Weight UO ₂ Kg	32	26.4						
Diameter in.	1. 360 41	1.41	1.360	1.360	1.360	1.360	1.360	1.360
Length in.			120.5	120.5	120.5	120.5	120.5	120.5
No. of Pellets	148	136						
Pellet Height in.	0.750	0.880						
End Threads			Male	Male	Male	Female	Female	Female
Pellet I.D. in.		0.600						
<u>Expansion Space in.</u>	8-19/32	8-19/32	7½	7½	7½	8-19/32	8-19/32	8-19/32
<u>Drawing No.</u>	E-7212A-1	E-7223-1	EE-1599A	E-1599M	E-1599Q	E-2900A-1	E-2910A-1	E-4075A-1

NRX FUEL ROD DATA CHART

TABLE NO. 1

end plugs to which the sheath is welded screw on to the uranium section.

The outer rod sheath in the calandria section for all rods is 0.040 inches thick aluminum. There is an insulating air gap between the outer sheath and the calandria tube with the rod hanging freely in the calandria tube.

The uranium fuel section is hung from the upper shielding plug by a tension member which is designed to break at from 600 to 900 pounds load. The tension member protects the outer rod sheath and the calandria tube from damage should the uranium fuel swell during irradiation to the extent that the inner sheath grips the outer sheath. Thermal cycles during reactor operation could cause the outer sheath to buckle longitudinally. Before the sheath could contact the calandria, however, the tension member would break and the rod would drop into the restriction below the expansion space shown in Fig. 12. The cooling water must then flow through a small hole in the lower inner sheath plug. The water pressure at the base of the rod which is monitored continuously by the Budenberg gauge system would drop immediately and shut down the reactor. In this case the rod would be replaced. The reactor must be shutdown when a fuel rod is changed.

Hansen quick connect-disconnect valve arrangements are used to connect hoses to the top and bottom of the rod when it is removed from the reactor. A small water flow is kept on irradiated rods at all times to remove decay heat while they are in the vertical travelling flask. (See Section IV-D).

3. Uranium Metal Rod Operating Conditions

The average cooling water flow through each uranium metal rod is about 18 Igpm. The temperature rise across a single rod varies from about 30°C to 75°C depending on its position in the reactor lattice. The bulk coolant temperature rise through the reactor is normally 40°C at 42 MW with a total flow of about 3,000 Igpm.

The maximum allowable heat output of uranium metal rods is 400 kW at 267 cm of heavy-water depth. The following temperature limits are also set:

- Center of Uranium = 668°C maximum (to prevent change from α -phase to β -phase)
- Aluminum/Uranium Interface = 200°C maximum (To prevent formation of UAl_3 alloy which weakens the rod sheath)
- Aluminum Surface = 130°C maximum (to prevent intergranular corrosion by cooling water)

Maximum outlet water temperature for any rod = 95°C

The scheduled burnup of the uranium metal fuel rods is 1300Mwd/ton equivalent to 78 MWd per rod. The irradiation takes from 8 to 15 months depending on the rod's position in the reactor. Rods are removed on schedule for the plutonium produced unless an inner sheath rupture or other fault occurs first. Uranium metal rods may also be removed before they have received their scheduled irradiation if the coolant flow drops to about 11 Igpm (equivalent to 56 lb/in² outlet pressure as indicated by the Budenberg gauges). This drop in flow would, in general, be caused by swelling of the uranium metal which reduces the size of the water annulus.

Any change in length of the uranium fuel section is measured by X-ray examination of each uranium metal fuel rod during its life in the reactor. All rods are X-rayed when their irradiation is completed. A rod may also be X-rayed up to three or four times during irradiation depending on the metallurgical treatment given the rod during fabrication. The X-ray equipment is located on top of the rod storage block. Rods are removed from the reactor and X-rayed during scheduled shutdowns. X-ray photographs may be taken for record purposes if required. Normally the fluoroscope unit is used.

When a rod has completed its irradiation it is removed from the reactor to the underwater trench system which is filled with de-ionized water. With underwater hack saws the rod is cut above and below the fuel section at the "Cut lines" shown in Fig. 12. The top and bottom shielding sections are saved and used for the assembly of other rods. The outer sheath is removed and the fuel, still enclosed in the inner sheath, is stored vertically in underwater bays until the rods are delivered to the United States for fuel processing.

B. Booster Rods

1. Enriched Uranium Aluminum Alloy Rods

The NRX reactor has a very large experimental load (which will be discussed later). Experimental rods replace uranium metal rods in the core, and reactivity thus lost to the reactor is made up by installing "booster rods" of enriched uranium/aluminum alloy. At the time of writing (July 1961) there are thirty of these rods in the reactor providing about 60 milli-k* of reactivity in excess of that provided by the uranium metal rods they are replacing.

The fuel in the booster rods is made up of ten eight-inch-long uranium/aluminum alloy slugs 1.36 inches in outside diameter separated by 1-inch-long aluminum slugs. The ten

* Reactivity in NRX is measured in milli-k (mk). One mk of positive reactivity added to a critical reactor will increase the multiplication factor "k" from 1.000 to 1.001.

slugs altogether contain approximately 446 grams of uranium of which 93% is U-235. These slugs are fitted inside a standard uranium metal fuel rod sheath, the full length being made up by a 9-7/8-inch aluminum plug at the top and a 26-3/8-inch-long plug at the bottom.

The rod itself is assembled with the same parts as used in the Type 5 and 6 uranium metal rods.

A booster rod provides 5.3 milli-k of reactivity in excess of a uranium metal fuel rod in the center of the core when fresh. The rods are removed at 144 MWd rod output by which time the excess reactivity is 1.0 milli-k; this takes, on the average, about 18 months irradiation. They are usually installed in the outer circles when new in order to not overheat the uranium metal fuel rods surrounding them.

When these rods are burned out they are cut into slugs after a six month cooling period and shipped to the United States for processing.

2. Plutonium/Aluminum Alloy Rods

At various times since 1951 three different designs of plutonium/aluminum rods have been used for booster rods.

1. Three rods were made by casting the alloy into aluminum cans 1 inch internal diameter and 1.36 inch outside diameter. These were screwed together and put inside a standard uranium metal fuel rod sheath. The nominal composition of the alloy was varied from 0.50 weight per cent in the two end cans to 20 weight per cent in the center to approximate a sine-squared distribution of approximately 340 grams of plutonium along the length. (See AECL-601 by O.J.C. Runnalls).

These three rods operated satisfactorily, the last rod of this type being removed from the reactor in December, 1955.

2. In order to conserve plutonium, a composite rod was designed made of a 20 weight per cent plutonium/aluminum central section of three slugs containing a total of 180 grams of plutonium and two detachable natural-uranium metal ends. The uranium ends were about 45 inches long and of standard fuel rod construction.

Twenty-one of these rods were irradiated. Unfortunately troubles were experienced during irradiation. Some of the slugs separated from the inner cans and other bulged the sheath. Five of the rods had to be removed because of excessive pressure drop. A slug in the last rod to be irradiated, No. 21, melted in July, 1955 and plutonium alloy penetrated the outer sheaths and the calandria tube.

The calandria tube was replaced in situ in seven weeks.

All these rods were then removed from the reactor and a new design was developed.

3. The latest design of plutonium rod was made up of twelve slugs each nine inches long containing 3.7 weight percent of plutonium machined to fit inside a standard fuel rod sheath. They were assembled like Type 5 uranium metal fuel rods.

The rod was 108 inches long and contained a total of 265 grams of plutonium. Twenty-seven of these rods were irradiated without incident, the last one being removed from the reactor in July, 1960. These rods are also described by O.J.C. Runnalls in AECL-601.

All enrichment in the reactor is now obtained from the U-235 alloy rods noted above.

C. UO₂ Rods

A program was initiated in 1958 to replace all uranium metal fuel rods with uranium dioxide fuel rods. The main reasons for making the change were to gain experience with UO₂ fuel, to replace the uranium metal rods with a more reliable fuel element and to increase the flux in NRX by up to 50% without an increase in reactor power.

Solid UO₂ rods will be used in the outer circles of the reactor such that their maximum heat output will be 180 kW. In order to prevent center melting of the fuel, annular UO₂ rods will be installed in the inner circles.

The dimensions of the solid rods are given in Table No.1. The fuel section is made up of 3/4-inch-high pellets ~~1.41~~ 1.41 inches in diameter. The fuel weighs 70 pounds. The inner sheath is 0.050 inches thick while the outer sheath is 0.065 inches thick.

As can be seen from Table No.1 the inner sheath of the UO₂ fuel rods is made of rather ~~annealed~~ ^{annealed} 57S aluminum alloy while the outer sheath is made of ~~strong~~ ^{strong} 65S aluminum. These two alloys are used to minimize the possibility of damage to the reactor should a UO₂ rod become defected. Water would enter the defect during reactor shut down, turn to steam at startup, and could explosively rupture the sheaths. It is hoped that the ~~brittle~~ ^{annealed} inner sheath will rupture before the steam pressure is too high while the ~~ductile~~ ^{strong} outer sheath will protect the calandria tube from damage. The rod is assembled essentially the same as standard fuel rods with slight differences for ease of fabrication.

Forty-five solid UO₂ rods are in the reactor at the time of writing. No trouble has yet been experienced with the rods.

Since UO₂ has a low thermal conductivity, large-diameter solid rods would melt in the center during irradiation in the central region of the NRX core. Annular UO₂ rods have, therefore, been designed with water cooling on both the inner and outer surfaces. The rods are made up of annular pellets 7/8 inches high, 1.41 inches in outside diameter having a 0.600-inch-diameter hole. The rod contains 58 pounds of UO₂. The 57S aluminum inner coolant tube is 1/2 inches inner diameter and 0.05 inches thick. The inner sheath of 57S aluminum is 1-7/16 inches inner diameter and 0.05 inches thick. The outer sheath of 65S aluminum is 1-~~6~~/~~8~~ inches inner diameter and 1/16 inches thick. The rest of the rod has essentially the same components as a standard fuel rod.

These rods have operated satisfactorily at heat outputs up to 350 kW.

It is expected that large scale installation of annular UO₂ rods will commence early in 1962.

In order to compensate for the loss of reactivity from the UO₂ rods as compared to the uranium metal fuel rods, additional booster rods will have to be installed. In order to cool the booster rods safely when the reactor flux has been increased by 50% a new design of booster rod is being developed. The most promising design at the present time is a seven-element U-235/aluminum alloy rod. The individual elements will be 1/4 inches in diameter, about 8 feet long, with a coextruded three finned 0.025-inch-thick aluminum sheath. The seven pencils will be held by spacer plates inside a 1.39 inch diameter outer aluminum flow tube.

It is expected that these rods will require a coolant flow of about 40 Igpm. At the time of writing the requirements for increased reactor coolant pumping capacity and/or redesign of the standard rod orifice section is being investigated.

D. The Fuel-Rod Removal Flask

The vertical travelling flask is shown in Fig. 2 and 3. A complete description of the flask is given in the NRX Design Manual No. A-35-5.

The flask is used for handling all types of fuel rods, isotope tray rods, and those special assemblies that are approximately the size of the NRX fuel rods. It is provided with auxiliary equipment in order to install and remove tray-rod isotope samples and loop fuel specimens. (See Section VI-A and VI-B).

As can be seen from the figures, the flask is held by a trolley which travels in a north-south direction, the trolley is in turn supported on a gantry which travels in an east-west direction on steel rails from the top of the reactor to the vertical rod storage block. In addition three parallel sets of north-south tracks enable the flask to service the vertical storage block as shown in Fig. 2.

The flask consists of a 3-1/4-inch-diameter stainless steel tube approximately 33-1/2 feet long encased in lead shielding. The lead shielding is 9-1/2 inches thick for about 21 feet from the base, tapering to three inches over the next five feet. The upper seven feet of the flask has no lead shielding since only the upper unirradiated part of an assembly is ever in this area. The lead shielding is sheathed along its entire length in a casing of 1/2 inch thick mild steel plate. This main flask assembly is made up of seven individual sections which are flanged and bolted together to facilitate dismantling the flask for repairs and modifications. The total weight of the flask itself is about 25 tons.

A donut shaped lead shield called the skirt is located around the bottom of the flask. It is 16 inches high, 46 inches outer diameter, 24 inches inner diameter and weighs 7800 pounds. During rod removal the skirt is lowered to provide shielding between the bottom of the flask and another shielding donut which rests on the revolving floor. The skirt is supported by three cables and is raised and lowered by an electric motor. Interlocks prevent the skirt being moved if this action would expose the radioactive portion of a normal length rod. Personnel are thereby protected from being irradiated by the central fuel section.

The flask has a 3-1/2-inch-diameter mild steel tube 33-1/2 feet long attached to the outside. This transfer tube is used to hold unirradiated rods which are to be installed in the reactor. In this way trips back and forth to the storage block while replacing an irradiated rod with a fresh one are not necessary. The tube is equipped with its own cable and hand winch.

The flask is provided with openings at various levels which are used to accommodate special components of the flask - eg. those used for isotope handling.

The flask is provided with hose reels which can be connected to rods being removed from the reactor. A small water flow removes the decay heat from the irradiated rods. The water is filtered as it leaves the rod to prevent the spread of radioactive contamination should the rod be ruptured.

The flask also has its own air exhaust and filter system. If a ruptured rod is being removed from the reactor, air contamination in the reactor hall from the rod is reduced by having air drawn past the rod through the flask air filter and

directly into the reactor air exhaust system.

There is an electrical and mechanical interlock system to ensure that the proper sequence of operation is followed during rod handling in order to prevent accidental irradiation of the working crew.

As mentioned previously, the flask is supported on a trolley assembly which travels in a north-south direction. The trolley travels on four wheels with a seven-foot wheelbase and is driven by a one hp electric motor through suitable gearing. The trolley assembly has two eight-foot-square platforms which hold the flask in place. The hand winches used to raise and lower rods and special assemblies are located on these platforms.

The trolley is supported by a gantry which runs on rails from the storage block to the top of the reactor. The gantry span is 16 feet, its wheelbase is 10 feet. It is driven by a 3 hp electric motor at a maximum speed of 28.3 feet per minute.

The reactor hall overhead service crane is too low to pass over the full height of the flask. For this reason spotlights on the flask actuate a photoelectric cell alarm system which sets off a warning buzzer in the crane cab if it gets within 25 feet of the flask. A hydraulic cylinder has been installed on the flask which tilts the upper four feet of the flask allowing the crane to pass over if such is desired.

V. CONTROL SYSTEM

A. Introduction

The purpose of the control system is to provide safe conditions during the startup, steady power and shutdown states of the reactor.

The NRX control system has progressed through several modifications since the original design. Report No. IOI-134-8 by H.E. Smyth discusses the changes to the control system from 1947 to 1958.

Mark I System (1947)

18 Shut-Off Rods - raised in two separate banks of nine each

4 Control Rods - manual operation

Weir Box - manual operation

Automatic partial heavy-water dump

Soon after startup in 1947 three of the control rods and the automatic dump were eliminated. As mentioned previously in Section III-B-6 the weir box was used as a coarse control and one control rod was normally used as a fine control of power and reactivity.

Mark II System (September, 1948 to December, 1952)

12 Shut-Off Rods - raised in banks of 4,3,2,1,1,1
 1 Control Rod - manual or automatic operation
 Weir Box - manual operation

This system was used until the accident to NRX in December 12, 1952.

On December 12, 1952 an uncontrolled power increase took place during low-power measurements on air-cooled uranium metal rods. When the reactor tripped some of the shut-off rods did not fall fully into the core. The calandria and some of the fuel rods were seriously damaged by overheating. The calandria was replaced in 14 months.

