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AEC RESEARCH AND DEVELOPMENT REPORT

LARGE HEAVY-WATER-MODERATED POWER REACTORS

An Engineering Feasibility Study

by

The Engineering Department, Design Division

Wilmington, Delaware

Coordinated by
H. J. Kamack

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AN ENGINEERING FEASIBILITY STUDY

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Coordinated by

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Approved by

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Technical Division, Reactor Engineering Section

November 1964

E. I. DU PONT DE NEMOURS & COMPANY
SAVANNAH RIVER LABORATORY
AIKEN, SOUTH CAROLINA

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ABSTRACT

The objective of this study was to develop a particular design concept for 3500-Mwt and 8300-Mwt heavy-water-moderated power reactors, cooled by heavy water or an organic liquid, in sufficient detail to form a judgment of their feasibility. During the study it was established that the plants can be designed and built as conceived, that the power costs will be in an economically competitive range, and that the plants can be operated safely and with the intended performance.

CONTENTS

Page

Page

FIGURES | Page

LARGE-HEAVY-WATER-MODERATED POWER REACTORS

INTRODUCTION

By request of the Atomic Energy Commission, a study was made of the engineering feasibility of constructing large nuclear power reactors moderated by heavy water and cooled by heavy water or an organic liquid. Reactor capacities of 3500 and 8300 thermal megawatts were specified. The Du Pont Company has been developing the technology of heavy-water-moderated reactors for electric power generation for several years, and recently has also investigated their applications to furnishing heat for water desalination and to breeding with the thorium-^{233}uranium cycle. The present study extends previous studies of smaller reactor sizes, which indicated that heavy-water-moderated reactors should be most economical in capacities of 3500 thermal megawatts and higher.

RESULTS AND CONCLUSIONS

Description of Reactors

A design concept of a vertical pressure-tube reactor developed in previous Du Pont studies has been further developed and applied to four cases of heavy-water-moderated reactors:

3500-Mwt Heavy-Water-Cooled
3500-Mwt Organic-Cooled
8300-Mwt Heavy-Water-Cooled
8300-Mwt Organic-Cooled

In each case the moderator, cold and unpressurized, is contained in a vertical calandria tank, through whose tubes pass zirconium-alloy tubes containing the fuel and the hot pressurized coolant. Coolant is supplied to and discharged from the reactor through ring headers which are located above and below the calandria, and which are connected to the pressure tubes by individual 3-inch pipes. The calandrias are about 25 feet in diameter for the 3500-Mwt reactors and 38 feet for the 8300-Mwt reactors.

Approximate equipment sizes and arrangements for the primary cooling loops were established. Six loops are used for the 3500-Mwt reactors and ten larger-capacity loops for the 8300-Mwt reactors. The pumps are single-stage centrifugals with face-type mechanical shaft seals; the steam generators are vertical U-tube units with moisture separating equipment in enlarged upper sections. Extensive use is made of carbon-steel materials in contact with the primary coolants.

A very preliminary concept for on-power refueling was developed based on a fuel handling machine of the type developed by Atomic Energy of Canada, Ltd., for the Douglas Point Nuclear Generating Station. Other reactor auxiliary systems were considered only to a minor degree, since they raise no serious feasibility questions. Plant facilities outside of the reactor system were not considered except that the turbine-generator sizes were established.

The reactors and their primary cooling systems are housed in spherical steel containment shells which are 250 feet in diameter for the 3500-Mwt reactors and 350 feet for the 8300-Mwt reactors.

Feasibility

Construction of these plants in the manner described appears feasible, if preceded by an appropriate development program. It is assumed that one or more smaller prototype plants would be built before the plants considered in this study. The reactor structures present some major problems because of the large sizes of the calandrias and shields, and because of the numerous coolant pipes connecting to the pressure tubes; but it is feasible to design and construct these structures. The large-capacity pumps and steam generators are considered feasible by major manufacturers of such equipment, and appear to represent reasonable and expectable advances over units currently being made. Except for fuel handling, the remainder of the reactor plant facilities involve no unusual design or construction problems.

The problems of on-power refueling were not analyzed in any detail in this study. AECL has demonstrated the feasibility of building a machine for this purpose, but reliability and safety of operation have not been fully established. The other major feasibility questions which will require operating experience of reasonable size power plants to resolve fully are heavy-water loss rates and longtime integrity of zirconium-alloy pressure tubes.

Capacity Limits

The potential capacity of heavy-water-moderated pressure-tube reactors of the type described in this report is limited mainly by the sizes of the primary cooling loop equipment and by the number of such loops that can be accommodated around the reactor in a practical arrangement, rather than by the size of the reactor itself.

The maximum capacity of an individual cooling loop is judged to be about 900 Mwt within the foreseeable future, and the maximum number of such loops is about ten or eleven. Thus, the plant capacity limit is about 10,000 Mwt. On-power refueling, which is economically important to such large plants, may set roughly the same limit. A refueling machine is probably limited by speed of operation to about 5000 Mwt, and the use of more than two machines does not seem feasible. The reactor structure itself, on the other hand, has no capacity limitation in this range, although the fabrication and erection problems of the large calandria and shield structures and of the numerous coolant pipe connections to the pressure tubes become progressively more difficult as the size increases.

Development Work

The major areas in which mechanical development work is required are fuel handling equipment, pressure-tube assemblies, and joints and seals, such as those for the pressure-tube closures. All of these items are applicable to any size reactor of the types described, except for on-power fuel handling equipment, which is economically justifiable only for large reactors (above about 1500 Mwt).

Costs

No cost evaluations were made in this study; therefore, the estimates of capital costs made in earlier Du Pont studies cannot be confirmed. In our judgment, the particular facilities examined in this study should be no more costly to construct than the corresponding facilities on which the earlier studies were based. There are indications that unit capital cost (dollars per kilowatt) will decrease only slightly as plant size is increased over the range covered in this study.

RECOMMENDATIONS

If a heavy-water reactor program leading toward the construction of large power plants is to be pursued, the following steps should be taken next, with respect to engineering development:

1. A current appraisal of the plant cost should be made for one of the four cases described. This should preferably be the 3500-Mwt heavy-water-cooled reactor, which is the most extensively developed concept. Estimates of plant costs for the other cases may be made later, if desired, based away from this estimate.

2. Experimental development work should proceed on pressure-tube assemblies and joints and seals, directed toward specific designs. Other areas of development should be deferred until a more definite objective of designing and building a plant is in sight.

DISCUSSION

I. BACKGROUND INFORMATION

A. Previous Studies

This study is part of a continuing program for development of the technology of heavy-water-moderated reactors for electric power generation. The Du Pont Company has participated in this program on behalf of the United States Atomic Energy Commission since 1956. The program is administered by the Savannah River Operations Office of the AEC and is performed under the prime contract between Du Pont and the AEC for operation of the Savannah River Plant and Savannah River Laboratory. Within Du Pont, the work is conducted by the Atomic Energy Division of the Explosives Department, with assistance from the Engineering Department on appropriate phases.

The principal efforts under the program have been the development of reactor physics data and fuel element technology for this type reactor. For the latter purpose, a fuel testing reactor was built at the Savannah River Plant (the Heavy Water Components Test Reactor, or HWCTR). This reactor went into operation in 1962. In addition, engineering studies and economic evaluations have been performed from time to time.

The first series of engineering studies and economic evaluations took place from 1956 through 1959 and was concerned initially with reactor plants of 100-Mwe size; larger sizes (up to 460-Mwe) were considered as the study progressed. Pressure-vessel and pressure-tube reactors were considered, and several coolants, including pressurized heavy water, boiling heavy water, heavy-water steam, and helium were investigated. The results were reported in references*(2) and (3). These studies did not lead to selection of a preferred reactor design concept; but they provided a basis of engineering information for the present study.

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* Numbers in parentheses refer to the references listed in the Bibliography on pages 94-98.

The economic evaluations performed during that period made plain that heavy-water-moderated reactors in the 100- to 460-Mwe size range then under consideration could not generate power competitively with light-water reactor plants or conventional steam power plants. However, they suggested a more favorable relative position at larger sizes, because the heavy-water plants are characterized by a relatively high capital cost and low fuel costs. At larger plant capacities the benefits from low fuel costs would be expected to outweigh the charges on the high capital investment.

Sargent and Lundy, Engineers, worked in cooperation with Du Pont on further studies starting in 1959 and continuing to the present. The Sargent and Lundy work was performed under a separate contract with the AEC (8-18).

In 1961 and 1962, the Savannah River Laboratory developed a computer program to make economic evaluations of heavy-water-moderated power plants and to facilitate optimization of basic design parameters for such plants (4). Information and equations for the evaluation of capital costs were supplied by the Engineering Department and Sargent and Lundy. The program was initially written for a 300-Mwe plant, and was based on a pressure-tube reactor concept (one of the concepts developed in the earlier engineering studies) and on pressurized or boiling heavy-water coolants. Later the program was extended to 500- and 1000-Mwe plants, and to organic coolants. In making these extensions, it was assumed that the reactor concepts that had been developed for the original 300-Mwe plant were feasible for these larger plants.