The reactor was started up in February, 1954 with the Mark III system.

Mark III System (February 1954 to 1956)

18 Shut-Off Rods - raised individually
 1 Control Rod - manual or automatic operation
 Weir Box - manual operation
 Automatic heavy-water dump (to half full)
 - under certain conditions (e.g. failure of all shut-off rods to fall)

After the accident in 1952 Montreal Engineering Company was asked to do an independent evaluation of the NRX Control System. They recommended a new design of shut-off rod and a backup shutdown device to compliment the shut-off rods. Therefore, the heavy-water dump was incorporated in the Mark III system as a backup shutdown device.

The American Machine and Foundry Company was given a contract to design and build a new type of shut-off rod.

The Mark IV control system was developed to consist of six of the new shut-off rods, automatic heavy-water dump, automatic heavy-water level control and no control rod. This system was installed in two stages in 1956 and 1958. The reactor is now controlled by this system which will be described below.

A complete description of the control system can be found in the NRX Design Manual IOI-21 (Rev.2) Section A-7. The general specifications of the system can be found in Report No. IOI-60.

B. System Components

1. Shut-Off Rods

The shut-off rod for the Mark IV control system was designed and built by the American Machine and Foundry Company.

The original shut-off rods had been designed with the philosophy that if the control system is to be able to shut down the reactor under all possible conditions which may arise, the reaction must be stopped in 0.5 seconds. This meant that the shut-off rods had to be accelerated into the reactor. Air pressure was used and the tolerance required in the mechanism lead to many malfunctions.

It was decided, following the accident in 1952, that if the reaction was stopped in one second most conceivable accidents could still be contained safely. With this new philosophy the shut-off rods could be allowed to drop into the reactor under gravity.

During reactor operation the rods are suspended above the core; when the reactor trips the rods are released to shutdown the reactor. During shut-downs, four shut-off rods, called the "safety bank", are held above the core. Should a reactivity or power surge occur during the shutdown the safety bank will be released by the tripping circuit and would contain the excursion. The rods can only be raised individually in a prescribed sequence in order to guard against local, unseen, flux increases that might be caused by raising adjacent rods. If the sequence is contravened the interlock system ~~will trip the reactor~~, prevents raising further rods.

The AMF shut-off rod absorber section is suspended freely from a 3/32-inch-diameter stainless steel cable 10 feet 9 inches long attached to a cylindrical drum. The drum is driven by an electric motor through a self-locking worm gear and magnetic disc clutch. The cable winds on the drum when the rod is being raised and unwinds when the rod is lowered. When the rod is raised

it compresses a spring during the last three inches of travel. On a reactor trip the magnetic clutch is de-energized and the compressed spring gives a small initial acceleration to the rod. The rods must fall half way into the reactor within 0.7 seconds from release. An unsafe fault annunciates if all shut-off rods are not fully down and all heavy-water dump valves not fully open within two seconds following a reactor trip.

The absorber section consists of 11.4 pounds of sintered boron carbide pellets inside a 1-1/4 inch outside diameter stainless steel tube 10 feet 8 inches long.

The headgear assembly which contains the rod motor and gears as described above can be used interchangeably on any of the shut-off rods.

The shut-off rod top shield assembly consists of a lead-filled stainless steel tube 1-1/2-inches in outside diameter and 15-1/2-inches long with a central hole for the lifting cable.

The upper and lower sections of the outer tube for the shut-off rod is made of stainless steel pipe of various diameters and lengths. The section in the calandria is made of a 12-1/2 foot long aluminum tube 1-1/2-inches in outside diameter and 1/16 inches thick.

A mechanical snubber assembly situated within the outer tube at an elevation corresponding to the top of the calandria serves to cushion the shock of the absorber section when it drops due to a broken cable or other malfunction. A hydraulic snubber in the headgear decelerates the absorber section during a normal rod fall.

There are limit switches in the headgear which indicate upper and lower travel limits of the rod and operate contacts in the reactor control system.

Cooling for the absorber section is provided by process air at 11 ft³/min to each headgear discharging through ports at the base of the rod into the reactor exhaust system.

These shut-off rods have operated very satisfactorily once initial trouble with the lower limit switches was solved. In five years only on three occasions has a rod failed to fall fully into the reactor when required. On each occasion the reactor was shut down safely since the remaining five rods fell in fully as normal.

2. Control and Dump Valves

These components were discussed in Section III-B-9.

C. Reactor Control

1. Triplicated Control Circuits

As discussed in Section III-B-9 the reactor power is controlled by variation of the heavy-water level. Heavy-water is pumped into the calandria at the rate of 215 Igpm and flows out at a variable rate regulated by three control valves on the calandria outlet lines.

The control valves are positioned by separate identical electronic circuits and electro-pneumatic controls called Channels A, B and C. A simplified schematic diagram of a single channel is shown in Fig. 13. The three channels operate as a two out of three coincident tripping circuit which means that two of the channels must annunciate a fault at the same time in order to shut-down the reactor. Thus unnecessary reactor trips because of component failure in one channel will not occur. In addition any single channel component can be tested, checked and maintained on a routine basis while the reactor is operating. In addition to trip levels on rates of power change, etc. for each channel, disagreement between all three channels will trip the reactor. As discussed below, if one channel disagrees with the other two it is rejected and a single-channel fault annunciates. This would not shut the reactor down. The equipment in that channel would be repaired as required while the reactor was operating.

2. Automatic Reactor Control by Neutron Power

A change in reactor power produces a proportional change in the neutron signal from the ion chambers in the North Thermal Column. Two ion chambers are connected to each channel making a total of six ion chambers for the three channels. As can be seen in Fig. 13, one ion chamber sends its neutron signal to the log amplifier which gives signals proportional to the log (of the) power and the log (of the) rate of power change and the other sends its signals to the linear power and linear rate amplifiers. Each channel will receive a set of four signals that is slightly different from the respective signals from the other two channels because of differences in the neutron flux reaching the various chambers.

The respective signals from the three channels are compared in four comparator units as shown in Fig. 13. Each comparator unit averages the incoming signals and gives out three signals which are the average of the three incoming signals. These average signals are used

as the actuating signals for the rest of each separate channel. If, because of a malfunction of an ion chamber or amplifier in a single channel, one of the input signals to a particular comparator does not agree within specified limits when compared to the others it is rejected and only the remaining two signals are averaged and used to give out three signals to the function generator. A single-channel fault would annunciate and the channel would be repaired.

From the comparators, we have, for each channel, four signals which are proportional to average values of the ion chamber signals, namely linear rate, linear power, log rate and log power. These four parameters are used to control the reactor through the three separate electronic channels.

The signals enter a function generator unit where they are compared to a fixed reference signal which comes from the power demand set point unit on the reactor control console. Any disagreement produces an error signal which changes the settings of the heavy water control valves, thus changing the moderator level in the calandria in such a manner that the error signal becomes zero when the required power level is reached. The reactor power then remains constant at the level demanded by the set point unit at the control console.

A valve position comparator as shown in Fig. 13 compares the positions of the three valves as regulated by their respective channels. If one valve position differs more than 30% from the other two that valve and its associated channel is rejected from the control system. The rejected valve will go to neutral, that is to the half closed position. Reactor control will continue, with a slight power-change, by the use of the remaining two channels and their associated valves.

The four signals of log rate, log power, linear rate and linear power that enter the function generator are each used to control the reactor at various power levels as shown in Fig. 14 and as discussed below.

(a) The Logarithmic Rate Control Circuit

This circuit controls the exponential rate at which the reactor power increases from subcritical levels up to approximately 20 MW. It also prevents overshoots in power whenever the demanded power level is changed between 4 MW and 20 MW. The rate of power increase is limited to 1% per second. At 20 MW the instantaneous maximum rate is, therefore, 0.2 MW per second.

(b) Linear Rate Control

This circuit limits the maximum rate of power increase to 0.2 MW per second at any reactor power above 20 MW and prevents power overshoots when the demanded level is attained.

(c) Linear Power Control Circuit

At any power between 4 MW and 40 MW this circuit levels off the reactor power at the demand level and keeps it steady at this level. This circuit constitutes the main controlling agent of the reactor power during steady state operation.

(d) Logarithmic Power Control Circuit

This circuit levels the reactor power at 1% of maximum power (approximately 400 kW) and keeps the power steady at this level. This circuit performs its function only when the control system has been switched to the "Low Power State" at the power set unit at the reactor control console. The reactor power cannot be raised above 1% when in this state. As will be explained later the reactor power can be increased to 1% without clearing all the tripping circuits in order to save startup time when the circuits are finally clear. Normally one goes immediately to "Normal Power State" and the log power circuit is not used.

3. Automatic Thermal Power Control

The temperature rise in the reactor bulk cooling water can be used to keep the reactor thermal power steady when the neutron flux varies because of changes in, for example, heavy water height. A circuit has been designed which will keep the thermal power constant and will level off the thermal power at the demanded level. The neutron control system overrides this thermal control unit during large changes in demanded level.

The thermal control unit operates at power levels between 5 MW and 40 MW. It may be switched out of service if desired.

At the time of writing (July 1961) this equipment has not yet operated satisfactorily and it is being modified.

D. Reactor Trip and Alarm Circuits

1. Power Monitoring Circuit

There are two main trip circuits which monitor the power of the reactor and shut it down if preset limits are exceeded:

(a) The reactor mean power trip circuit receives its signals from the linear power comparator of the respective channels as shown in Fig. 13. It automatically shuts down the reactor when the power exceeds ~~3%~~^{7.5%} of the demanded power level thus protecting the reactor from power excursions. This circuit protects the reactor at any power from 4 MW to 40 MW.

(b) Auxiliary Absolute Trip Circuit

(i) Overpower Trip Circuit

This circuit is completely separate from the other circuits in the reactor control and protective system. The three channel-overpower trip signals are taken from three independent ion chambers situated in horizontal experimental holes around the reactor. The trip levels are set manually at 10% above the operating neutron level. During reactor shutdown the trip level is set at 10% of full scale.

(ii) Trip on Loss of Main Coolant Flow

Signals are taken from the cooling-water pressure drop across the reactor, which is proportional to the coolant flow through the reactor. The trip level on Channel A is set 10 lb/in² above and below the operating pressure while Channel B and C limits are set 35 lb/in² above and 25 lb/in² below the operating pressure. In general, these settings do not have to be changed during operation.

2. Reactor Systems Monitoring Circuit

A large number of devices continuously monitor temperature, pressure, height, flow and neutron and gamma-ray flux conditions of the various systems associated with reactor operation. Examples are the Budenberg gauge system which monitors the coolant pressure at the base of each fuel rod and the radiation monitors throughout the reactor building.

A fault signal initiated by any one of these devices will cause the condition to annunciate in the reactor control room either as a trip or an alarm fault, depending on the importance of the condition being monitored.

Signals that are trips automatically shutdown the reactor. The six shut-off rods are released, dropping halfway into the reactor within 0.7 seconds from initiation of the release. The heavy-water moderator is dumped to 140 cm within three minutes forming the backup shutdown device. Trips are divided into two classes, absolute trips being those that automatically shut down the reactor at any power level and conditional trips that shut down the reactor only when the power level is above 1% of maximum power.

Thus when the ~~conditional~~ ^{absolute} trips have been cleared the reactor can be started up to the "Low Power" state as outlined in Section V-C-2-d.

3. Sequence Control Circuits

This system prevents the reactor from being started incorrectly or when an unsafe condition exists. The circuits are arranged to permit the reactor to be started in the following sequence only:

- (1) all absolute trips must be clear
- (2) energize primary circuits manually
- (3) raise shut-off rods in the correct sequence
- (4) dump and control valves close when the last shut-off rod is up.
- (5) the heavy water circulating pump can be started
- (6) all conditional trips must be clear if reactor power is to be increased above 1% of full power
- (7) set the power demand unit for the desired power level. The reactor power will increase at the controlled rate automatically and level off at the demanded level as shown in Fig. 14.

If this sequence is contravened at any time interlocks will trip the reactor back to the shutdown state.

VI. EXPERIMENTAL ASSEMBLIES IN THE NRX REACTOR CORE

General

Many different experimental assemblies have been installed in the NRX core since operation commenced in 1947. Some of these, such as the loop experiments and the pneumatic carrier, are permanent equipment while others such as the UO₂ pellet rods are installed for particular tests only.

Those assemblies that are installed in fuel rod positions are limited in outside diameter to that which will pass freely through the 2-1/8 inch holes in the upper thermal shields. Assemblies in the central thimble must pass freely through a 5-1/2 inch diameter hole.

A. Loop Facilities

1. Purpose

The term loop refers to the high-temperature, high-pressure, test facilities for reactor fuels in which the heat produced in the fuel under test is removed by a recirculating coolant. Hence the term "loop" since the piping connecting the in-reactor test section to the out-of-reactor equipment (pumps, coolers, etc.) forms a closed loop. A simplified flow sheet of a typical loop is shown in Fig. 15.

The loop experiments are used to test possible power reactor fuels such as natural and enriched uranium metal, uranium dioxide and uranium alloys. Various cladding materials have been investigated such as aluminum, zircaloy and stainless steel.

Some of the information required is:

1. The dimensional stability of the fuels under irradiation at power reactor conditions.
2. The corrosion resistance of the cladding materials.
3. The effectiveness of fabrication techniques.
4. The nature of the radiation signal from a defected fuel element.

Although NRX was not designed to accommodate loops, shielded rooms and steel platforms have been built in the reactor hall to hold the necessary out of the pile equipment.

A complete history and description of loop development in NRX is given in AECL-1095 by F.A. McIntosh and AECL-1273 by R.O. Sochaski. Detailed descriptions of each loop can be obtained from the NRX Operating and Design manuals for the particular loop.

2. Loop Equipment

(a) X-1, X-2, X-3 and X-6 Loops

These four loops were installed in fuel rod positions in the core in 1954 and 1955. The first three loops were built by the United States

under an agreement by which AECL would use one for their own experiments. The X-6 loop was installed in a cooperative arrangement between AECL and the United States.

The loops are designed to operate at about 2000 lb/in² and 500°F. The temperature rise across a normal fuel charge is about 25°F. Design heat outputs vary from 50kW to 100kW.

A simplified flow sheet of these loops is shown in Fig. 15. X-1, X-2, X-3 loops are of stainless steel piping throughout while the X-6 loop has carbon steel piping with a stainless steel in-reactor pressure tube. The four loops have a common Dowtherm cooling system and a common water make-up system.

As can be seen in Fig. 15 the test fuel is placed in the reactor by inserting in a normal fuel rod position a stainless steel pressure tube. The standard reusable pressure tube is shown in Fig. 16. The in-calandria section is a stainless steel tube 1-1/8 inches in outside diameter and 0.100 inches thick. The cooling water enters at the bottom and leaves at the top as shown except for the X-3 loop in which the flow is downwards. A "Unibolt" nut is used to close the loop, the fuel elements being suspended from a blanking plate held by the nut. The calandria tube is protected from the high loop coolant temperature by a gas annulus divided by a layer of aluminum foil as indicated in Fig. 16.

The fuel elements are usually hung together to form a complete charge of from eight to ten specimens. The normal length of each specimen is about seven inches while the diameter is limited to about 0.8 inches. Thermocouples are sometimes attached to the elements.

The main loop piping is of 1 inch stainless steel. Each loop has two centrifugal canned rotor pumps rated at 56 Igpm at a 320 foot head each driven by 15 hp motors. One pump operates while the second is on standby to start automatically upon failure of the normal pump.

The loop heaters automatically control the loop water temperature at the inlet to the test section, usually at about 500°F. They are composed of twelve electric immersion heater elements of 3.5 kW each. The heaters are cast into a matrix of aluminum around a sixteen-turn, 9-7/8 inch-diameter helix of one-inch loop piping.

The loop cooler removes from the loop the heat produced by the test specimens. It is a double-pipe exchanger cooled by Dowtherm which is in turn cooled by

the high-pressure water system as noted in Section III-D-3.

The surge tank is used to maintain loop pressure and to allow for changes in volume of the total loop coolant. The coolant is circulated through the surge tank which collects dissolved gases that are then vented from the loop. The tank, made of 9/16-inch-thick stainless steel, is 4 feet ~~7 inches~~ high and 6-5/8 inches in outside diameter. It holds five gallons of water. Eight strip heaters of 2.2 kW each automatically maintain the loop pressure by boiling the water in the tank.

As can be seen from Fig.15, each loop has an ion-exchange and filter circuit which has a flow of about 0.1 Igpm. The purification cooler reduces the temperature of the water to 100°F in order to protect the ion-exchange resins. The ion-exchange columns are of stainless steel 2 feet 9 inches high and 3-1/2 inches in outside diameter. They contain mixed-bed lithium hydroxide resins and keep the pH of the loop at between 9 to 10.5 which inhibits corrosion in the loop and stops crud formation on the fuel elements. The points for sampling the loop coolant are on the purification circuit as shown.

The make-up system supplies pure distilled water to provide for losses due to leakage and sampling. In addition, should the loop water become badly contaminated from ruptured fuel elements, the make-up system can be used to add a decontaminant solution to clean the loop.

The gamma-ray monitor and delayed-neutron monitor detect ruptures in the test specimens from the radioactive fission products which would be released to the coolant. Sometimes holes are drilled through the sheaths of the test fuel specimens in order to ascertain how the fuel would act in a power reactor if the cladding ruptured.

The loop temperature is controlled by the loop heater, and the loop pressure is controlled by the surge-tank heater as shown in Fig. 15.

The loops are all connected electrically to the reactor protective trip system. For example, excessive loop temperature or a loop pump failure would annunciate on the single-fault trip circuit and would shutdown the reactor as outlined in Section No. V-D-2.

(b) The X-4 Loop

The original X-4 loop called the EEC or Leo loop was designed and fabricated by the English Electric Company for the United Kingdom Atomic Energy Authority. It was installed in NRX in September 1956.

The loop is constructed of mild steel and the loop cooler is a water boiler type cooled directly with water. The loop pumps are connected in series. Otherwise the loop was in general similar to the X-1, X-2, X-3 and X-6 loops. In 1960 it was decided to modify the loop to use steam cooling.

At the time of writing (July 1961) the X-4 steam cooled loop is undergoing out of reactor commissioning tests. A simplified flow diagram of the loop is shown in Fig. 17.

The steam cooled loop is divided into two sections, a water-circulating section and the steam-filled section. Eight hundred pounds of water per hour are turned to steam in the boiler. The steam temperature is raised to about 700°F in the superheater and enters the bottom of the in-reactor test section. Leaving the test section it first enters the out-of-reactor steam-cooled test section and then the boiler where the superheat is removed. It then enters the jet condenser where the steam is condensed by the loop water which then returns to the pumps.

Most of the equipment, aside from new loop heaters, the boiler and the superheater were used in the old loop.

(c) The X-5 Loop

As mentioned previously the first loop in NRX was installed in the central thimble in 1951.

The loop presently in the central thimble, X-5, differs from the smaller loops primarily in size. The primary loop piping is 2-1/2-inch stainless steel while the other loops have one-inch piping. The coolant flow is normally about 85 Igpm compared to 12 to 15 Igpm of the small loops. Test fuels in the loop have had heat outputs as high as 330 kW while 100 kW is about the maximum that a small loop can handle.

The X-5 pressure tube is made of Zircaloy-2, 4-1/2-inches in outside diameter and 3-1/4 inches internal diameter. As noted before the X-5 loop is a re-entrant type with the inlet and outlet primary coolant lines

entering the test section as concentric pipes. A flow-divider extends down inside the pressure tube. The primary coolant flows down the annulus between the flow divider and the pressure tube and up the inside of the flow divider tube past the fuel specimens. The X-5 loop has its own Dowtherm cooling system.

The X-5 loop has been used on a cooperative basis by AECL and Westinghouse Atomic Power Division (WAPD).

The large size of the irradiation space in the X-5 loop has enabled test irradiations to be done on NRU flat fuel rods and on small bundles of test fuel for the NPD and CANDU reactors.

(d) The X-7 Loop (Organic Loop II)

The first organic cooled test section was installed in 1959 in a horizontal experimental hole tangential to the calandria.

After a few months operation it was decided that a higher, more uniform neutron flux was required and after modifications the organic loop, X-7, was installed in the reactor core.

A simplified flow diagram is shown in Fig. 18.

The loop was designed for the use of organic coolants that are liquid at room temperature, but trace heating can be added for use with organics not liquid at room temperatures. In addition a 20kW induction heater has recently been installed on the main loop piping.

The pressure tube is a 1-1/2-inch outer diameter stainless steel pipe with a 0.049 inch thick wall. A cooling water and air annulus around the test section protects the calandria tube from the high loop coolant temperatures. The loop is designed to operate up to 800°F and 280 lb/in².

The main loop piping is made of carbon steel.

The main loop pump is a mechanically sealed centrifugal pump rated at 24 to 48 Igpm at 800°F. The standby loop pump, which is operated only when the reactor is shutdown, is rated at 20 Igpm at 800°F.

The loop heaters are rated at 60 kW. A nitrogen pressure system attached to the surge tank maintains the proper operating pressure on the loop.

Ortho and meta terphenyls containing 30% high boilers have been used as the coolant in the loop. The rest of the loop components are of about the same size and have the same purpose as for the other loops. The out-of-reactor

equipment is housed in a cement block enclosure on the reactor hall floor near the horizontal experimental holes.

B. Isotope Tray Rods

A large part of the NRX effort is devoted to the production of radioactive isotopes. Most of these isotopes are produced in tray rods. As mentioned previously, the self-serve positions also are used in isotope production.

The first isotope irradiations in the reactor core were done in an air cooled plug in the central thimble. Later tray rods were put in the core replacing fuel rods. At the time of writing (July 1961) fifteen tray rods are in the NRX core and one tray rod is in the J-rod annulus.

A simplified diagram of the tray rod is shown in Fig.19(A).

The upper outer sheath is made of an aluminum tube 2-1/4 inches outer diameter, and the calandria section is 1-3/4 inches outer diameter and 1-5/8 inches inner diameter, the same dimensions as a standard fuel rod outer sheath. A stainless steel shielding plug 30 inches long and 7/8 inches in diameter is located in the lower rod section. A wire screen at the bottom of the rod is held in place by the lower air hose connection piece which is made of stainless steel. The screen would catch and hold radioactive material that could be released from a ruptured capsule.

Cooling air at the rate of 25 standard cubic feet per minute enters through the inlet air hose, passes up past the samples, and out through holes in the outer sheath situated just above the elevation of the top of the calandria.

The section of rod which is removed to change samples is called the tray assembly. It consists of an upper mild steel sand-filled shielding section and the aluminum capsule section. The capsule section is made of an aluminum tube 10 feet long and 1-1/2 inches outer diameter from which fifteen slots 6-1/4 inches long and 1-1/4 inches apart are cut as shown in Fig. 19(A) and 19(B). Each slot is fitted with two spring steel clips which each hold a sample capsule.

The capsules, identical to the self-serve capsules, are of aluminum 1-3/4 inches high, 7/8 inches in outer diameter with a 1/16 inch thick wall as shown in Fig. 19(C). They are sealed by cold welding in a hydraulic press which evacuates the container and presses the cover on with a 10,000 lb/in² pressure.

Each rod can hold 30 samples. Eight of the fifteen rods in the reactor are completely filled with 30-gram cobalt samples. The cobalt is in the form of nickel plated

metal pellets 1 mm long and 1 mm in diameter weighing seven mg. NRX produces about 120,000 curies of Co-60 each year at a normal maximum specific activity of 50 curies/gram.

The seven remaining rods are used for miscellaneous commercial production isotopes as well as irradiations for fundamental research on the project and elsewhere. The reactor produces about 14,000 curies per year of such isotopes by the irradiation of samples ranging from antimony to zinc. Iridium, tellurium and gold form the bulk of this production.

The tray rods can be removed from the reactor and the samples changed without shutting down the reactor. Samples are removed by hand tongs with the rod in the travelling flask and are installed in lead flasks for shipment from the project.

There is one tray rod in the J-rod annulus which is used for research and production samples. It is cooled by the reactor cooling air which passes down through the annulus as described in Section III-E-4.

The flux in the tray rods ranges from 0.7×10^{13} to 6×10^{13} n/cm²/sec. Samples are limited in heat output to 50 watts calculated at the maximum flux. Materials which might decompose under irradiation or might damage the reactor cannot be irradiated.

C. Pneumatic Carrier

In 1949 a pneumatic carrier tube was constructed to carry irradiation samples from the Research Chemistry Building directly to a central fuel-rod position in NRX.

Several modifications have since been made to the system. At the present time the total capsule run is about 900 feet. The travelling time is 12 seconds equivalent to an average speed of 50 mph.

Samples can be installed from the top of the reactor or from the Chemistry building into either a central core position or into a rod in the J-rod annulus. Irradiated samples can be received at the top of the reactor or in one of four receiving stations in different laboratories in the Chemistry building. In this way short-lived radioactive samples can be measured in the laboratory within seconds of their irradiation in the reactor.

The line between the two buildings is of 3/4 inch outer diameter polythene tubing. Compressed air is used to drive the capsule and also to cushion the capsule as it enters

the rod and stops at the maximum flux elevation.

Air at the rate of 2 ft³/min cools the sample while it is being irradiated.

The rod upper shielding plug is a sand-filled aluminum cylinder 2-1/4 inches in outside diameter and about 12 feet long. The sample tube and the compressed air tubes make one convolution through the shield plug to prevent radiation streaming into the upper header room. The outer sheath of the in-calandria section is made of aluminum and is 2 inches in outer diameter and has a 1/16 inch thick wall.

The samples are normally inside a standard iron pneumatic-carrier capsule 1-1/2 inches long and 9/16 inches in diameter. They may also be put inside a small polythene capsule which fits inside the iron capsule.

The size and heat-output limitations on the samples means that most weigh less than one gram.

Unperturbed fluxes as high as 8×10^{13} n/cm²/sec have been measured in the pneumatic carrier in the central core position.

D. Fast Neutron Rods

The fast-neutron rods, formerly called transformer rods, are used to irradiate samples in a higher fast-neutron flux than is normally available in a thermal neutron reactor. Most of the samples are metal tensile or impact test specimens of materials to be used for pressure tubes and fuel sheaths in power reactors.

A description of the development of fast-neutron rods can be found in IOI-235, "Development of Transformer Rods In The NRX Reactor" by T.R. Kirkham. A full description of the rods is given in the NRX Reactor Handbook, IOI-225.

The samples are irradiated inside a hollow cylindrical uranium section of the rod. Thermal neutrons enter the uranium causing fissions which increase the number of fast neutrons striking the specimen inside the cylinder.

In general, the rods connect directly into the normal fuel rod water coolant headers although some of the designs have their own separate coolant headers.

Fast-neutron rods are of two types, one for irradiations from 120°F to 250°F called the low-temperature (Mark I and Mark II Rods), the second type for irradiations at 500°F to 800°F called the high-temperature rods (Mark IV). The temperature of the samples in the Mk I and Mk II rods is controlled (normally at 120°F) by passing cooling air over

the samples. The high temperature rods have no cooling air but depend on the placement of the rod in a reactor position at the proper neutron flux to give the desired temperature conditions in the metallurgical sample.

The sample section in the Mk I rod is a hollow uranium cylinder 8 inches long, 1.36 inches in outer diameter and one inch in inner diameter. Both surfaces of the cylinder are water cooled. In order to decrease the reactivity load of the rod a length of standard uranium metal fuel rod 44-3/4 inches long is attached to the bottom of the sample section. An aluminum flow tube 1-1/4 inches in outer diameter attached from the sample section to the top of the rod allows cooling air at a rate of about 3 ft³/min to flow down the rod past the samples and out the top of the rod to the Buffalo exhaust fan as described in Section III-E-4. The samples are installed through the flow tube and are hung on the end of a sand-filled shielding section.

The MK II low-temperature rod is similar to the Mk I rod except that the sample section is 29 inches long allowing several specimen irradiations at one time. The lifetime of individual transformer rods has been short with most failures occurring in the lower uranium slug.

The high temperature rods have no sample air cooling and the inner part of the uranium tube has no water cooling. There is no uranium slug on the end of the rod.

The Mark III Fast Neutron Rod has been designed in order to increase the lifetime of rods, the size of the samples and to make possible the installation of temperature control furnaces. It can be operated as a high temperature or low temperature rod as required.

The sample section is 8 feet long, 1-3/4 inches in outside diameter and 1-3/8 inches in inner diameter. The fuel section is made of co-extruded annular uranium alloy containing two percent by weight of zirconium with 0.015 inch thick zircaloy-2 sheaths inside and outside. The samples are provided with air cooling.

It is hoped to put assemblies such as creep-rate-measurement machines into this type of fast-neutron rod. Space limitations made design work very difficult.

At the time of writing (July 1961) four low-temperature rods and two high-temperature rods are in the reactor.

E. The Hydraulic Rabbit Facility

The NRX Hydraulic Rabbit Facility was installed in NRX in 1958 to carry out a large number of short-term irradiations of fuel similar to the loop fuel specimens.

It is impractical to do brief irradiations in the reactor loop facilities because both the loop and the reactor must be shut down for fuel specimen changes. Hydraulic rabbit samples can be installed and removed while the reactor is at full power.

Over two hundred irradiations ranging from 30 seconds to 40 minutes long have been done in the hydraulic rabbit since it was installed.

A complete description of the hydraulic rabbit can be found in the NRX Design Manual, IOI-4, Section A-17-6, while, IOI-236, "The Development of the NRX Hydraulic Rabbit Facility" by F.A. McIntosh tells of the operation and modifications to the facility since the original installation.

A schematic flow diagram is shown in Fig. 20(B) while Fig. 20(A) shows a typical fuel element with the guide section connected to it by a ball and socket joint.

The samples are loaded in the reactor hall basement and are propelled by the cooling water through a 7/8-inch-diameter inlet tube of stainless steel. The sample is held by a retractable stop at the maximum flux position. An outer aluminum sheath two inches in diameter encloses the calandria section of the tube, which is of aluminum also. The sample leaves through a 7/8-inch diameter stainless steel tube.

A 33 foot length of Saran tubing (snake) is inserted from the basement loading station after the sample has entered the reactor to ensure that an irradiated sample cannot return to the bottom of the reactor (as happened in 1958, see IOI-236).

When the sample has been irradiated for the required time the flow stop is opened by the air motor and the sample is pushed by the cooling water up to the top thermal shields where it is held by another Saran tube inserted through the discharge tube of the receiving flask.

Samples are allowed to decay for 24 hours in the vicinity of the upper thermal shields. The snake is then removed through the receiving flask in the north recombination room. The sample is installed in one hole of a rotating eight-hole magazine in the flask. Five inches of lead shielding are provided.

Up to four specimens can be installed in one hole of the magazine so quite frequently four elements are irradiated in succession.

Water taken from the high-pressure system at the rate of 6 Igpm provides both the propelling force and the coolant for the specimens.