By use of the computer program, evaluations were made during 1962 and 1963 of the costs of power from 500- and 1000-Mwe power plants, which tended to confirm the belief advanced earlier that these plants would be competitive with other types of power plants, provided that the assumptions as to the feasibility of the particular design concepts used for the program were justified (5). The heavy-water-cooled and the organic-cooled reactors gave the most favorable costs, with the organic-cooled reactor slightly lower than the heavy-water-cooled reactor. From a short-term viewpoint, however, the apparent cost advantage of organic cooling was outweighed by the more advanced technological status of heavy-water-cooled reactors.

At about this same time, two other developments created fresh interest in heavy-water-moderated power reactors. The first was the Seaborg report to the President on the Civilian Reactor Program, in November 1962. This emphasized the need for advanced converter reactors which would make more efficient utilization of fissionable materials than existing types of power reactors during the period until breeder reactors would become self-sustaining on their own output (49). The second was the Oak Ridge National Laboratory proposal for using very large reactors in dual purpose plants to produce electricity and distill sea water (50). Heavy-water-moderated reactors are particularly well suited to both of these uses. They have good neutron economy and can be made to provide a high conversion ratio and perhaps to breed (with a core designed appropriately for this purpose). Their advantage for furnishing heat for desalination lies in their low fuel cost and potential capability for being built in very large sizes.

Du Pont has made preliminary evaluations of thorium-fueled heavy-water-moderated power reactors for breeding (6), and is currently making an engineering study and cost estimate of a 300-Mwe prototype reactor capable of operating on uranium or thorium fuel. Du Pont and Sargent and Lundy have also participated in studies of dual-purpose reactors for electricity and desalination by contributing information on 3500-Mwt and 8300-Mwt heavy-water-cooled-and-moderated reactors to the Bechtel Corporation, as part of an over-all study requested by the Office of Science and Technology and conducted by a subcommittee of representatives from the various government agencies concerned (7). Sargent and Lundy independently also made a study of such plants for ORNL (19).

These various developments revealed a need for more detailed engineering examination of the large reactors considered in the studies referred to. In 1963 the Atomic Energy Commission requested Du Pont to perform detailed engineering feasibility studies of the pressure-tube heavy-water-moderated reactor concept, in sizes of 3500 and 8300 thermal megawatts, and with heavy-water and organic coolants. To avoid unnecessary complications, the study was based on a plant for electrical power generation only, without special consideration of breeding or desalination, since such considerations would lead to relatively minor design changes having little bearing on the basic feasibility questions.

By direction of the AEC, an interim report on this study was furnished in August 1963, covering principally the reactor structure only, for heavy-water coolant. The study has continued into other aspects of the reactor plants that are significant with respect to feasibility and into the organic-coolant cases. It has been completed, and this is the final report.

B. Objective of Study

The objective of this study was to develop a particular design concept for 3500-Mwt and 8300-Mwt heavy-water-moderated power reactors, cooled by heavy water or an organic liquid, in sufficient detail to form a judgment of their feasibility. Feasibility means that the plants can be designed and built as conceived, that the power costs will be in an economically competitive range, and that the plants can be operated safely and with the intended performance.

C. Scope of Study and Course Followed

Only those aspects of the plant which raised significant feasibility questions were considered in any detail. These include principally the reactor structure, the primary cooling system, the fuel handling system, reactor containment, the turbine-generators, and a few broad questions relating to costs and safety, such as heavy-water losses and reactor controllability. The plant descriptions are treated from this point of view and are not intended to be comprehensive descriptions of facilities.

The study was confined to preliminary design development and engineering analysis of the features mentioned. No experimental development work was performed and no cost estimates were made.

The background of information developed in the previous studies described in Section I, A, was utilized in this study, as was the extensive technology on heavy-water reactors available from the design, construction, and operation of the Savannah River production reactors. Technology developed in connection with other heavy-water reactors in this country and Canada has also been very helpful. These are the Nuclear Power Demonstration Reactor (NPD) in Ontario (21), the Carolinas-Virginia Tube Reactor (CVTR) in South Carolina (22), the Plutonium Recycle Test Reactor (PRTR) at Hanford (23), and the Heavy Water Components Test Reactor (HWCTR) at Savannah River (24), all of which are in operation; and the Douglas Point Station reactor (CANDU) in Ontario (20), which is in an advanced stage of construction. For a recent broad survey of heavy-water reactors, see reference (1).

As Du Pont has no background of experience on organic-cooled reactors, all information for these cases is based on the work performed by Atomics International Division of North American Aviation and by AECL, and on work done in connection with the Experimental Organic-Cooled Reactor (EOCR) at the National Reactor Testing Station (25) and the Piqua Organic Moderated Reactor Plant at Piqua, Ohio (26).

On some major equipment items, we consulted with major manufacturers; these include mainly the primary coolant circulating pumps and steam generators, and, through Sargent and Lundy, Engineers, the turbine-generators. However, the conclusions drawn regarding these equipment items are our own, and, in the case of the turbine-generators, those of Sargent and Lundy.

The basic design parameters for the plants were established initially by the use of the Savannah River Laboratory Computer Program (4), with modifications as the study progressed.

II. PLANT DESCRIPTIONS

Reference designs for four reactor plants are described in this Section, as follows:

A. 3500-Mwt D_2O-Cooled

B. 3500-Mwt Organic-Cooled

C. 8300-Mwt D_2O-Cooled

D. 8300-Mwt Organic-Cooled

The descriptions focus on plant features considered significant with respect to feasibility. Plant facilities of a conventional nature or which do not raise feasibility questions are given little or no attention. The features described are illustrated in the Figures. The important design parameters of the plants are summarized in Table 1.

The four plants have the following basic design similarities.

1) The plants are designed solely for economic generation of electricity.

2) The plants operate on an indirect closed cycle in which heat is transported from a reactor to steam generators by a circulating primary coolant stream which is hot pressurized heavy water in Cases A and C and which is a low-vapor-pressure organic mixture in Cases B and D. The secondary heat transport system is also on a closed cycle in which light water steam raised in the steam generators flows to electric turbine-generators, is condensed, and is returned to the steam generators with regenerative feedwater heating. In the heavy-water-cooled cases, the turbines

operate on saturated steam; in the organic-cooled cases on superheated steam. Refer to flow diagrams, Figures 1, 17, 26, and 27.

3) The reactor is moderated with heavy water and fueled with low-enrichment uranium. The reactor is a vertical pressure-tube type in which the primary coolant flows through numerous pressure tubes containing the fuel. The moderator is contained in a calandria tank at low temperature and pressure, and completely segregated from the primary coolant.

4) Reactor refueling may be accomplished without shutting down the plant, if this is economically justifiable.

5) The reactor and its primary cooling system, fuel handling facilities, and certain other reactor auxiliary systems are enclosed in a spherical steel containment shell.

In addition to these common basic features, the plants have many similarities of detail which are noted in the descriptions that follow.

A. 3500-Mwt Heavy-Water-Cooled Reactor

1. FUEL AND CORE

The design of the fuel elements and the core for this reactor (and the other reactors to be described) was made by the Savannah River Laboratory and lies outside of the scope of this study, which is concerned with the permanent components of the plant. However, a brief description of this design will aid in understanding the other facilities to be described.

The fuel elements are tubes of uranium oxide, enriched to 1.2% ^{235}U, clad in Zircaloy-2. A fuel assembly, shown in cross-section in Figure 6, consists of three such tubes of various diameters nested in a Zircaloy housing tube, with end fittings attached, the whole contained in a Zircaloy pressure tube. The assemblies are approximately 20 feet long with a 15-foot active section; the outermost fuel tube is approximately 3-1/2 inches OD. From a fabrication standpoint, the fuel assemblies may be made as full length units; however, from the standpoint of handling them in and out of the reactor, it may be desirable to make them in two or three segments, particularly if on-power refueling is used.

A fuel tube is fabricated in the following steps:

1) Zircaloy-2 sheath tubes are extruded and drawn to 20-mil wall thickness;

2) Crushed, fused, and out-gassed uranium oxide is loaded into the space between a pair of sheath tubes, held concentric with temporary end plugs;

3) The tube is vibratory compacted to 85% of theoretical density and swaged to 90-92% of theoretical density;

4) The tube is cut to proper length and counterbored;

5) End plugs containing expansion chambers for fission product gases are welded to the tube; and

6) Zircaloy-2 spacer ribs are welded to the sheath tubes.

A set of three tubes is nested, and end fittings are attached to form a complete fuel assembly. The assembly is designed for coolant flow over both surfaces of each fuel tube.

The reactor core consists of 516 such fuel assemblies, each in a Zircaloy-2 pressure tube, arranged in a 10-inch square-pitch pattern, as shown in Figure 4. This spacing is slightly greater than the optimum spacing from a physics standpoint, but is the minimum adequate for installing the inlet and outlet coolant pipes.

The reactor lattice contains two zones, a central flat zone and an annular buckled zone. The flat zone contains control rods in lattice positions, as shown in Figure 4, which maintain a flat radial flux distribution in the zone. The outer buckled zone contains no control rods and has a positive buckling of the magnitude required to maintain criticality. Surrounding the lattice is a heavy-water reflector zone which is 24-inches-thick above and below the core, and 20-inches-thick radially.

The design of the fuel and the core is based on physics parameters that were experimentally determined by the Savannah River Laboratory and on various hydraulic, thermal, and metallurgical limits on operating conditions that derive from current fuel studies at Savannah River Laboratory. The operating limits that have been specified for this study may be extended upward as further testing results provide a firm base for doing so. The principal limits are on the fuel heat rating ($\int k d\theta$), cladding surface temperature, safety factor on heat transfer burnout, and flow velocity.

The fuel heat rating determines the central uranium oxide temperature. If excessive, dimensional changes and distortions of fuel elements caused by release of fission product gases from the UO_2 particles and plastic deformation of the UO_2 can lead to cladding failures, either directly by overstressing or indirectly through development of hot spots on the wetted surfaces. The limit selected for design is about 40 watts/cm. Because of burnup of fissionable isotopes, the heat rating of each fuel assembly will decrease during its exposure in the reactor, so that the value for the maximum rated fuel, averaged over its exposure life, is about 35 watts/cm. The maximum fuel temperature is around 1750°C.

The wetted surface temperature of the Zircaloy cladding is currently limited to about 330°C, on the basis of data on "breakaway corrosion" and hydriding of Zircaloy.

The design dimensions of the fuel shown in Figure 6 are calculated to achieve the following principal objectives.

1) The minimum burnout safety factors on heat flux from each fuel tube at the maximum fuel heat rating are equalized. In this design, the maximum heat flux is 500,000 pcu/(hr)(ft^2), and the minimum burnout safety factor is 1.7, based on the SRL correlation for burnout heat flux and a "hot-spot factor" of 0.7, which is an estimated value based on SRL experience with this type of fuel.

2) The coolant temperatures are as high as possible consistent with the corrosion limit on the temperature at the wetted surface of the cladding. The coolant flow rates and temperature rise through the fuel are adjusted to achieve an economic balance between the costs associated with flow and the costs associated with temperature. Also, the maximum coolant velocity in any fuel subchannel is held to about 50 ft/sec, which is considered to be a reasonable extension of present technology.

3) An economic balance is achieved between the costs associated with fuel enrichment, fuel exposure life, and the physical size of the reactor. This balance determines the fuel enrichment, average exposure, total number of fuel assemblies, and number of flat-zone assemblies, as listed in Table 1.

2. REACTOR STRUCTURE

a. General Arrangement

The reactor proper, shown in Figures 3, 4, and 5, comprises a vertical calandria with pressure tubes passing through the calandria tubes. The calandria contains the heavy-water moderator at essentially atmospheric pressure, and the pressure tubes contain the fuel and heavy-water coolant at high temperature and pressure. Above and below the calandria are axial shields, and the calandria is also surrounded by radial thermal shield tanks. All of the shields are cooled by light water. Below the lower axial shield and above the upper axial shield are coolant headers from which coolant inlet and outlet pipes run to the ends of each pressure-tube assembly. This entire reactor complex is 54-1/2 feet high by 34 feet in diameter.

The space enclosed by the shields, outside of the calandria, is flooded with carbon dioxide at slightly more than atmospheric pressure, and is referred to as the gas space. This gas also fills the annular space between the pressure tubes and the calandria and axial shield tubes.

The space above the reactor, containing the coolant outlet piping and header, is called the Upper Header Room and contains air which during reactor operation is near the primary system temperature, as is the piping and header space immediately below the reactor. Both spaces are enclosed by thermal insulation. The hot air in these spaces is confined and circulated to recover heavy water.

The area below the reactor complex is occupied by the refueling machine. Surrounding the reactor radially is a massive concrete biological shield from which the reactor structures are supported. Spanning the reactor complex overhead is a concrete shielding floor on which the control and safety rod drive mechanisms are located. In the construction of the reactor complex, it will be necessary to install the calandria and shields before this floor and the upper part of the containment building are constructed; and these structures will not be removable.

b. Calandria

The calandria (Figure 3) is the atmospheric-pressure vessel which contains the heavy-water moderator. It is a closed vertical tank 25 feet 3 inches inside diameter by 20 feet 2 inches high, containing a pattern of tubes that enclose the reactor components. The shell and end plates are type 304 stainless steel, and the tubes are Zircaloy-2.

There are 553 tubes (for 516 fuel positions and 37 control rod positions) arranged on a 10-inch square-pitch lattice pattern in a 22-foot-diameter region. The tubes have a 0.060-inch wall thickness, based on external hydraulic pressure. There are also 40 smaller interstitial tubes for safety rod positions and several additional interstitial tubes for in-core instruments. The 20-inch-wide annular region of the calandria contains only heavy water and acts as a neutron reflector.

The calandria contains a thermal shielding plate near the bottom to protect the bottom plate and the lower axial shield against radiation heating. This shielding plate also serves to form a heavy-water inlet plenum at the bottom of the calandria; holes in the plate are arranged to produce a uniform distribution of flow through the cross section of the calandria. A similar arrangement of thermal shielding plate and outlet plenum is provided at the top of the calandria.

c. Axial Shields

The upper and lower axial shields (Figure 3) are the main supporting structures for the reactor complex, and their design is dictated as much by this function as by their shielding function. The shields are of shell and tube sheet construction, with tubes in positions matching those of the calandria. Material of construction is type 304 stainless steel. The space inside the shields contains stainless-steel plates and light water to reduce the neutron flux outside the shield during reactor operation to below "machine tolerance" [10^5 n/(cm^2)(sec)], at which level the induced radiation from steel shortly after reactor shutdown is less than 1 mr/hr. The shields also contain water distribution piping and baffles.

The shields are 29-1/2 feet diameter by 30 inches deep, constructed with 1-inch-thick shells and end plates, and tubes of 4.87 inch inside diameter by 1/2 inch wall thickness. The tubes extend approximately 6 feet above the top end plate of the upper shield and an equal distance below the bottom end plate of the lower shield.

The upper shield supports the pressure tube assemblies and fuel. The lower shield supports the calandria and the radial thermal shields. Weights and loads on the shields are as follows:

	Loads, tons	
	Upper Shield	Lower Shield
Dry weight	205	205
Live load	415	604
Total load	620	809

Each shield is independently supported from the radial biological shield structure.

d. Radial Thermal Shield

The radial thermal shield (Figure 3) consists of an annular array of about six separate tanks 11 inches thick by 20 feet high, which are set around the outside of the calandria. They are constructed of type 304 stainless steel and are filled with light water and stainless-steel plates in equal parts by volume. Welded staybolts join the inner and outer shell plates. The function of this shield is to reduce the radiation energy flux incident on the concrete biological shield to a maximum of about 20 pcu/(hr)(ft^2) during reactor operation.

e. Pressure-Tube Assemblies

The pressure-tube assemblies are shown in Figure 5. There are 516 of these assemblies, each consisting of a Zircaloy tube 18 feet long, a stainless-steel extension about 8 feet long on each end, and transition sections and end fittings, for an over-all size of 40 feet long by 4.08 inches inside diameter. The assemblies are designed for an internal pressure of 2000 psig at 320°C.

The Zircaloy tubes are seamless Zircaloy-2 (ASTM B-353, Grade RA-1) extrusions cold-drawn about 25% to provide a minimum ultimate tensile strength of 49,200 psi at 320°C. It is expected that these tubes will be designed in accordance with the rules and criteria of the ASME Nuclear Code, which will permit (for Zircaloy) an allowable membrane stress of one-third of the ultimate tensile strength, or 16,400 psi, requiring a wall thickness of 0.27 inch.

The pressure-tube extensions are made from seamless pipe, ASTM A-312 type 316 alloy steel, machined to the same inside and outside diameters as the Zircaloy tubes. Between the extension and the Zircaloy tube is a transition section, consisting of a tandem-extrusion Zircaloy to stainless-steel joint which is welded to the Zircaloy tube and the stainless-steel extension.

At each end of the pressure-tube assembly is a short stainless-steel end fitting containing a mechanical closure with a pressure seal. The closure for the lower end fitting will be designed for operation by the refueling machine. The closure for the upper end fitting will be of a relatively simple design for manual operation. Each end fitting includes a side-entering 3-inch connection to which a coolant inlet or outlet pipe is welded. The upper end fitting is shouldered to rest on the upper shield tube extension, which supports its weight. A gasketed seal is provided at this point between the end fitting and the shield tube extension. The lower end fitting makes a sliding fit in the lower shield tube extension, and is connected to it with a bellows seal to permit approximately 1-3/4 inches of longitudinal expansion in the pressure-tube assembly.