A gamma-ray monitor on the outlet-water line gives warning of a rupture of a specimen in which case the sample would be immediately removed. The outlet water bypasses the first delay tank so that the coolant activity monitors will not trip the reactor if a specimen ruptures.

Most fuel specimens are UO_2 pellets sheathed in stainless steel or zircaloy. As shown in Fig. 20(A) the end cap on the fuel element is provided with milled fins so that the turbulent water flow will more or less center the element in the central channel during irradiation.

F. Miscellaneous Experimental Rods

Many different types of experimental rods have been installed in NRX at various times. Most of these rods have used the standard fuel-rod coolant connections and in many cases the end sections of a standard fuel rod. Some of these special rods are discussed below.

Eight UO_2 pellet rods have been installed in NRX up to the present time. These tests have been to measure the thermal conductivity of UO_2 fuel and the gas adsorption of the UO_2 fuel while under irradiation.

Flat fuel rod elements for NRU have been irradiated in NRX in order to test the performance of the rods when purposely defected. Rods consisting of seven pencil elements of the new nineteen-element fuel rod for NRU have also been irradiated.

Standard fuel elements from the Canada India Reactor (CIR) and the German FR-2 reactor have been tested by irradiating them in an NRX fuel rod position.

Irradiation of natural-uranium specimens from the UKAEA Calder Hall reactor fuel is to begin soon.

Aluminum nitride slugs fitted inside a standard NRX natural uranium metal fuel rod inner sheath have been irradiated in NRX. This irradiation was done for the Commercial Products Division of AECL for carbon-14 production.

Thorium metal and thorium oxycarbonate rods have been irradiated both in the NRX core and also in the J-rod annulus. The rod in the core as well as being used for U-233 production was used for convenience in reactor loading.

The High Temperature Capsule Irradiation Facility is a rod in the core that is being used to study the decomposition of organic fluids under gamma-ray and neutron bombardment. The rod consists essentially of an aluminum tube leading into the reactor core into which calorimeters can be lowered. Similar experiments are being done in the J-rod annulus.

A biological test facility has been installed on the North Thermal Column as shown in Fig. 1. Biological specimens such as rats, tree seeds or insects can be pushed into a cavity in a block of bismuth situated inside the main door of the column. The bismuth screens out the gamma-rays so that the effects of neutron irradiation alone can be evaluated.

A cobalt wafer rod was irradiated in which 1900 grams of standard cobalt pellets 1 mm in diameter and 1 mm long weighing 7 milli-grams each were pressed inside a cylindrical aluminum sandwich similar to the cobalt rods in the NRU reactor. It was felt that this type of rod might be used to replace the cobalt tray rods.

VII. THE FUTURE OF NRX

Since initial startup in 1947, NRX has changed from an experimental and plutonium-production reactor to an engineering-test and isotope-production reactor. As can be seen from Section VI many different types of experimental assemblies have been tested in the reactor.

The maximum reactor power has been increased from the design limit of 20 MW to its present operating power of 42 MW.

As noted in Section IV-C a start has been made in converting NRX from natural-uranium-metal fuel to natural-UO₂ fuel. This conversion, to be completed in 1962, will increase the maximum flux by 50% without any increase in power. The usefulness of the reactor for loop fuel tests and long-term fast-flux metallurgical sample irradiations will thus be increased.

It is known that the calandria tubes are corroding from the damp air between the fuel rods and the tubes. Plans are being made for the replacement of the calandria when such is necessary.

A spare calandria vessel is being modified to have large calandria tubes so that large experimental assemblies can be installed. Three 3-1/2-inch-inside-diameter tubes equipped with expansion bellows are being installed for the use of large loop experiments midway in size between the central thimble and the present small loops. Three tubes are being fitted with bellows to be used with loop pressure tubes of the present small size. The bellows are designed to protect the calandria tubes from being overheated by the high temperature loop experiments. The tubes will extend up to the upper header room where the bellows will be situated. Four 3.5-inch tubes without bellows will be installed in the central core to be used for large fast-neutron-irradiation facilities. It will be possible to replace these tubes without removing the calandria. The upper, inner steel thermal shields will be replaced with aluminum shields.

It is expected that the new calandria and thermal shields will be ready for installation by the end of 1961. No date has yet been set for the changeover.

In addition to these programs which are going ahead, studies have been made and are continuing on the following proposals:

1. Increasing the power of the reactor up to 100 MW.
2. Changing fuel rods under power.
3. Utilize the thermal output of the reactor to produce low grade heat to be used for space heating.

It appears that NRX will continue to be a very useful reactor for many years to come.

FIG. 1
The NRX Reactor

LEGEND

1. Floor slabs over recombination rooms.
2. Revolving floor.
3. Vertical fuel rod removal flask.
4. Vertical rod storage block.
5. Horizontal storage holes for plugs from the experimental holes.
6. Horizontal flask used to remove plugs from the experimental holes.
7. Research equipment set up at the experimental holes.
8. Biological facility in front of the north thermal column.
9. Floor slabs covering the heavy water storage tanks.
10. Self-Serve units.

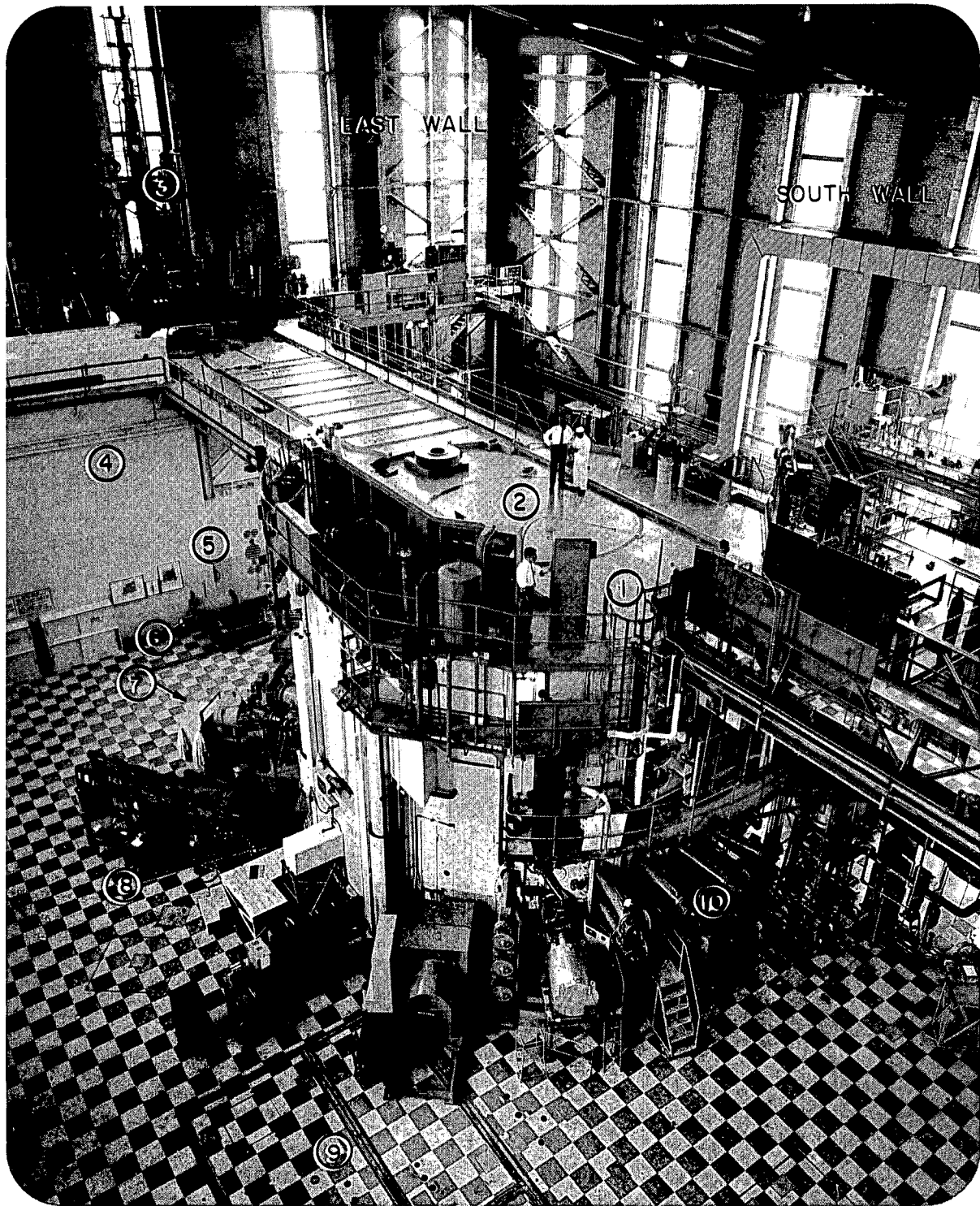
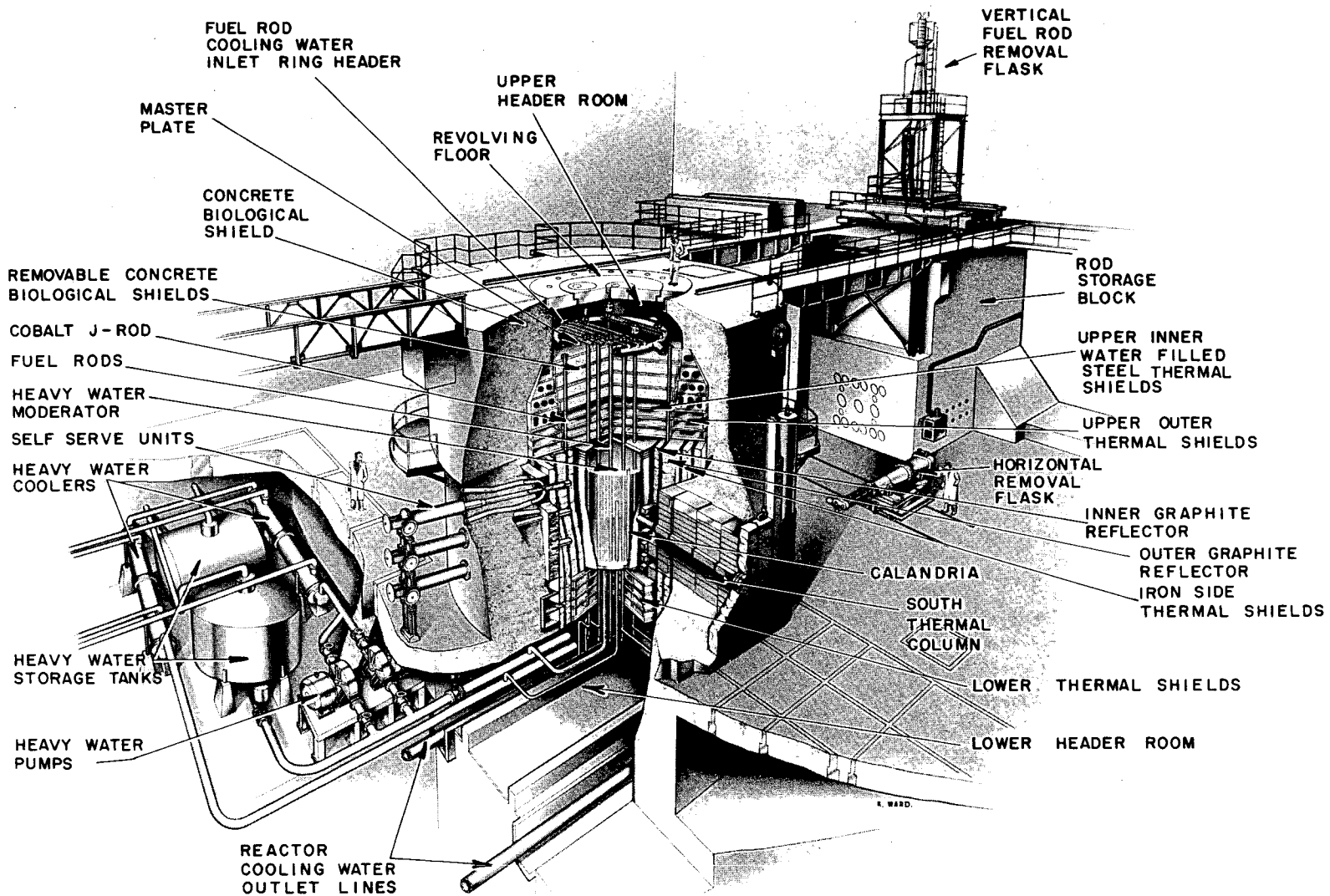