The pressure-tube assemblies are semipermanent components of the reactor; that is, they are designed to be replaceable with the reactor shut down but without any major dismantling of the reactor structure or piping. The sequence of installation is as follows: a pressure-tube assembly, complete except for the upper end fitting, is inserted from below the reactor through the shield and calandria tubes; the upper end fitting is welded to the stainless-steel extension by a full-penetration radiographable butt weld made from inside the tube; the bellows seal is welded to the lower end fitting; and the coolant pipes are welded to the stubs on the end fittings. For removal of an assembly this sequence of operations is reversed, and the field-welded joints are broken using automatic cutting equipment.

f. Coolant Headers and Piping

The reactor is designed for upflow cooling, with an inlet header below the reactor and an outlet header above it (Figure 3).

From these headers, supply and discharge pipes run to the ends of each pressure-tube assembly, as shown in Figures 3 and 4. Headers and piping are carbon steel.

Each header is a torus 32 feet in axial diameter by 28 inches outside pipe diameter, with 2-inch wall thickness. Both headers are supported and guided so as to permit radial thermal expansion; in addition, the lower header is spring-supported to permit vertical motion in response to the expansion of the pressure-tube assemblies.

The nozzles for the coolant pipe connections are arranged on an approximately 12-inch triangular pitch covering 180 degrees of the pipe circumference.

The coolant pipes connecting the headers to the pressure-tube assemblies are 3-inch (3-1/2 inches OD) carbon steel pipes with a short stainless-steel end section for welding to the end fitting on the pressure-tube assembly. The maximum velocity of flow through these pipes is 35 ft/sec. The pipes cross the face of the axial shield in lanes between the shield tube extensions with up to thirteen pipes in a lane, vertically one above another. The pipe runs are designed to provide enough flexibility to allow for differential displacements of the individual pipes and header without imposing excessive stresses on the piping or excessive reaction forces or movements on the pressure-tube assemblies.

The entire assemblage of supply piping and inlet header is insulated from the lower axial shield above it and the refueling room below it. The pipes are not individually insulated. A similar arrangement is employed for the outlet piping, where the entire space between the upper axial shield and the concrete floor above it is insulated.

3. PRIMARY COOLING SYSTEM

a. General Arrangement

The primary cooling system consists of six loops each containing one steam generator, one circulating pump, two block valves, and connecting piping, through which hot pressurized heavy water flows from the reactor outlet header to the reactor inlet header. The material of construction of all of the piping and most of the equipment is carbon steel. The heavy water will be maintained in an alkaline and reducing condition by chemical treatment to minimize corrosion. The operating conditions for the loops are shown in the flow diagram, Figure 1.

Each loop circulates 58,300 gpm of heavy water at a reactor inlet header condition of 267°C and 1740 psia and removes 591 thermal megawatts (1.1×10^9 pcu/hr) of heat. The system design pressure is 2000 psig.

An arrangement of the loops in the Reactor Building is shown in Figures 12, 14, and 15. This is only one of several arrangements under consideration as discussed in Section III, C, 7, b, "Design Problems."

b. Pumps

The pumps are vertical overhung single-stage centrifugal pumps with mechanical face-type shaft seals. Each pump delivers 58,300 gpm of heavy water at 267°C at a total dynamic head of 530 feet. Each pump is direct-driven by an 1800 rpm 10,000 hp motor. A flywheel is mounted on the shaft between the pump and motor to provide inertia to maintain adequate flow during the period immediately following a failure of the motor.

The pumps are 24" x 30" x 30", measure about 7-1/2 feet by 5-1/2 feet over the casing, and are about 9 feet high; each weighs about 30 tons. The material of construction of the structural parts, including the casing, is carbon steel. Other materials are used for special parts such as the shaft, impeller, and seal. The motors are 5-1/2 feet diameter by 10-1/2 feet high and weigh about 26 tons each.

The pumps require a minimum net positive suction head (NPSH) of about 110 to 200 feet, depending on the particular design.

c. Steam Generators

The steam generators are vertical U-tube natural recirculation boilers with integral moisture separating equipment. Each boiler consists of a single shell, as shown in Figure 12, in which the bottom head is the water channel, the 12-foot-diameter cylindrical section is the boiler section containing the U-tubes, and the 16-foot-diameter upper section contains the moisture separating equipment. The primary, or D_2O side, is inside the tubes. Steam is generated on the shell side in the boiler section, flows through the separating section, and leaves the top of the generator as saturated steam. Recirculating water and water collected in the separator section flows down through the annular space between a shroud surrounding the tube bundle and the shell, to enter the tube bundle just above the tube sheet. Boiler feed water is introduced into the annular downcomer space to mix with the recirculating water.

The generators are carbon steel with either carbon steel or Inconel tubes. If Inconel tubes are used, the water channel will be clad with Inconel by overlay welding. Each generator contains approximately 12,000 tubes, 1/2 inch OD by 45 feet long, providing a heat transfer surface area of 70,600 square feet. The tube sheet is about 24 inches thick and is drilled for tubes on a 3/4-inch triangular pitch. The units will be designed under the ASME Nuclear Code for a design pressure of 2000 psig on the primary side and 750 psig on the secondary side. Each generator has a dry weight of 330 tons.

d. Piping

The piping for the primary cooling system is 28 inches OD x 24 inches ID, and is made from carbon steel plate. The maximum flow velocity is 46 ft/sec. Each loop contains two motor-operated gate valves for isolating the pump and steam generator from the reactor inlet and outlet headers.

4. SECONDARY COOLING SYSTEM

The secondary cooling system consists of a single loop with one turbine-generator which handles the entire steam flow of 14,300,000 lb/hr. Operating conditions are shown in Figure 1.

Steam from the steam generators in the Reactor Building, at 665 psig saturated (500°F), is brought out through the containment shell in six separate lines provided with isolation valves, to a header from which it flows to the turbine-generator building at a throttle pressure of 635 psig.

The turbine-generator is a single-shaft machine with a tandem-compound, six-flow, 44-inch last-stage blade length turbine driving a nominal 1000-Mwe generator. The generator is cooled with 60-psig hydrogen and generates at 26,000 volts. The over-all length of the turbine-generator is 202 feet. The heaviest component during erection is the generator inner frame (300 tons) and after erection is the generator rotor (205 tons).

The steam flows from the turbine to a condenser operating at 1.5 inches Hg absolute; the condenser requires 1,000,000 gpm of cooling water at 65°F. Condensate is pumped to feed water heaters and then back to the steam generators.

5. REACTOR COMPONENTS HANDLING SYSTEM

The reactor components handling system for fuel and other reactor components is partly inside and partly outside the Reactor Building. The fuel handling facilities within the Reactor Building are shown in Figure 7, 8, and 9 for a concept of refueling from below the reactor. An alternate scheme of refueling from above the reactor is shown in Figures 10 and 11. The facilities shown in these figures are dimensioned on the assumption that the fuel would be handled in one-third length segments. The principal facilities are described below, based on handling half-length segments. (See note in Section III, E.)

a. Assembly Area

In this area, outside the Reactor Building, fuel tubes and housing tubes are received, stored, unpacked, inspected, assembled into nested fuel assemblies, flow tested, and stored for delivery to the Reactor Building. Operations in this area are performed directly and semimanually.

b. New Fuel Transfer Area (Figures 8 and 9)

In this area, located in the Reactor Building, fuel assemblies are loaded into transfer tubes from which they may be accepted by the Fuel Handling Machine. The assemblies are handled horizontally throughout. After it is loaded with fuel, a transfer tube is filled with heavy water, pressurized to reactor operating pressure, and heated to reactor operating temperature. Several tubes are provided, operating on staggered cycles of pressurization, heating, cooling, and depressurization. Operations in this area are conducted directly because no radiation problems are involved.

c. Fuel Handling Area

This area is located in the Reactor Building below the level of the reactor, as shown in Figures 13 and 15, and is the area in which the Fuel Handling Machine operates.

The Fuel Handling Machine, Figure 7, is a pressure vessel containing a rotating magazine for storage of fuel and fixtures and a ram mechanism, mounted on a carriage in such a way that the axis of the vessel can be rotated to either a vertical or a horizontal position. The vessel may also be moved axially relative to the carriage through a distance of 5 feet. The machine itself is unshielded; it operates in a shielded area; and it will be designed for completely remote, automatically controlled sequential-programmed operation. The pressure vessel of the machine is filled with heavy water and is maintained at reactor inlet temperature and pressure at all times during its operation. Auxiliary systems for the machine, not shown in the figures, are required to maintain and control temperature, pressure, heavy-water composition, etc.

The operating cycle of the Fuel Handling Machine is as follows:

1) The machine, with the vessel axis horizontal, as shown in Figure 8, moves on a traveling bridge which positions it in line with a new fuel transfer tube which contains a new fuel assembly in a hot pressurized condition.

2) The vessel advances so that its snout engages and couples to the transfer tube.

3) The ram advances, delatches the seal plug at the end of the vessel snout, withdraws it, and deposits it in the magazine. The ram retracts, the magazine rotates to a second position, and the ram removes and stores the seal plug of the transfer tube in a similar fashion.

4) The magazine rotates to a fuel storage position and the ram advances, engages a half-length fuel assembly in the transfer tube, withdraws it and deposits it in the magazine, and retracts. The magazine rotates to a second fuel storage position and the ram withdraws the second half of the assembly into the magazine in the same way.