FIG. 1 THE NRX REACTOR
VIEW FROM NORTH WEST CORNER OF REACTOR HALL



NRX REACTOR

ATOMIC ENERGY OF CANADA LIMITED - CHALK RIVER, ONTARIO

FIG. 2

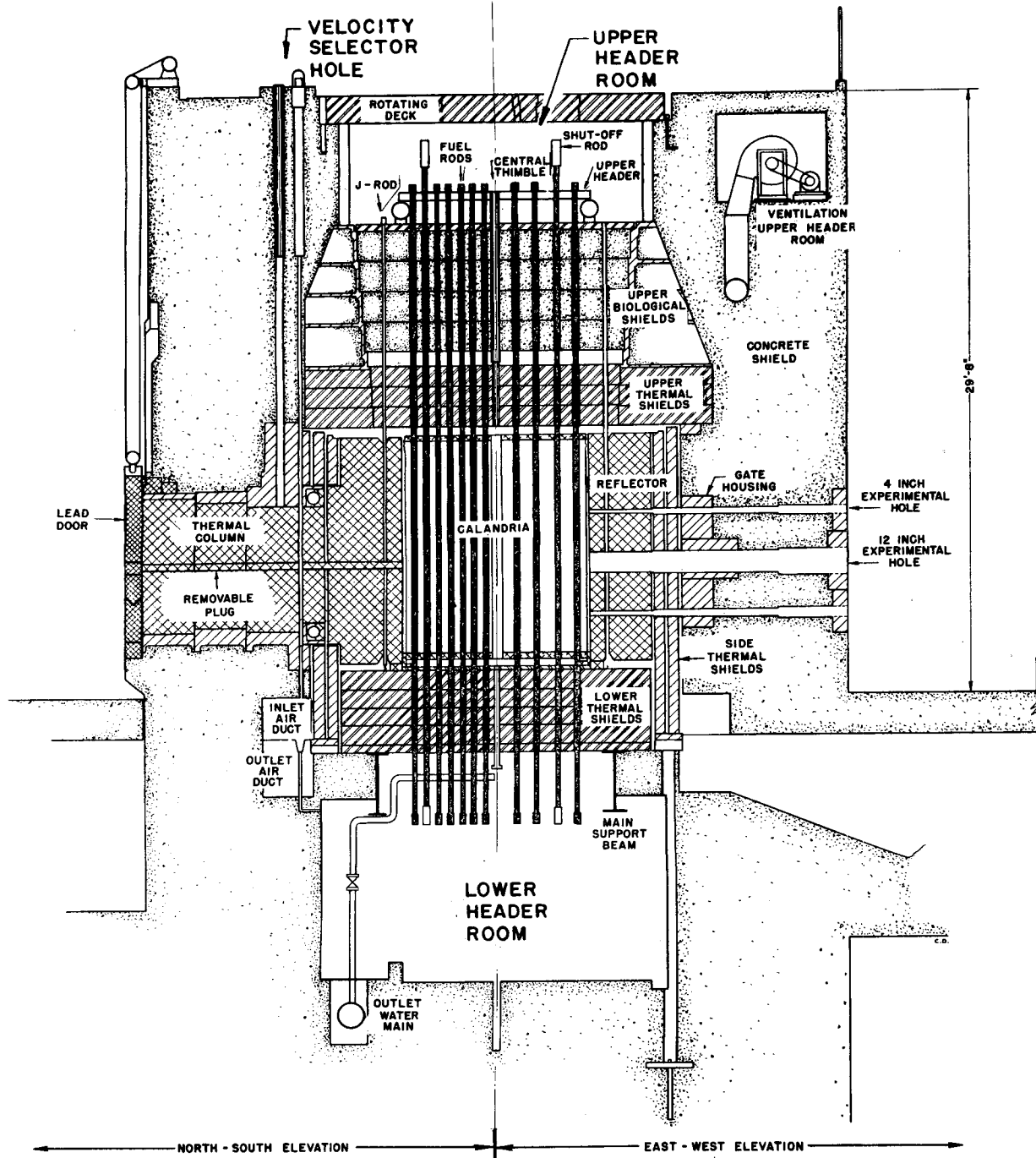


FIG. 3 ELEVATION CROSS-SECTION OF THE NRX REACTOR

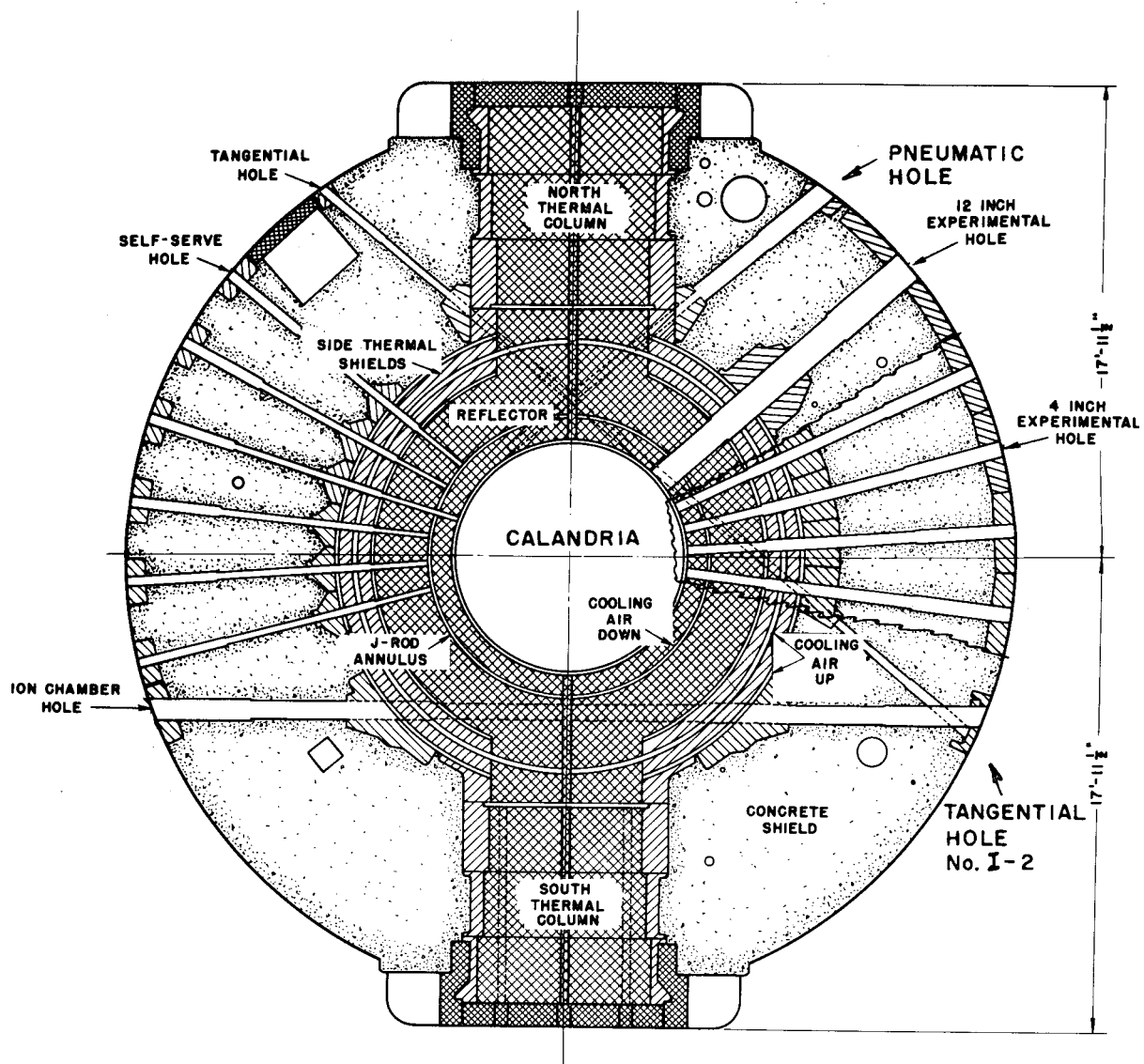
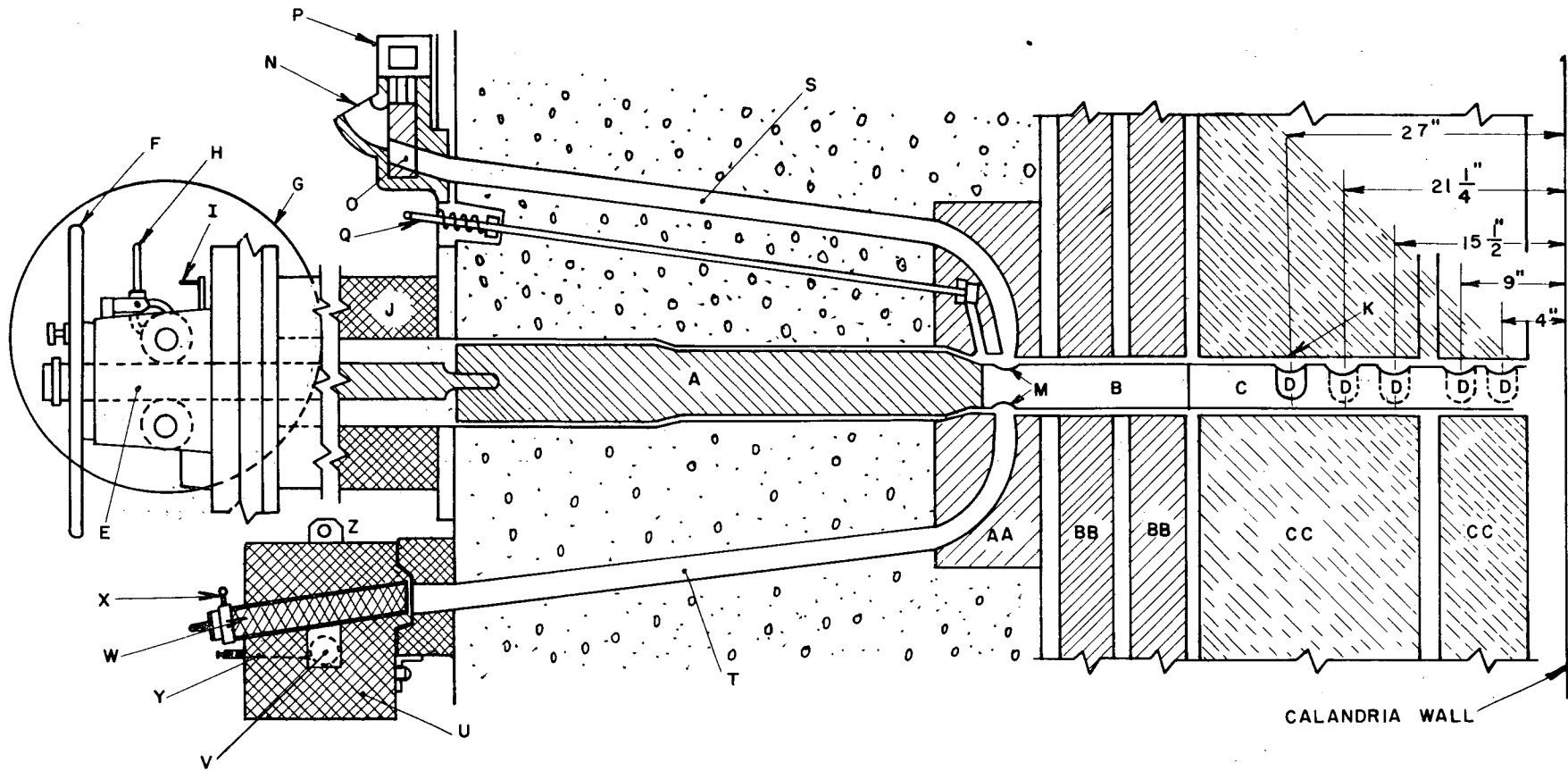


FIG. 4

PLAN CROSS SECTION THROUGH THE REACTOR STRUCTURE



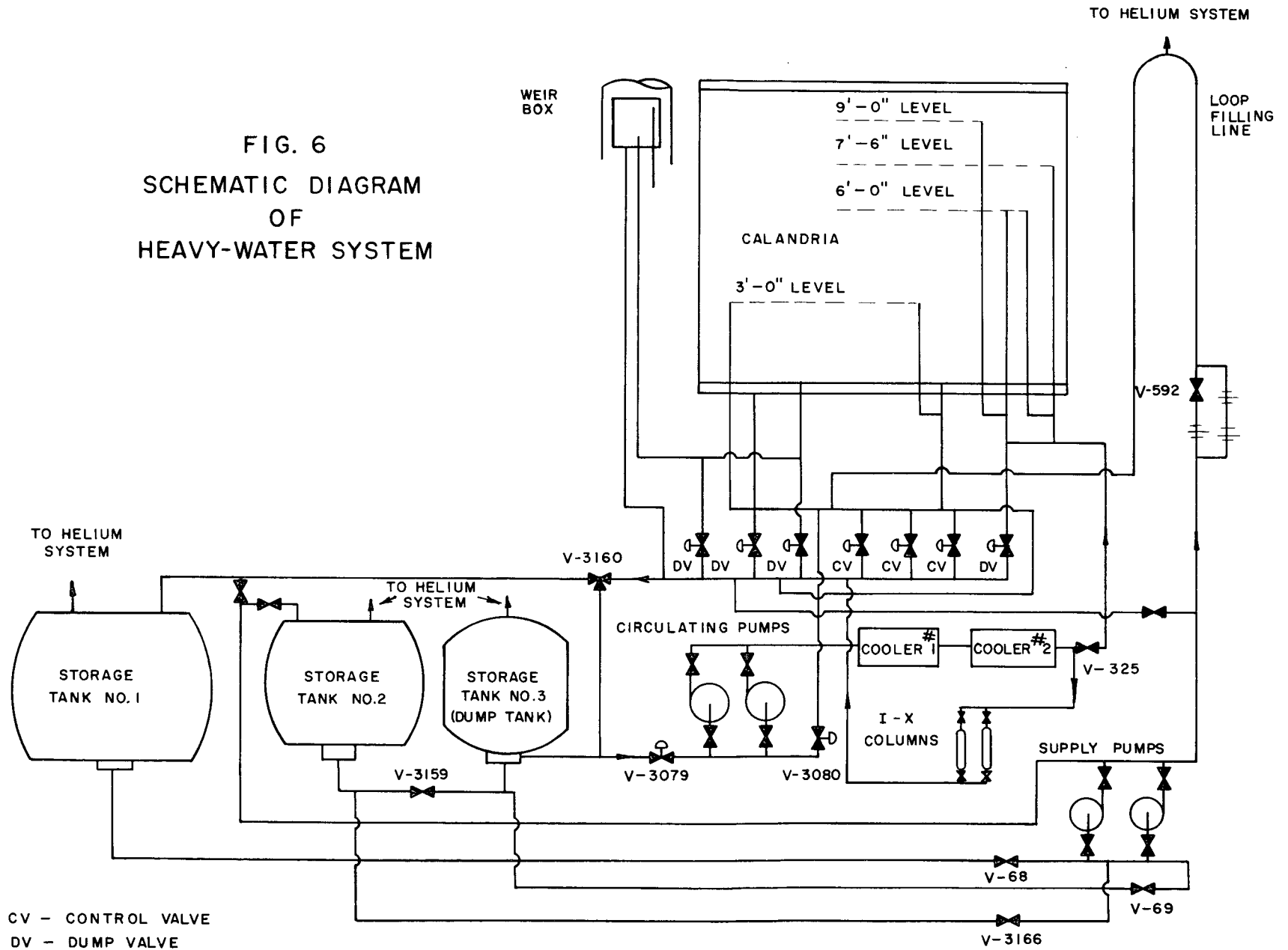
A - SAMPLE PLUG, SOLID STEEL SECTION
 B - SAMPLE PLUG, STEEL TUBE SECTION
 C - SAMPLE PLUG ALUMINUM TUBE SECTION
 D - RECEPTACLE IN SAMPLE PLUG
 E - RACK
 F - HANDWHEEL FOR ROTATING RACK
 G - HANDWHEEL FOR SLIDING PLUG
 H - GEAR RELEASE LEVER
 I - LOCATING LEVER (FOR PLUG LOCKING PIN)
 J - SAMPLE PLUG FLASK

K - ALUMINUM TUBE THROUGH REFLECTOR
 L - AIR OUTLET HOLE TO ANNULUS
 M - AIR INLET HOLE TO SAMPLE PLUG
 N - INLET FUNNEL FOR SAMPLE BALL
 O - BALL RELEASE GEAR
 P - GATE SOLENOID
 Q - TRIGGER ROD
 S - CHARGING TUBE
 T - DISCHARGE TUBE

U - SAMPLE BALL FLASK
 V - POCKET FOR BALL
 W - PLUG FOR SAMPLE FLASK
 X - PLUG LOCKING PIN
 Y - FEELER WIRE
 Z - SUSPENSION LUG
 AA - INNER CASTING
 BB - SIDE THERMAL SHIELDS
 CC - GRAPHITE REFLECTOR

FIG. 5
 CROSS SECTION OF SELF SERVE UNIT

FIG. 6
SCHEMATIC DIAGRAM
OF
HEAVY-WATER SYSTEM



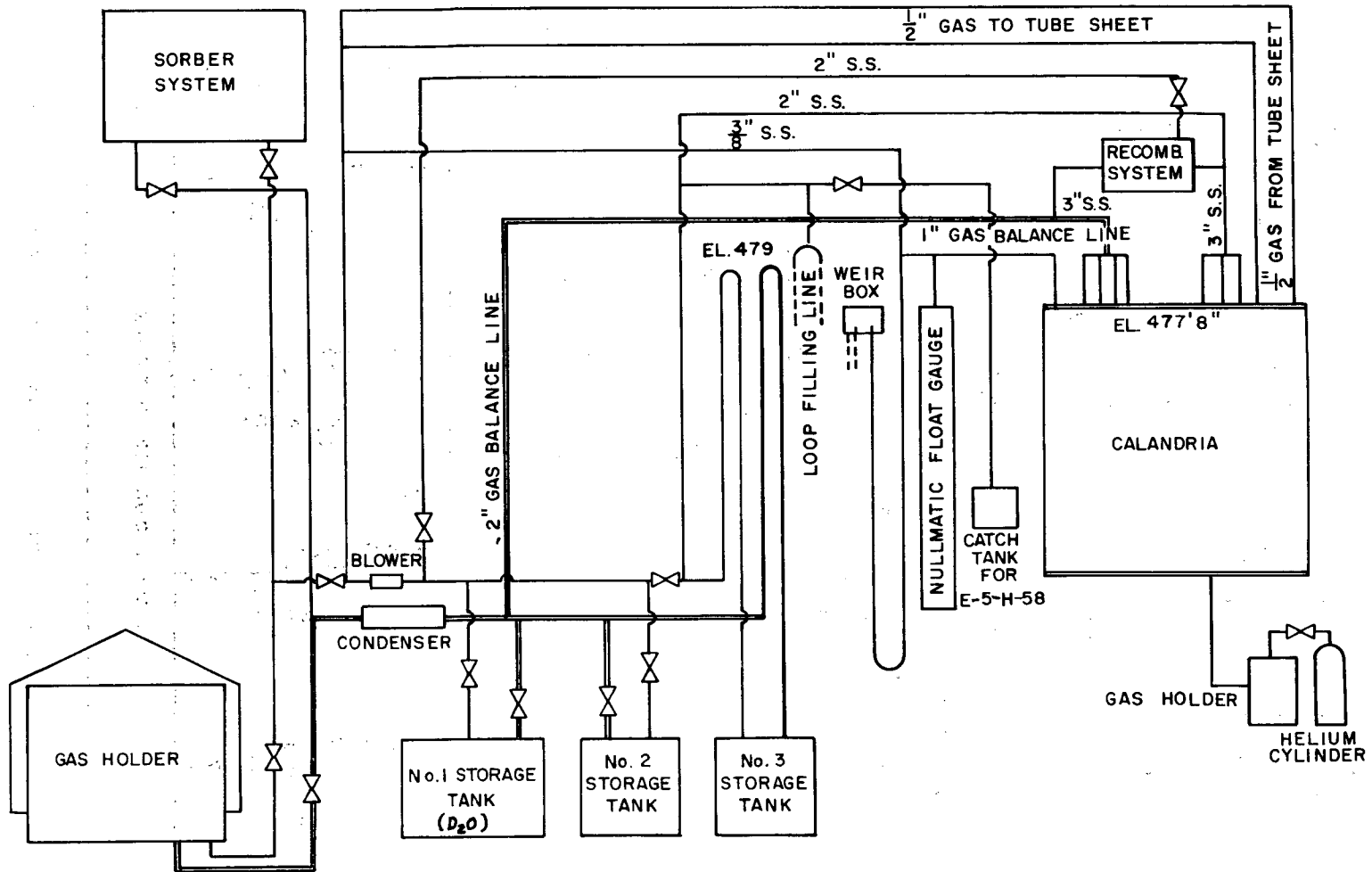


FIG. 7
SCHEMATIC DIAGRAM OF THE HELIUM SYSTEM

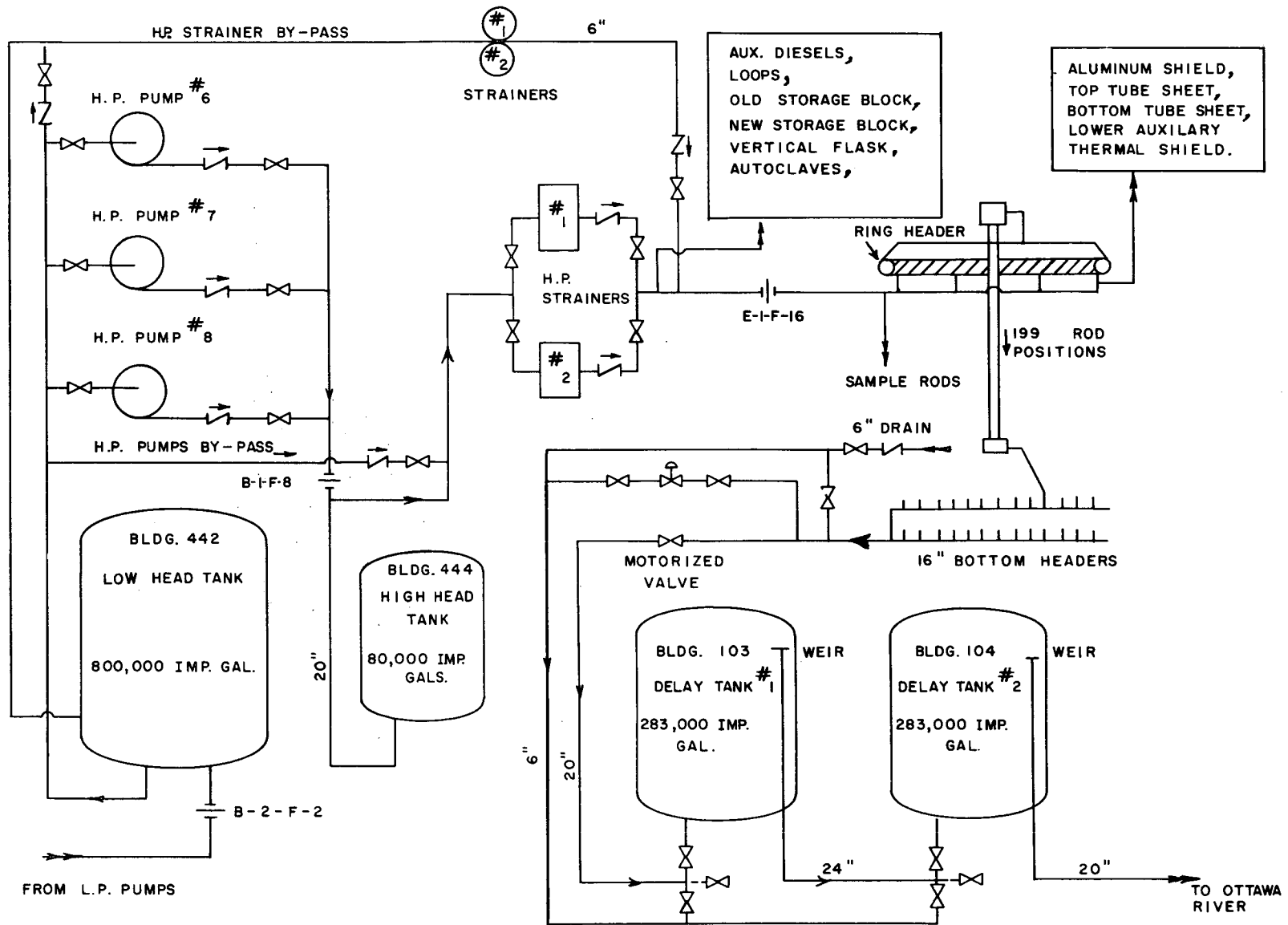


FIG. 9
HIGH PRESSURE LIGHT WATER SYSTEM

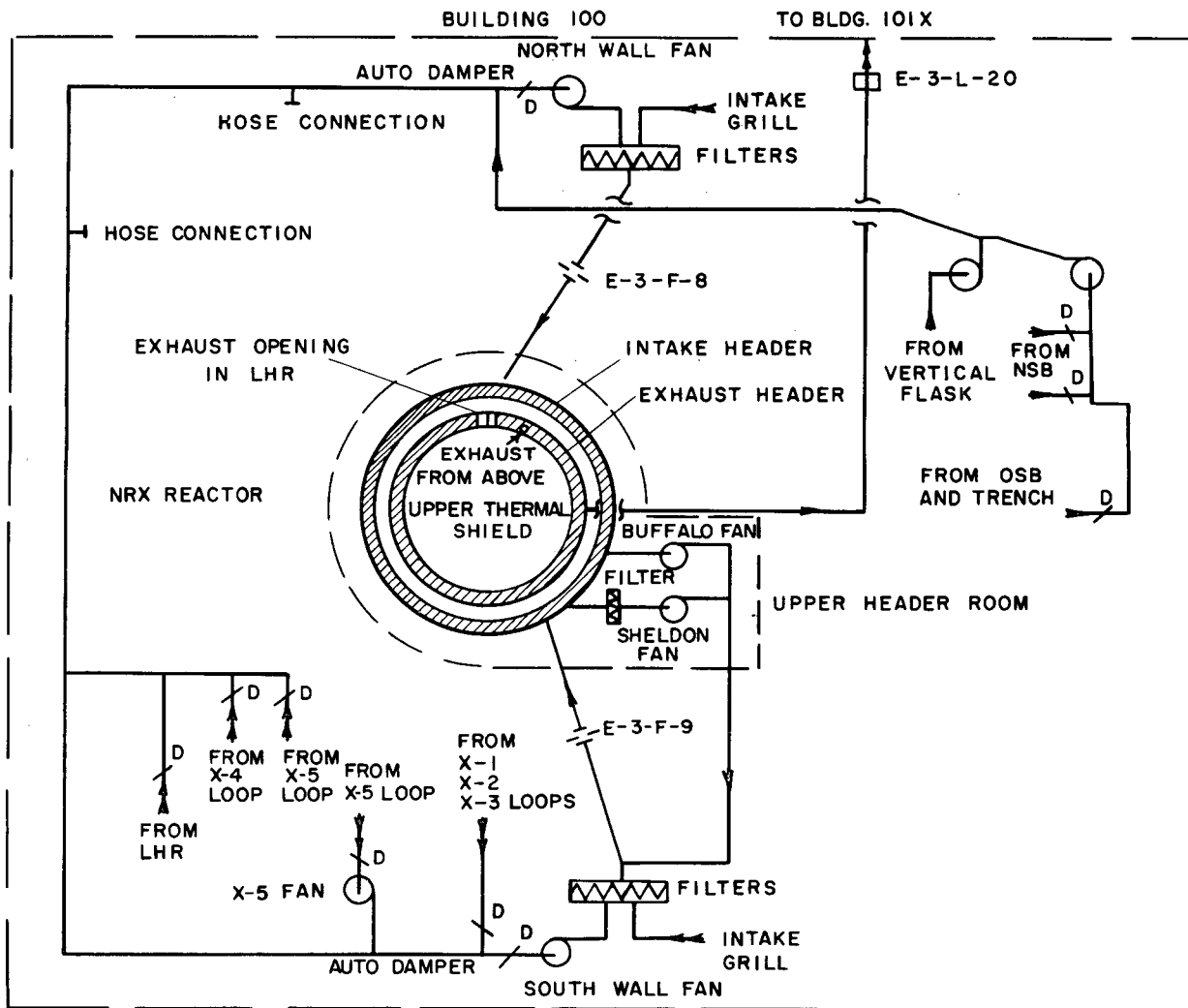


FIG. 10
SCHEMATIC DIAGRAM, EXHAUST AIR SYSTEM

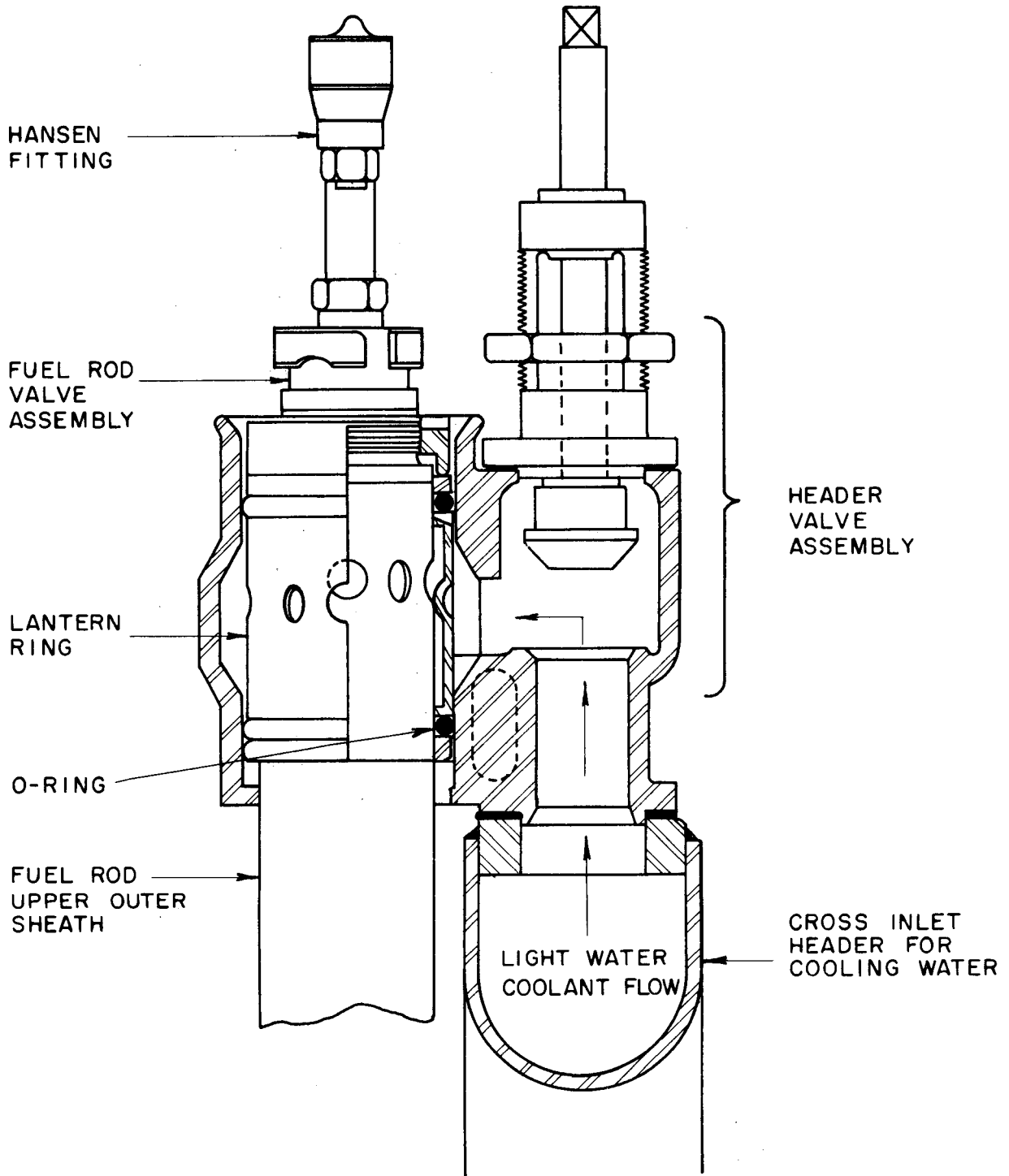