5) The ram replaces the plugs for the transfer tube and Fuel Handling Machine in the reverse sequence to step 3).

6) The snout disengages from the transfer tube and the vessel retracts.

7) The bridge moves the Fuel Handling Machine to the central position. The carriage drives off the bridge and onto the rotating bridge in the Lower Header Room.

8) As shown in Figure 7, the vessel rotates to a vertical position.

9) The bridge and carriage locate the machine below a fuel position.

10) The vessel advances upward and engages a pressure tube assembly.

11) The ram removes the seal plugs on the snout and pressure tube as in step 3). It then removes the muff from the pressure-tube assembly.

12) A fuel assembly is removed into the third and fourth storage positions in the magazine in a manner similar to step 4).

13) The new fuel assembly is charged into the pressure tube in the reverse sequence to step 12).

14) The muff and plugs are replaced in the reverse sequence to step 11).

15) The vessel disengages from the pressure tube assembly and retracts. The bridge rotates and the carriage moves to the horizontalizing position. The vessel horizontalizes and the machine moves out onto the traveling bridge.

16) The bridge positions the machine in line with a spent transfer tube which is in a hot pressurized condition. The machine repeats steps 2) and 3).

17) The spent fuel assembly is deposited in the transfer tube in a reverse sequence to step 4).

18) The machine repeats steps 5) and 6). The cycle then starts over as in step 1).

The cycle time for the machine is not known, but the available time on the basis of an average fuel exposure of 15,000 megawatt days per metric ton is 13 hours per assembly, which is judged to be more than adequate.

d. Spent Fuel Transfer Area

In this area, Figures 8 and 9, spent fuel received from the Fuel Handling Machine is transferred out of the tubes into a water-filled canal and thence through a lock to the Spent Fuel Storage Area. The operations are functionally similar to those in the New Fuel Transfer Area but are carried out remotely in the shielded area.

e. Spent Fuel Storage Area

In this area, located outside the Reactor Building, irradiated fuel assemblies are transferred to underwater storage, disassembled, and loaded into shipping casks for off-site shipment.

f. Fuel Handling Machine Maintenance Area

This area is located in the Reactor Building adjacent to the Fuel Handling Area, as shown in Figures 13 and 15. It is used for inspection, checkout, decontamination, maintenance, and removal of the Fuel Handling Machine. Spare component assemblies will be provided to minimize out-of-service time for the machine.

g. Shutdown Components Handling

Reactor components other than fuel, including pressure tubes, control and safety rods, and instruments, are removed with auxiliary casks and equipment while the reactor is shut down.

h. Alternate Fuel Handling System

An alternate fuel handling system is shown in Figures 10 and 11. The principal differences are that the Fuel Handling Machine operates from above the reactor and the fuel assemblies are handled only in the vertical position within the Reactor Building.

New fuel assemblies are brought into the Reactor Building and placed in vertical storage racks, from where they are charged to the new fuel transfer tubes by a transfer dolly. The fuel handling machine operates in a shielded canyon on the elevation +77' floor over the reactor, and functions in the manner previously described except that it is not necessary to horizontalize the pressure vessel.

Spent fuel transfer tubes located below the +77' floor receive irradiated fuel from the Fuel Handling Machine. The fuel is removed from these tubes by a transfer dolly and placed on a carriage which operates in a transfer canal and conveys it to the outside Spent Fuel Storage Area.

A heavy-duty bridge crane, operating over the top of the Fuel Handling Machine Canyon, is provided for emergency operations.

6. CONTROL AND SAFETY ROD SYSTEM

The reactor is provided with 37 control positions, which are lattice positions uniformly distributed through the flat zone of the core, as shown in Figure 4. Each of these positions contains a cluster of four rods for power level control and flux shaping, plus one shim rod. The control-rod complement during reactor operation consists of partial-length rods located approximately at the reactor midplane and other full-length rods either fully inserted or dipping in from the top. The axial power distribution is maintained by slight movements of the partial-length rods. The radial distribution is controlled by the relative movement of the control rods moved either individually to trim out local perturbations, or moved in gangs to adjust the flux distribution over larger areas.

The reactor also contains 40 positions for single safety rods, located interstitially to the fuel pattern, as shown in Figure 4.

All of the control clusters and safety rods are in cooling tubes which pass through the calandria tubes and axial shield tubes, as shown in Figure 3. In the case of the bottom refueling scheme, described in Section A, 5, c, the cooling tubes extend above the +95' floor over the reactor, where the rod drives are located. The rod drives are of the rack and pinion type, the racks being provided with extensions which are delatchable from the rods themselves. The safety-rod drive shafts are provided with electromagnetic clutches which release on loss of electrical supply and permit the rods to drop into the core by gravity.

In the case of the top refueling scheme, described in Section A, 5, h, the rod drives are located below the reactor, as shown in Figure 11, and a stored energy source such as springs or hydraulic pressure would be used to drive in the safety rods.

7. REACTOR AUXILIARY FLUID SYSTEMS

In addition to the primary cooling system, the reactor requires a number of auxiliary cooling systems and other liquid and gas systems to function properly. Some of these are indicated on the flow diagrams, Figures 1 and 2.

a. Pressure-Volume Control System (Figure 1)

The primary cooling system is provided with a gas pressurizer to control and limit normal fluctuations in the primary coolant pressure and volume, resulting mainly from temperature changes. The pressurizer vessel is connected to the reactor outlet header, is partially filled with heavy water, and is blanketed with helium. The normal pressure and volume fluctuations of the system are controlled by removing or adding heavy water to maintain a constant level in this vessel. Helium is added or removed only for pressurization and depressurization of the system on startup or shutdown, or to change the operating pressure level, or to adjust for slow losses of gas by leakage or accumulation by decomposition of coolant.

Protection of the primary cooling system equipment, against a pressure rise too rapid for the pressurizer to cope with, is provided by liquid relief valves which discharge to a quench tank. The quench tank may be either inside the containment shell or physically outside the shell but vented to the interior, so that functionally it is part of the containment space.

The secondary side of the steam generators is provided with standard three-element control systems for pressure and volume control, and with steam safety valves which also discharge to a quench tank.

b. Shutdown Cooling System (Figure 1)

A shutdown cooling system in parallel with the primary cooling system and having a capacity of 5% of the main cooling system is provided for removing decay heat from the fuel when the reactor is shut down.

c. Primary Coolant Purification System (Figure 1)

A purification system consisting of filters and ion-exchange resin beds is provided to maintain the purity and chemical composition of the primary coolant system and to remove radioactive contamination. The purification system is operated at low temperature (less than 40°C) and essentially atmospheric pressure. A purge stream of primary coolant, cooled and depressurized, provides the feed to the system. Part of the purified heavy water is pumped back to the primary coolant system pressure and used to cool the seals on the main circulating pumps. The remainder is reheated by heat exchange with the purge stream and pumped directly to the primary system. The system is sized to handle the contents of the primary system in two days.

d. Moderator Cooling System (Figure 2)

The moderator absorbs heat from gamma radiation and neutron thermalization, amounting to about 4% of the fission power, plus an additional 1% by transfer of sensible heat from the primary coolant in the pressure tubes. This heat is removed by circulating the moderator through external coolers to maintain a maximum temperature of 90°C. The heat is rejected to cooling water because its low temperature makes it unprofitable to utilize.

The calandria is vented through a low-pressure seal into the containment building to prevent excessive pressures in the event of a pressure tube failure followed by rupture of a calandria tube.

The equipment and piping for this system are carbon steel except for the calandria, which is stainless steel and Zircaloy-2.

e. Moderator Purification System (Figure 2)

Since the moderator cooling system is mainly mild steel, a purification system is provided to maintain its cleanliness and to maintain an alkaline pD condition. The system consists of filters and ion-exchange resin beds and operates on a small side stream from the moderator cooling system. It is sized to handle the contents of the moderator cooling system in two days.

f. Isotopic Purification System (Figure 2)

Both the primary coolant and the moderator tend to become degraded with respect to isotopic purity by absorption of H_2O from various sources. A common isotopic purification system is provided to maintain the isotopic composition of both systems above 99.75%. The system consists of two vacuum distillation towers with steam-heated reboilers. The overhead stream from the last tower, at a composition of about 20% equivalent H_2O, is shipped off-plant for recovery in a heavy-water manufacturing plant.

g. Control and Safety Rod Cooling System (Figure 2)

The control and safety rod tubes require cooling to remove the heat of nuclear reaction and, even more so, to remove the heat transferred from the upper and lower header spaces through which the tubes pass. This cooling is done with heavy water taken as a side stream from the moderator cooling system.

Downflow of coolant is used so that the coolant flow does not impede gravity dropping of safety rods. Inlet and outlet ring headers are provided, as shown on Figure 3, and individual coolant pipes connect the headers to the individual coolant tubes.

h. Shield Cooling System (Figure 2)

Heat is generated in the reactor radial and axial shields by absorption of radiation. In the axial shields, a much greater heat load is introduced by losses from the primary coolant system, as shown in the following table.

	Heat Loads, pcu/hr		
	Upper Axial Shield	Lower Axial Shield	Total
Radiation absorption	370,000	370,000	740,000
Sensible heat			
Through insulation	20,000	13,000	33,000
Conduction along tube extensions	200,000	170,000	370,000
Across gas gap within shields	10,000,000	8,200,000	18,200,000
Total - Axial Shields	10,590,000	8,753,000	19,343,000
Radial Thermal Shields			110,000
Total - All Shields			19,500,000 (10 Mw)

This heat is removed by a circulating system of deionized light water.

i. Dry Blanket Gas System

The interconnecting gas space between the calandria tubes and the pressure tubes, between the shield tubes and the pressure tubes, between the calandria end plates and the axial shields, and between the calandria shell and the radial thermal shields, is filled with an inert gas (probably carbon dioxide) to prevent the formation of radioactive gases and nitrogen oxides (by radiolytic action) in this space. This gas system is operated at a few inches of water gage pressure to prevent in-leakage of air.

The gas is circulated at a low rate, possibly with some cooling, through a dryer, and provided with moisture detection instrumentation and sampling means (for analyzing for heavy water). The system is vented through a low-pressure seal into the containment building proper or to the quench tank of the primary system pressure-volume control system.

j. Wet Blanket Gas System

Low-pressure heavy-water tanks required for the various systems described above are blanketed with an inert gas such as helium or nitrogen to minimize degradation of heavy water by moisture in the air.

k. Off-Gas System (Figure 2)

The moderator releases deuterium and oxygen by radiolytic decomposition of heavy water, so that purging of this gas is required to prevent build-up of deuterium gas. The purge gas is passed through a cooler to remove the bulk of the heavy-water vapor, then through a deuterium recombiner, then through moisture traps, and finally vented to the atmosphere by a stack. Gas bled off from the primary coolant pressurizer is also released through this system.

l. Gas Recovery System

The purge stream from the primary coolant system to the purification system, and other heavy water discharged from the primary coolant system for pressure-volume control, evolve helium and perhaps deuterium (if it is added to the primary system for oxygen suppression). This gas is released in the deionizer supply tank, compressed, and returned to the pressurizer.

m. Hot Air System

The upper and lower reactor header spaces containing the inlet and outlet cooling pipes (Figure 3) will be reasonably well-sealed spaces in which the air will be heated by the piping. This air will be circulated and monitored for heavy water, and possibly equipment will be provided for recovery of heavy water from this air.

8. REACTOR CONTAINMENT

The reactor enclosure is a spherical containment shell 250 feet in diameter of 1-1/4-inch-thick A-201 Grade B steel plate. The shell is designed for an internal pressure of 25 psig at 230°C and an external live load of 60 psf as a Class B vessel, in accordance with the ASME Nuclear Code. The internal design pressure is based on a postulated incident involving release of the entire contents of the primary coolant system plus the contents of the secondary side of one steam generator to the building, with no loss of heat from the building to the environment, no heat absorption by internal building structures, and no pressure suppression by sprays or other means. The shell diameter is governed by the equipment arrangement in the building, particularly the primary coolant loops.

The systems contained include the reactor itself, the primary coolant system (including the pressurization system and the shutdown cooling system), the moderator cooling system, and the fuel handling system, up to the point where spent fuel is safely under water in the transfer canal. Pressure relief valves on these discharge to the containment volume. It is considered admissible here to utilize one or more quench tanks physically outside the building, but vented to the building; provided that these tanks, and the lines connecting them to the building, have a design pressure not less than the building itself and are otherwise in accordance with the ASME Code requirements for the building. The quench tanks are then in effect an adjunct to the building. The pressure relief from the secondary side of the steam generator also vents to the containment building (in the same sense as described above for the primary system) and, to accomplish this, isolation valves are provided on the steam lines from the generators where they pass through the containment shell*.

Other systems are located inside the containment shell only to the extent dictated by construction costs, operating convenience, or similar factors. For example, the purification systems may be located in a building outside the reactor containment shell for operating convenience. A breach of the purification system has no effect on the primary system, and the radioactivity from such a breach can be confined by proper design of the purification area.

The reactor containment shell has the following usual features: (1) all lines passing through it and opening to the building or the contained systems in it have isolation valves; (2) electrical conduits and wires are sealed; (3) personnel access when the reactor is operating or pressurized is only through air locks. The building is expected to have a leakage rate of less than 0.1% per day of the contained volume at the design pressure.

Additional containment protection is provided by a water spray system and a halogen absorption system. The spray system consists of a network of piping with low-pressure spray nozzles distributed through the building, and supplied with water by gravity from a 200,000-gallon reservoir in the dome. Sump pumps in the basement of the building return this water through coolers to the reservoir, permitting operation of the system at 15,000 gpm for an indefinite period. This system removes reactor decay heat and lowers the building pressure, with attendant reduction in the leakage rate. It also reduces the concentration of radioactive contamination dispersed in the air in the building.

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*It seems logical to relieve the secondary side of the generators to the containment building; for, otherwise, the tubes of the generator, which are the thinnest part of the primary system wall, would be the only part of this wall whose breach would lead outside the containment building.

The halogen absorbers are integral units consisting of a blower, demister, and activated charcoal filters. Four units of 12,500-cfm capacity each are at suitable points in the building. These units handle an amount of air (and steam) equivalent to the building volume in two hours and have an absorption capacity equal to the maximum halogen burden in the fuel. The units draw from and discharge to the containment building and considerably reduce the radioactivity in air leaking from the building after the postulated incident.

The halogen absorbers and the pumps for the spray system will be designed to operate in the atmosphere of steam at about 110°C, corresponding to the postulated incident.

9. REACTOR BUILDING ARRANGEMENT

The structural arrangement and disposition of major equipment in the Reactor Building is shown in Figures 12 through 16, some features of which have been touched on in the previous description. The arrangement shown is for on-power refueling from below the reactor. (Note that the relative positions of the new and spent fuel transfer areas is reversed as compared with Figure 8.) Refueling from above the reactor involves some rearrangement, as indicated in Figure 11.

The Reactor Building steel shell extends 53 feet below grade level and 197 feet above grade level and is set on a concrete foundation extending 58 feet below grade. External steel columns support the lower portion of the shell during erection; these may be removed or left in place, as more detailed design dictates. Inside the building, loads are carried through concrete columns and walls, including a wall abutting the shell up to elevation +95 feet, to an interior subgrade foundation which transmits the loads directly to the exterior concrete foundation. Elevations in the building are referred to a Zero Elevation, which is approximately at grade level.

The reactor calandria is located near the center of the building, surrounded radially by a 9-1/2-foot-thick concrete shield (Figure 12). This shield, which permits personnel access to the adjacent process rooms during reactor shutdown, is thicker than needed for this purpose (about 6 feet is required); the additional thickness is dictated by structural reasons because of the need to cantilever this shield to provide space for the reactor inlet header.

The primary coolant loops are in two rooms on either side of the reactor shield, and are surrounded by 4-foot concrete shield walls (Figures 12 and 14) required because of the high-energy gamma radiation emanating from the water during reactor operation. A 4-foot concrete shielding floor spans the reactor and primary coolant systems at elevation +95 feet (Figure 12). The Fuel Handling Area, Spent Fuel Transfer Area, and Fuel Handling Machine Maintenance Area are similarly enclosed by shielding walls (Figure 15). This arrangement makes the floor above elevation +95 feet and the entire annular space outside the process rooms from elevation Zero to elevation +95', (except for the Spent Fuel Transfer Area) accessible to personnel during reactor operation as well as shutdown.

Nonradioactive equipment for which some surveillance and maintenance during reactor operation is desirable will be located in the annular space on various elevations. Access to the elevation +95' floor will be mainly for surveillance of the control-rod and safety-rod drives. The only operating work required in the building is new fuel handling, which is performed in an area at elevation +11' near the principal personnel entrance to the building (Figure 15).

An additional equipment area is provided at elevation -26' (Figure 16). Interior shielding walls in this area permit part of it to be used for mildly radioactive facilities and part for clean facilities.

The principal personnel access to the building is through an air lock at grade level, elevation Zero. An emergency exit air lock at the elevation +95' floor level leads to an outside staircase to the ground. A 5-ton-capacity freight and passenger elevator provides access to all levels from elevation -26' to elevation +95'. Two principal staircases on opposite sides of the building also provide access to all floors from elevation Zero to elevation +95', and other stairs are provided from elevation Zero to elevation -26' and between other levels, as required.

A 400-ton gantry crane capable of lifting any removable equipment in the building operates on the elevation +95' floor. The steam generators, weighing 330 tons each, may be replaced with this gantry, as shown in Figures 12 and 15. One of the 150-ton steel shield doors to the process area is raised by the gantry hoist, and a welded or bolted panel in the containment shell is removed. A generator is removed by lifting it slightly, removing its supports at elevation +67', and lowering and horizontalizing it onto two trucks on tracks at elevation Zero. It is then pulled out of the building by winches. A generator may be installed by the same procedure in reverse.

The reservoir (deluge tank) for the building spray system is supported from the building dome, and a platform is provided at this elevation on which are mounted gravity head tanks for pump seal water supply, etc. The platform is accessible from the gantry walkway.