FIG. II

NRX FUEL ROD UPPER VALVE ASSEMBLY

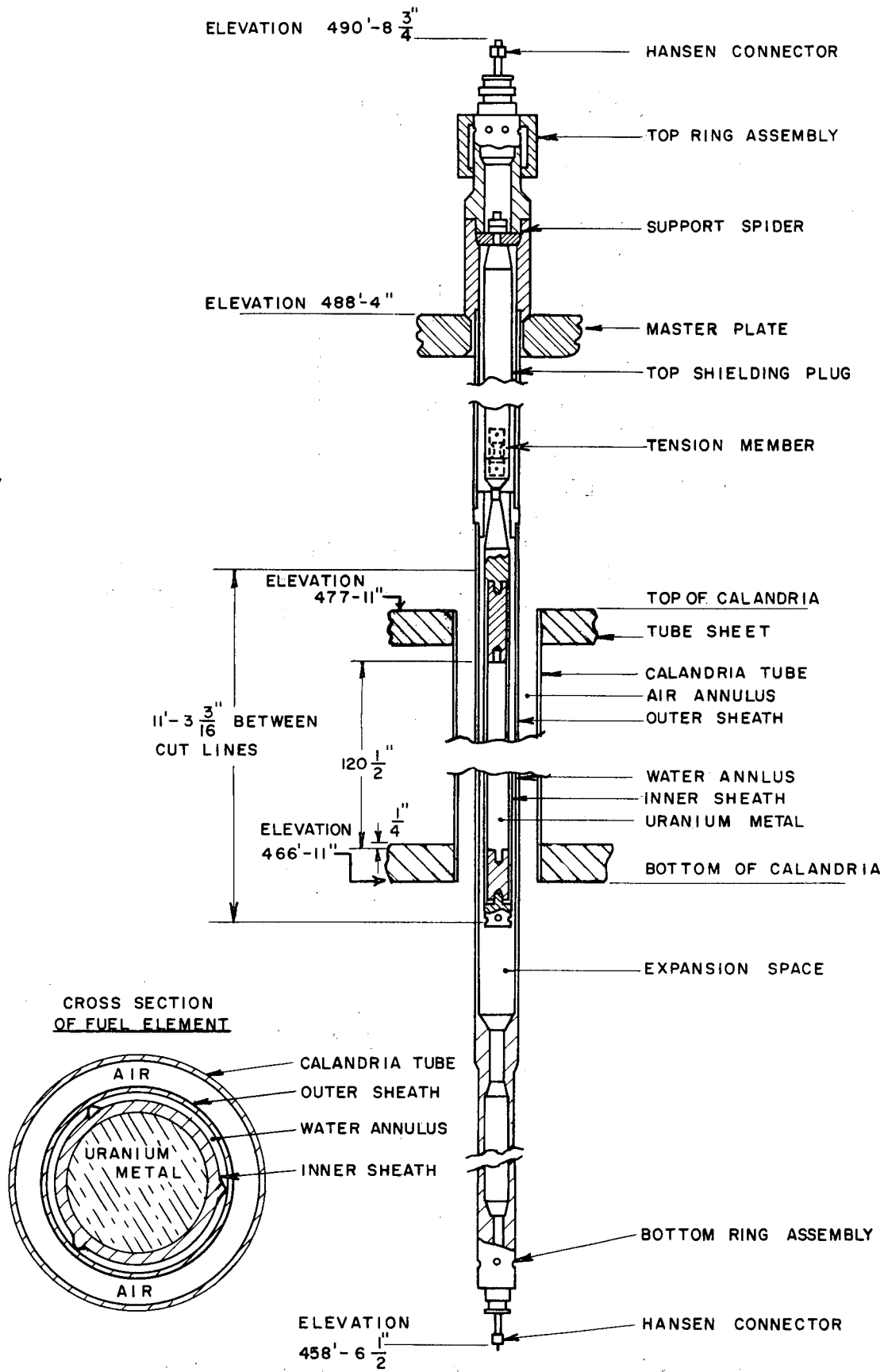


FIG.12
TYPE FIVE URANIUM METAL ROD

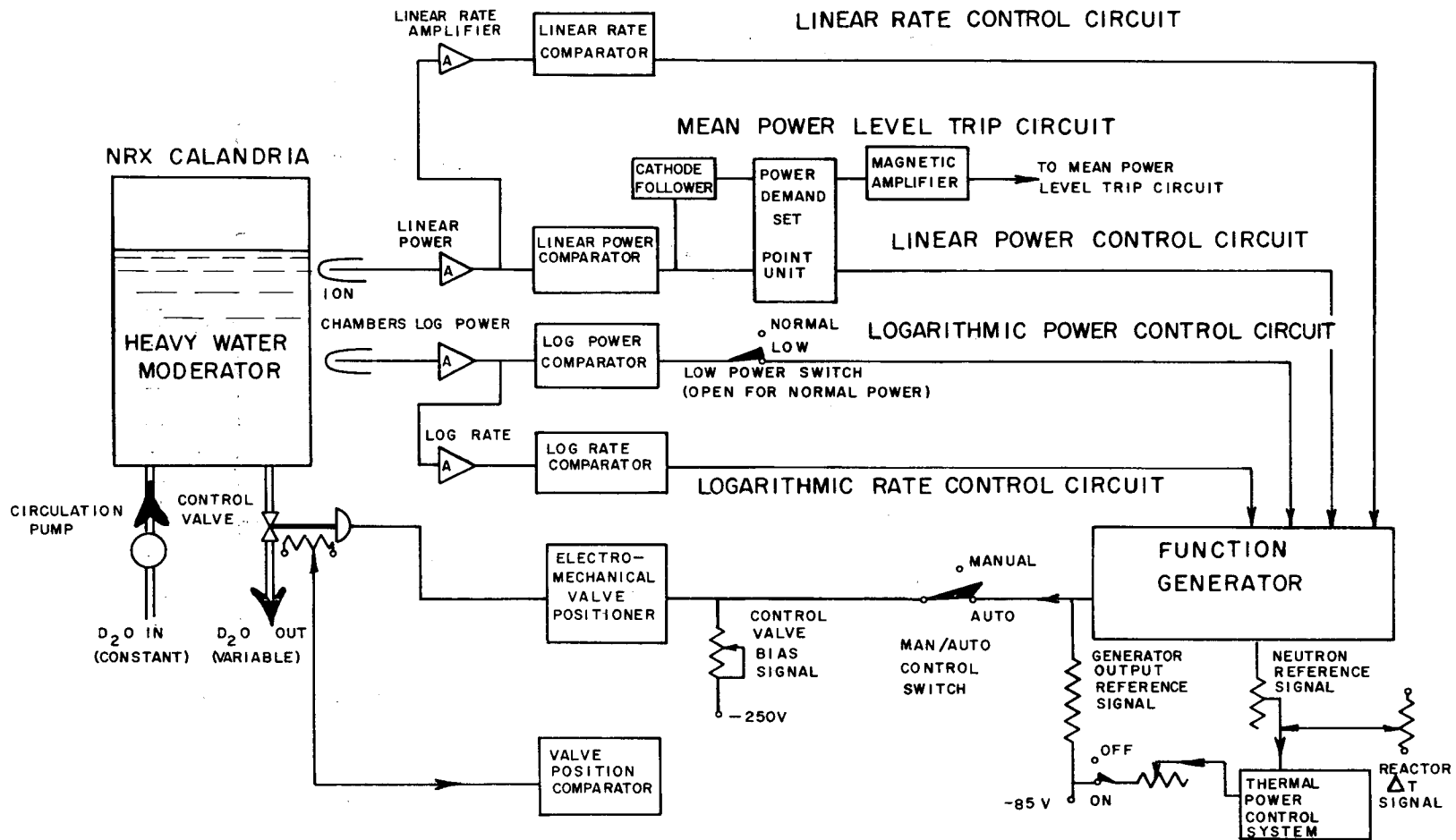


FIG. 13

NRX AUTOMATIC CONTROL SYSTEM - SINGLE CHANNEL SCHEMATIC

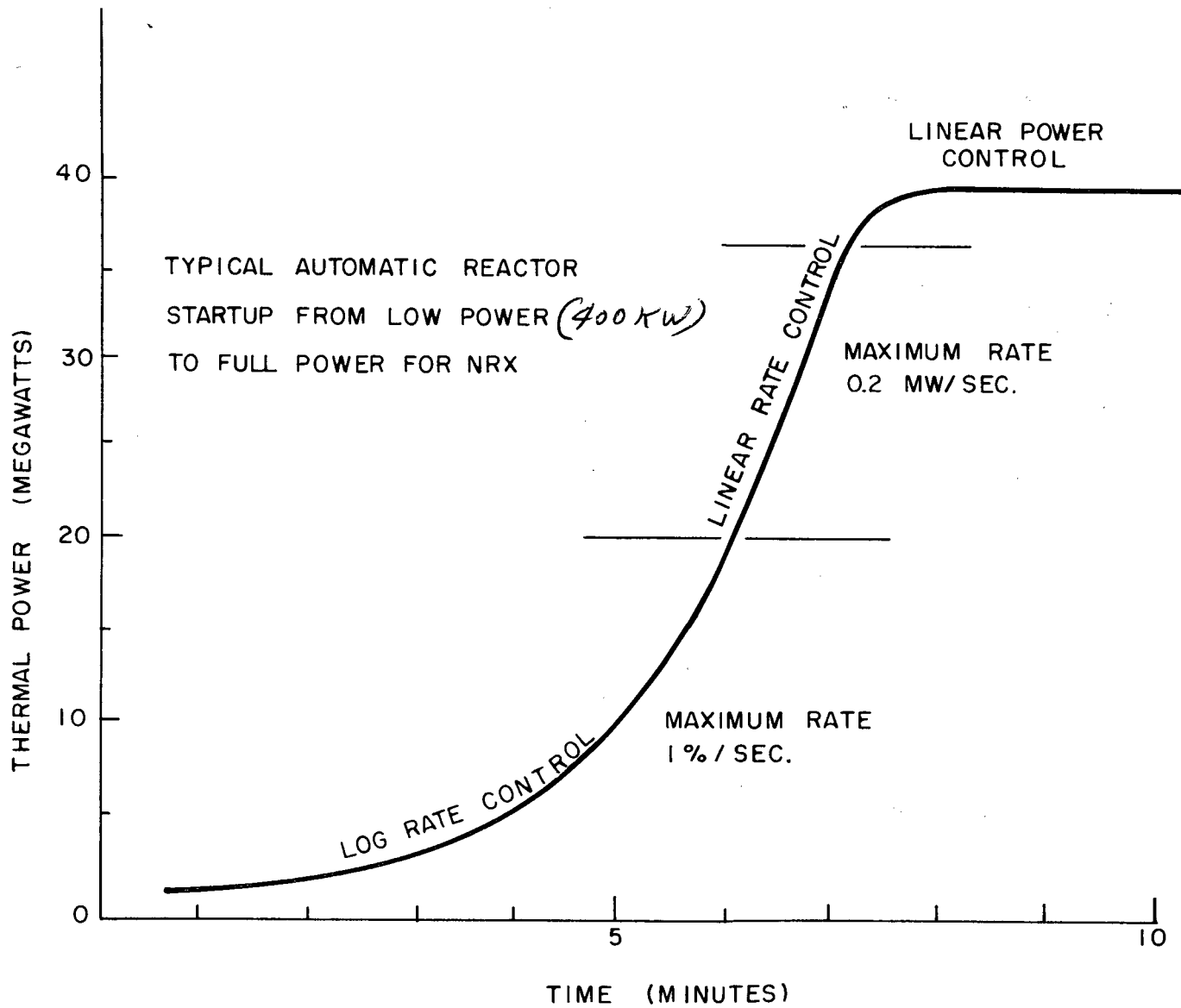


FIG. 14

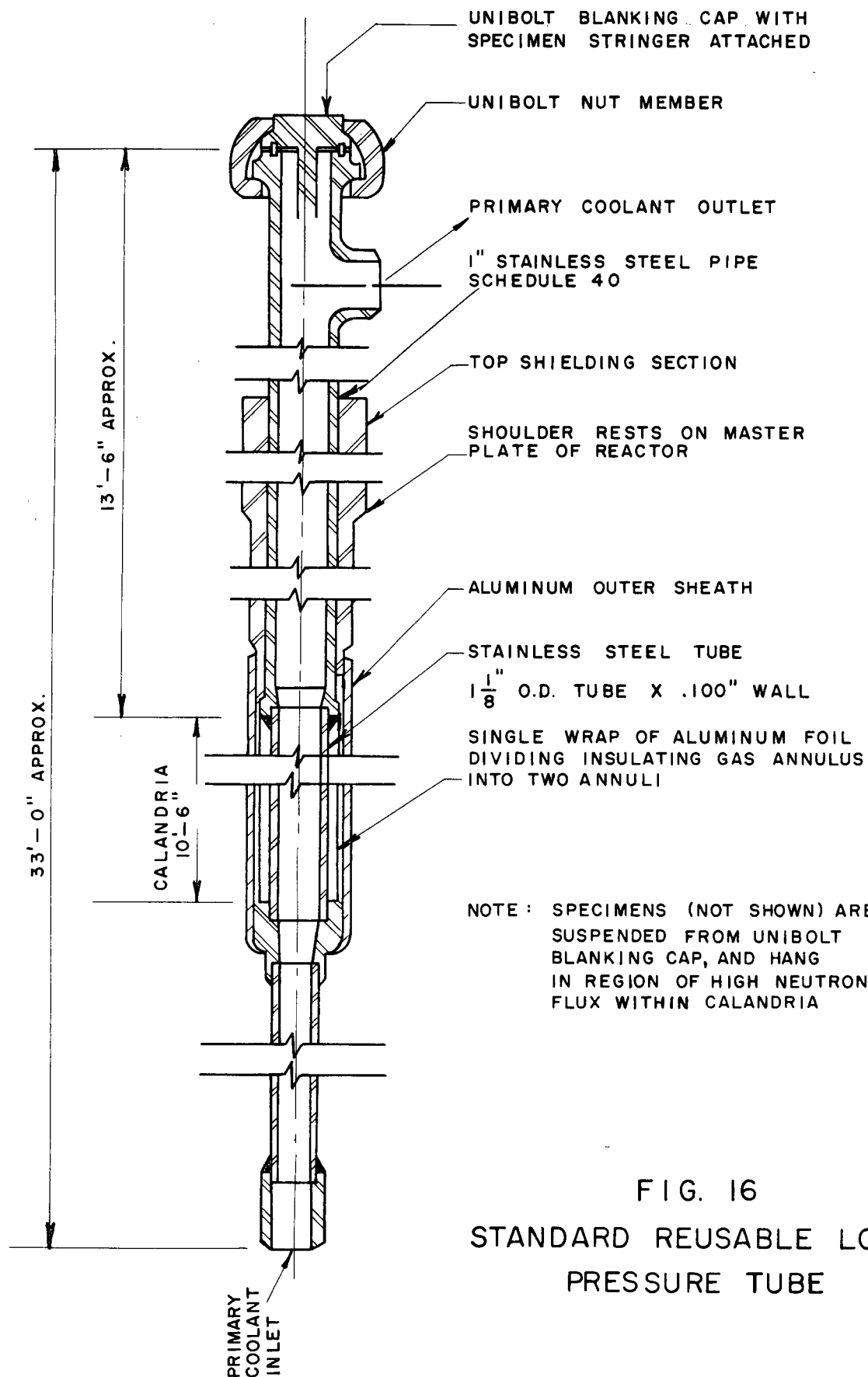


FIG. 16
STANDARD REUSABLE LOOP
PRESSURE TUBE

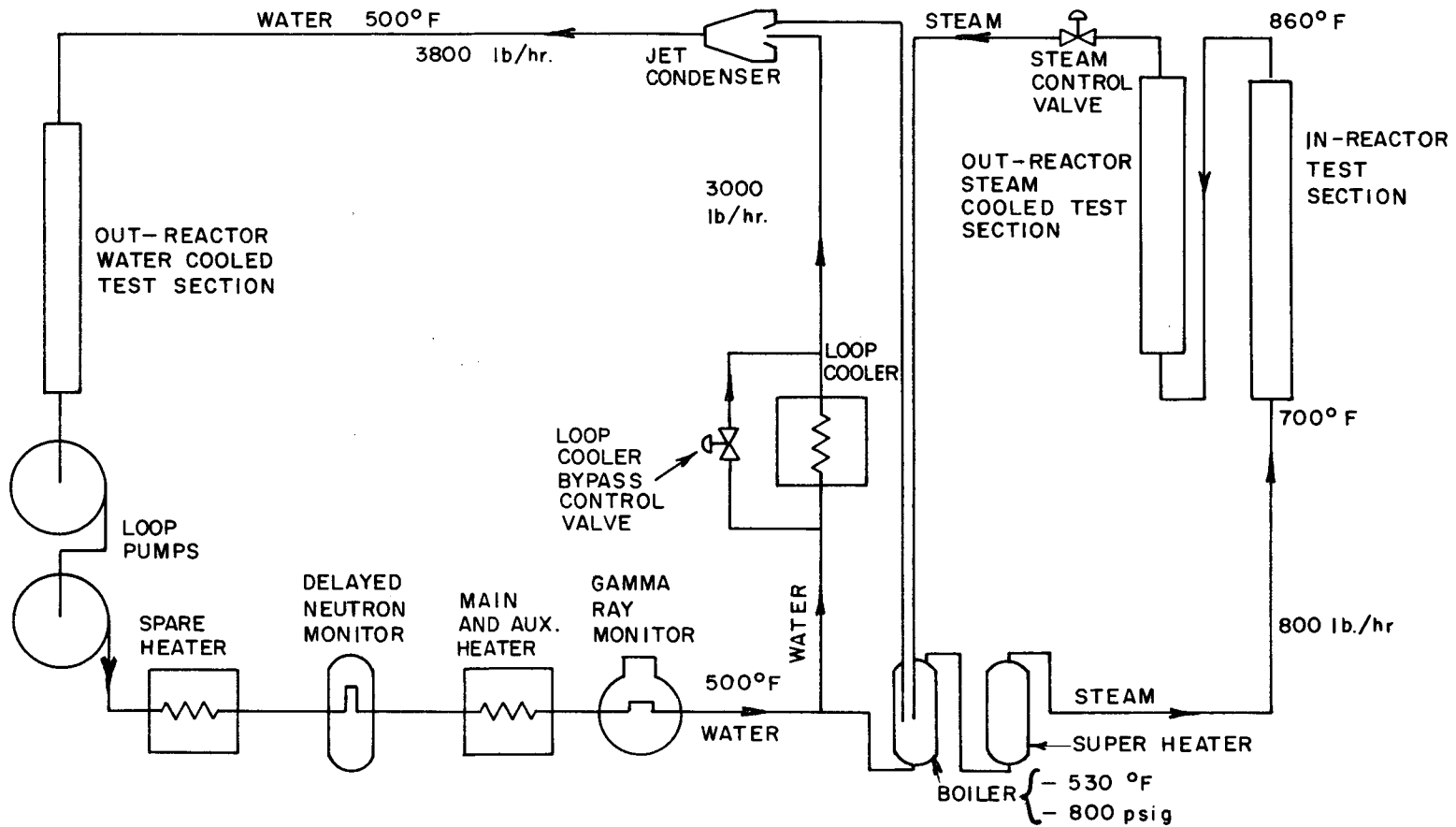