B. 3500-Mwt Organic-Cooled Reactor

1. FUEL AND CORE

The fuel elements for the organic-cooled reactor are tubes of uranium monocarbide enriched to 1.2% ^{235}U, clad in SAP (Sintered Aluminum Product), which is a mixture of 4 to 12% by volume of aluminum oxide dispersed in aluminum. A fuel assembly, shown in cross section in Figure 22, consists of three nested tubes, contained in a Zircaloy pressure tube. The assemblies are approximately 25 feet long, with a 20-foot active section; the outermost tube is 3.3 inches OD.

A fabrication procedure for these tubes has not been developed, but they might be made by arc-melting and casting the uranium carbide cores, assembling the cast pieces in SAP sheath tubes, and attaching end plugs by brazing (5).

The reactor core consists of 604 fuel assemblies, each in a Zircaloy pressure tube, arranged in a 9-1/4-inch square-pitch pattern, as shown in Figure 20. The lattice contains a central flat zone 14 feet in diameter within which the control rods are located, and an annular buckled zone 22.1 feet in diameter, surrounding which is a radial heavy-water reflector zone 20 inches thick and an axial reflector 24 inches thick.

The design of the fuel and core is based on considerations similar to those described for the heavy-water-cooled reactor, but the heat flux is much less because of the relatively poor heat transfer properties of the organic coolant. Consequently, a larger number of longer tubes is required to deliver the specified heat output. The reactor inlet and outlet temperatures are based on a maximum surface temperature for the SAP cladding of 470°C.

2. REACTOR STRUCTURE

The reactor arrangement, shown in Figures 19 and 20, is similar to that described for the heavy-water-cooled reactor, with some dimensional differences. The calandria tank is about the same diameter because the larger number of fuel positions is offset by the smaller lattice pitch. The length is greater because of the increased length of the fuel. The materials of construction are stainless-steel shell and Zircaloy-2 tubes.

The axial shields are structurally similar to those previously described, and have comparable weight loads and heat loads.

The pressure-tube assemblies are shown in Figure 21. The material for the in-core part of the pressure tube is assumed to be Zircaloy-2 or some similar material having a satisfactory resistance to the organic coolant. (See Section III, B, 4, d.) The tubes are a smaller diameter than the pressure tubes for the heavy-water cooled reactor because the fuel assemblies are smaller in diameter. Based on the mechanical properties of Zircaloy-2, a wall thickness of only 0.062 inch is adequate for both the internal design pressure of 350 psig or against external collapse in case the gas space should become flooded. The Zircaloy tubes are joined to stainless steel end fittings by tandem extruded joints. The wall thickness at the transition joint will probably be greater than for the in-core part of the tube, and the ends of the stainless steel extensions are increased to 1/4-inch wall thickness to withstand the pipe reactions. The installation of the assemblies in the reactor is performed in the same way as described in Section II, A, 2, e.

The coolant headers are 36-inch-OD carbon steel pipe with 5/8-inch wall thickness. The coolant pipes from the headers to the pressure tube assemblies are 3-inch (3-1/2-inch OD) Schedule 40 carbon steel; the maximum flow velocities are 30 ft/sec. Although the flow rates for the organic coolant are somewhat less than for the heavy-water coolant, the same size is required for the coolant pipes to the pressure tubes to avoid excessive pressure drop. However, this pipe size permits a smaller lattice pitch in the organic-cooled reactor, because the pressure tube outside diameter is smaller and therefore the shield tubes are smaller in diameter. It is the clearance between the shield tube extensions required for the inlet and outlet coolant pipe lines that determines the minimum practical lattice pitch.

3. PRIMARY COOLING SYSTEM

a. Coolant

The primary coolant for the organic-cooled reactor is a commercially available mixture of terphenyl isomers having an ortho:meta:para weight ratio of approximately 1:5:2.8. This mixture is produced as a byproduct in diphenyl synthesis and is much less expensive (current price is about $0.17/lb) than the individual isomers which compose it. It is made by several firms, including Monsanto Chemical Company, who market it under the trade name of "Santowax-R". A great deal of information on the physical properties of this material has been developed and published by Atomics International Division of North American Aviation Company, and extensive information on its behavior as a reactor coolant has been obtained by Atomics International, Atomic Energy of Canada, Ltd., and others.

Pertinent physical properties of "Santowax-R" at the design conditions of this reactor are listed in Table 2. This material undergoes pyrolytic and radiolytic decomposition under reactor environmental conditions, producing gas and high molecular weight polymer products. The gas is 50 to 75 mol percent hydrogen, the balance being chiefly methane and longer-chain hydrocarbons in decreasing proportion. On a weight basis, almost all of the loss of terphenyl goes into the polymeric decomposition products, which are called the High Boiler Residue (HBR). Both the gas and the high boiler residue must be continuously removed from the coolant to maintain a stable condition. For design purposes, the equilibrium HBR content of the coolant stream is assumed as 30%; actually, it should be considerably lower than this in normal operation.

b. General Arrangement

The primary cooling system consists of six loops, each containing one steam generator and one circulating pump. All equipment and piping are carbon steel. Operating conditions are shown in the flow diagram, Figure 17.

Each loop circulates 46,700 gpm of "Santowax-R" at a reactor inlet header condition of 280°C and 300 psia, and removes 590 thermal megawatts (1.1×10^9 pcu/hr) of heat from the reactor. The system design pressure is 350 psig.

An arrangement of the loops in the Reactor Building is shown in Figures 23 and 24.

c. Pumps

The pumps are horizontal-shaft single-stage centrifugal pumps with bearings at both ends of the pump shaft and two mechanical seals. Each pump delivers 46,700 gpm at 280°C, at a total dynamic head of 687 feet. The pumps are direct-driven by 1200-rpm, 9000-hp motors.

The pumps require a minimum net positive suction head (NPSH) of 50 to 60 feet. This is provided by connecting the pump suction lines to atmospheric-pressure surge tanks 66 feet above the elevation of the pumps.

d. Steam Generators

The steam generator in each loop comprises a superheater, a boiler, and an "economizer", in separate shells. The organic coolant flows through the shell side of the superheater and economizer, with steam and water, respectively, in the tubes. In the boiler, the organic is on the tube side and water is boiled on the shell side. The organic coolant flows countercurrent to the water in the secondary system; that is, the hot organic liquid from the reactor flows first through the superheater, then through the boiler, and finally through the economizer before returning to the reactor. All three units are fabricated entirely of carbon steel.

The boiler is a vertical U-tube unit similar in design to the steam generators for the heavy-water-cooled reactor described in Section II, A, 3, c. The boiler section is shorter because one-fourth of the heat is transferred in the economizer and superheater. On the other hand, almost the same quantity of steam is generated, and at a lower pressure; consequently, the diameter of the upper disengaging and moisture separation section is larger. The units are 45 feet long over the heads, with an 18-foot-OD upper section and an 11-1/2-foot-OD lower section. Each unit contains 7770 tubes, 5/8 inch OD by 28 feet long, on a 7/8-inch triangular pitch. Total heat transfer surface area is 35,600 square feet per unit. The tube sheet is about 15 inches thick. The units will be designed, under the ASME Nuclear Code, for a pressure of 600 psig on the shell and tube sheets and externally on the tubes.

The superheater is a vertical straight-tube heat exchanger with fixed tube sheets, approximately 35 feet long over the heads by 5 feet 8 inches OD. Each unit contains 4010 tubes 5/8 inch OD by 31 feet long, to provide a heat transfer surface area of 20,400 square feet. The tubes are on a 1-inch triangular pitch.

The economizer is a horizontal straight-tube heat exchanger with fixed tube sheets, 16 feet 4 inches long by 5 feet 8 inches OD, containing 3910 tubes 5/8 inch OD by 12 feet long, on 1-inch triangular pitch, for a surface area of 7700 square feet per unit. Both the economizer and the superheater will be designed, under the ASME Nuclear Code, for pressures of 350 psig on the shell and 600 psig on the tubes, heads, and tube sheets.

e. Piping

The piping for the primary system is 36 inches OD with 5/8-inch wall thickness. Although the flow rate is somewhat less than it is in the heavy-water coolant loops, the low cost of the organic coolant as compared with heavy water makes it economical to increase the line size to reduce the system design pressure and the costs associated with pumping the coolant.

As before, two block valves are provided in each loop. All piping and equipment in the loops are steam-traced and provided with drains at all low points, because the organic coolant is solid at room temperature.

4. SECONDARY COOLING SYSTEM

The secondary cooling system is a closed loop, as shown on Figure 17. Superheated steam at 503 psig (358°C) is brought from the Reactor Building to the turbine-generator plant at a throttle pressure of 477 psig. At this pressure, two turbine-generator units are required for the steam flow of 13,150,000 lb/hr.

Each turbine generator is a single-shaft machine with a tandem-compound, four-flow, 43-inch last-stage blade length turbine driving a nominal 600-Mwe generator. The maximum capacity of the turbine is about 675 Mwe. The generator is cooled with 45-psig hydrogen and generates at 24,000 volts. The over-all length of each machine is 186 feet. The heaviest component during erection is the generator inner frame at 260 tons, and after erection is the generator rotor at 163 tons.