FIG. 17

SIMPLIFIED FLOW DIAGRAM OF X-4 STEAM COOLED LOOP

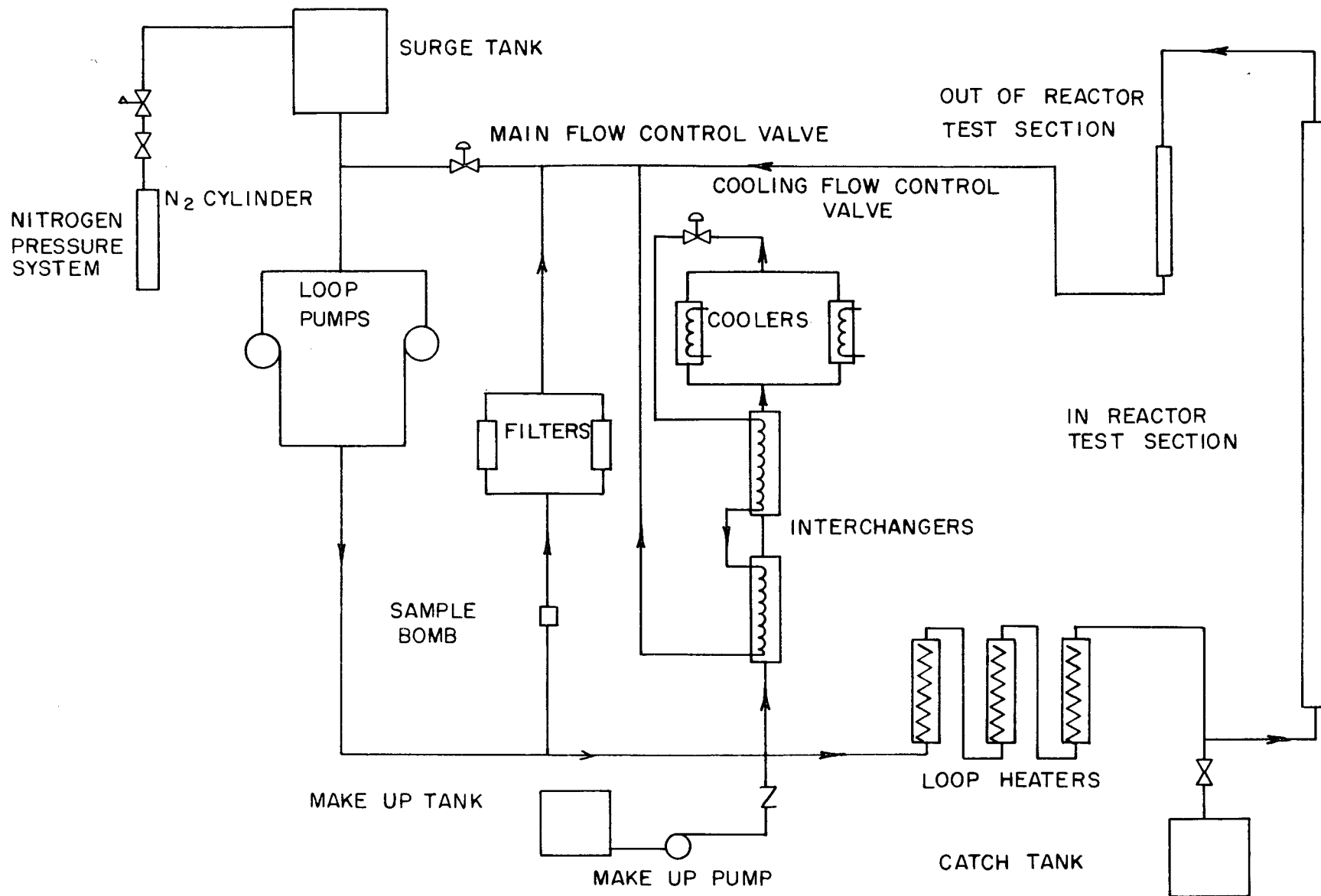


FIG. 18
FLOW DIAGRAM X-7 LOOP

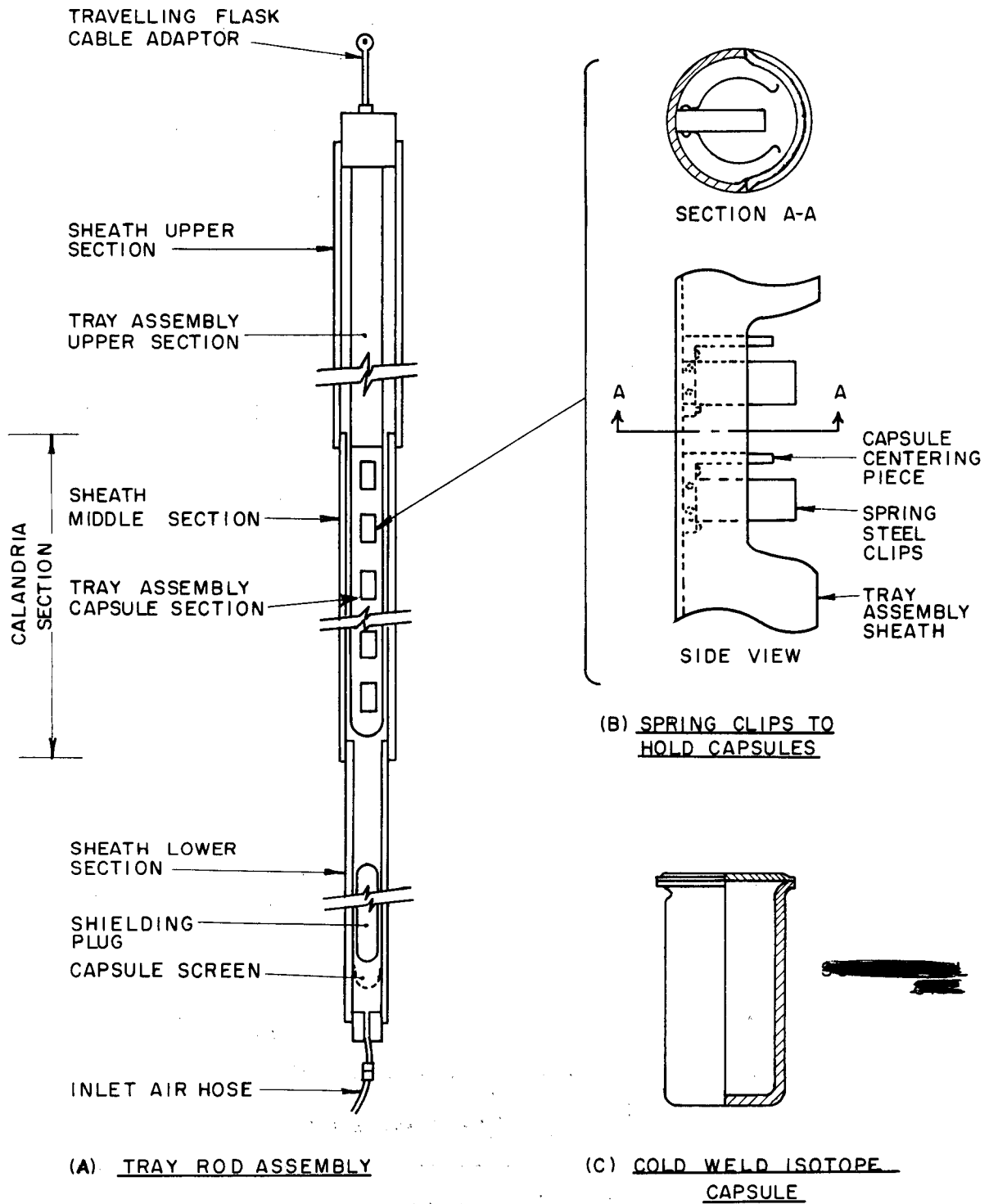
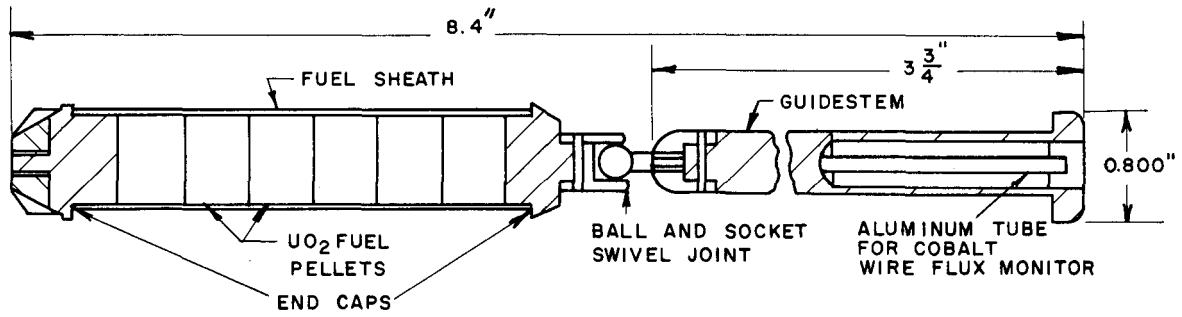
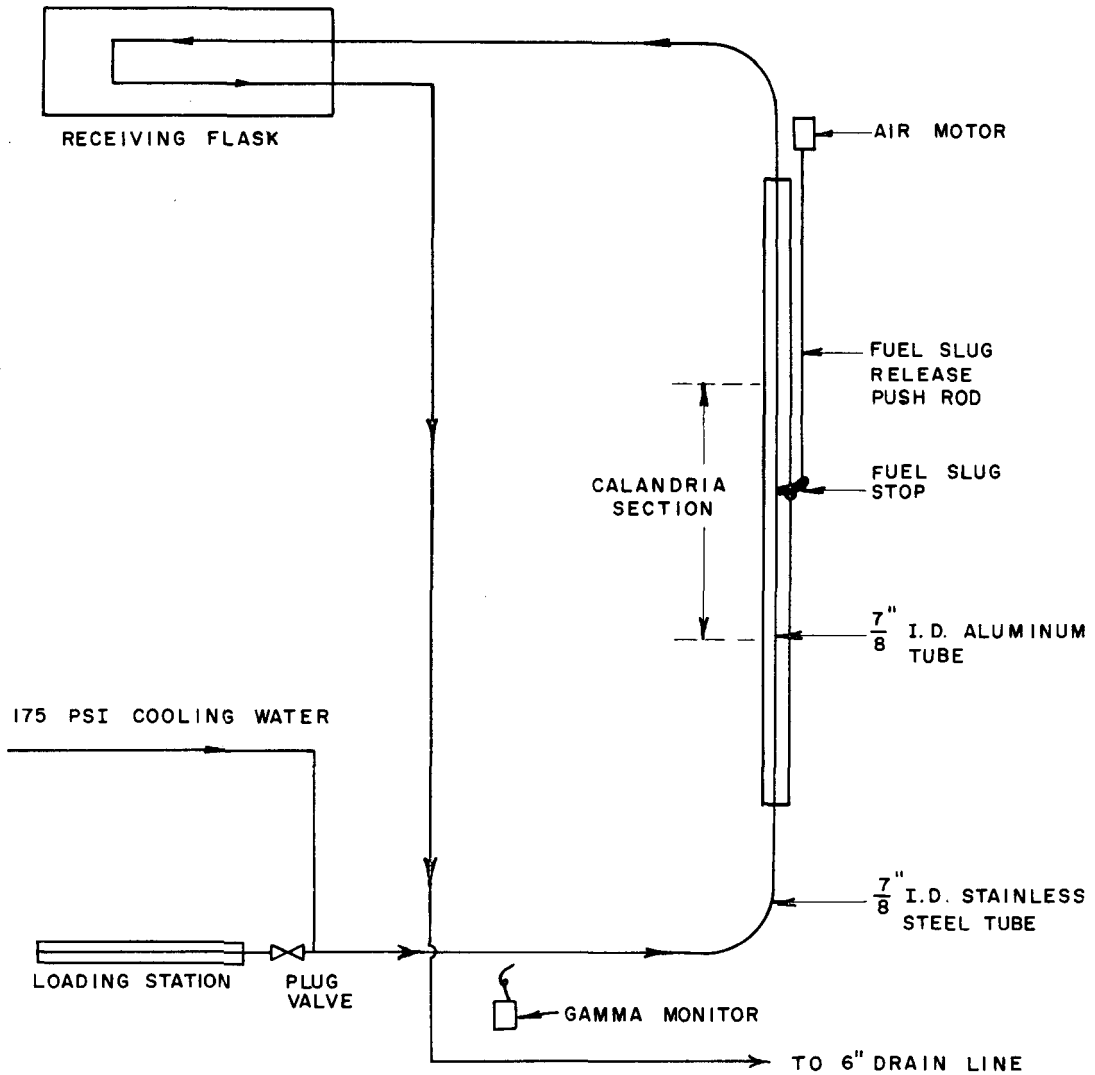


FIG. 19



(A) TYPICAL TEST FUEL ELEMENT FOR HYDRAULIC RABBIT IRRADIATION.



(B) SCHEMATIC DIAGRAM

FIG. 20
HYDRAULIC RABBIT