The remainder of the loop is conventional, as described for the heavy-water coolant plant. The condensers require 1,200,000 gpm of cooling water at 18°C.

5. REACTOR COMPONENTS HANDLING SYSTEM

The facilities for handling fuel and other reactor components are similar to those described in Section A, 5 for the heavy-water-cooled reactor. Some modifications in the new and spent fuel transfer areas will be required, and the Fuel Handling Machine will be longer because of the greater fuel length. This will introduce some problems of space for horizontalizing in the case of the refueling from below the reactor.

The available cycle time for removing and replacing one fuel assembly, on the basis of an average exposure of 16,000 megawatt days per metric ton, is about 25 hours, which is longer than for the heavy-water-cooled reactor.

6. CONTROL AND SAFETY ROD SYSTEM

The control and safety rod system is the same as that described in Section II, A, 6, with 37 control-rod clusters and 40 safety rods. The locations of the rods are shown in Figure 20.

7. REACTOR AUXILIARY FLUID SYSTEMS

The principal reactor auxiliary fluid systems are indicated in the flow diagrams, Figures 17 and 18.

a. Pressure-Volume Control System (Figure 17)

The pressure in the primary cooling system is mainly that developed by the circulating pumps. About 60 feet of pump suction pressure is required to avoid cavitation, and this is provided by an elevated pump suction tank which also serves as a volume surge tank. This tank is blanketed by an inert gas, such as helium or nitrogen, at atmospheric pressure. This tank serves to control the volume of coolant and limit the pressure fluctuations in the primary system in exactly the same way as the pressurizer for the heavy-water-cooled system described in Section II, A, 7, a.

Pressure relief valves on the primary and secondary cooling systems are provided, as described in Section II, A, 7, a.

b. Shutdown Cooling System (Figure 17)

The primary cooling system pumps are capable of operating under reactor shutdown conditions as well as reactor operating conditions, because the temperatures are maintained by steam tracing and the pressures are the same as during operation. It is probably necessary to maintain circulation in this system during shutdown to avoid freezing of the coolant in some spots. An auxiliary cooling system of low capacity is provided for emergency shutdown cooling, or in case it is necessary to drain the primary system for maintenance work.

c. Primary Coolant Purification System (Figure 17)

The principal function of the organic purification system is to remove the gaseous and polymeric decomposition products formed in the primary cooling system by heat and radiation. The system also serves to decontaminate the coolant, because most of the radioactivity will remain in the high boiler residue. The system is sized to handle a feed stream containing at least 30% HBR and for a flow rate adequate to maintain the normal HBR content of the coolant below 10%.

Degasification is carried out in a steam-heated vacuum tank into which the hot organic liquid is sprayed. Hydrogen, low molecular weight hydrocarbons, and any water that may be present in the organic liquid are separated in this step and taken through an off-gas system along with some terphenyl vapor. The condensable vapors are collected in condensers and freeze traps and the noncondensable gases are discarded through a stack.

Particulate materials are removed in clay-bed filters.

High boiler residues are removed by vacuum distillation. The rate of formation of HBR in the coolant is estimated to be 500 to 1000 lb/hr. The equipment is sized for 1100 lb/hr. The separated HBR is stored in decay tanks to lower the activity level and then burned.

d. Moderator, Control and Safety Rod, and Shield Cooling and Purification Systems (Figure 18)

These systems are the same as the corresponding systems for the heavy-water-cooled reactor described in Sections II, A, 7, d, e, f, g, and h.

e. Gas Systems

Gas systems are provided similar to those described in Sections II, A, 7, i, j, k, l, and m.

8. REACTOR CONTAINMENT

The reactor enclosure is a spherical containment shell 250 feet in diameter, made of 1-1/4-inch-thick A-201 Grade B steel plate. The shell is designed for an internal pressure of 25 psig at 230°C and an external live load of 60 psf as a Class B vessel, under the ASME Nuclear Code. The internal design pressure is based on a postulated incident in which all of the hot organic liquid in the system plus the contents of the secondary side of one steam generator are released to the building. It is further assumed that just sufficient additional water is available (from shield and moderator circulation system leaks, for example) to produce saturated steam by contact with the organic material, with no additional liquid water left for cooling. The pressure calculation assumes the fluids reach thermodynamic equilibrium with no loss of heat to the building structure or the environment, and no cooling by sprays or other means. The pressure does not include allowance for an organic fire; however, the building will withstand a fire without rupture, as discussed in Section III, I, 4.

The size of the containment shell is based on the arrangement of equipment within the building, particularly the primary coolant loops.

The philosophy regarding the facilities to be enclosed within the containment shell, as well as the additional containment protection by water sprays and halogen absorbers, is the same as that described for the heavy-water-cooled reactor containment building in Section II, A, 8.

9. REACTOR BUILDING ARRANGEMENT

The structural arrangement and the arrangement of major equipment in the Reactor Building is shown in Figures 23, 24, and 25. The arrangement is basically the same as that described in Section II, A, 9.

The heavy shield walls around the primary coolant loop equipment may not be necessary during normal operation because the radiation level from the organic coolant is much less than from heavy water; they are probably desirable, however, because of the possibility of high radiation levels from fission products released to the coolant by a fuel cladding rupture. They are also useful for structural support.

The building is provided with fire protection facilities appropriate to the use of the organic coolant, including high-pressure fog nozzles at suitable locations. This system is distinct from the spray system referred to in Section II, B, 8.

C. 8300-Mwt Heavy-Water-Cooled Reactor

This plant is quite similar but on a larger scale to the 3500-Mwt heavy-water-cooled reactor plant described in Section II, A. The principal differences are described below.

1. FUEL AND CORE

The fuel assemblies, shown in cross section in Figure 33, are identical with those for the 3500-Mwt reactor; however, the dimensions of the pressure tube and calandria tube are different because of the use of a higher-strength zirconium alloy for the pressure tube.

The reactor core contains 1258 fuel assemblies in Zr-2.5% Nb alloy pressure tubes, arranged on a 10-inch square pitch lattice. The flat zone, containing the control rods, is 20 feet in diameter. The entire lattice is 34-1/2 feet in diameter and 15 feet long, and is surrounded by a heavy-water reflector that is 20 inches thick radially and 24 inches axially.

The operating conditions and design limits specified for the fuel and core are the same as those described for the 3500-Mwt reactor, although, no doubt, these conditions will have been improved by the time a reactor of this size is designed in detail.

2. REACTOR STRUCTURE

The reactor structure consists of a calandria, pressure tube assemblies, inlet and outlet coolant headers and piping, and radial and axial shields, all arranged as described previously for the 3500-Mwt reactor.

The calandria is 38 feet in diameter and 21 feet long, constructed of stainless steel with Zircaloy-2 tubes, as described in Section II, A, 2, b.

The axial shields are 42 feet in diameter by 45 inches deep, constructed as described in Section II, A, 2, c. Weights and loads on the shields are as follows:

	Loads, tons	
	Upper Shield	Lower Shield
Dry weight	500	500
Live load	1200	1200
Total weight	1700	1700

The radial thermal shield tanks are as described in Section II, A, 2, d.

The pressure tube assemblies are 4.08 inches ID by 48 feet long and are fabricated in the same manner as described in Section II, A, 2, e. The materials of construction are zirconium-2.5% niobium alloy for the in-core section and A-376 type 347 alloy steel for the extensions. The Zr-2.5% Nb alloy is a precipitation-hardening alloy which is heat treated by solutionizing at 880°C, quenching, and tempering at 500°C. The ultimate tensile strength of this material is taken to be 78,000 psi, which permits a membrane stress of 26,000 psi, under the methods used in the development of the ASME Nuclear Code. The required wall thickness for the tubes is 0.17 inch for a design pressure of 2000 psig. The stainless steel extensions have an allowable stress of 18,400 psi at 315°C (the design temperature is slightly lower than for the Zr-2.5% Nb because of the absence of significant radiation absorption in the extensions) and therefore require a 0.24-inch wall thickness. The transition joint is made up to this thickness and reduced to 0.17 inch at the zirconium alloy end for welding to the pressure tube. In order to avoid making the shield tubes larger than necessary (which is important because their diameter affects the lattice spacing of the reactor), the annular clearance between the pressure tube extensions and the shield tubes is reduced to 0.055 inch (as compared with 1/8 inch in the 3500-Mwt reactor), which increases the heat load on the shields by loss of heat from the primary coolant.

The coolant headers are 30 inches ID with 2-3/8-inch wall thickness. The individual piping between the headers and the pressure tube assemblies is 3-1/2 inches OD, as before.

3. PRIMARY COOLING SYSTEM

The primary cooling system consists of ten loops, each containing one steam generator and one pump, circulating 83,000 gpm of heavy water and removing 841 thermal megawatts (1.6×10^9 pcu/hr). The operating conditions for the loops are shown in Figure 26. The system design pressure is 2000 psig.

Two alternate arrangements of the loops in the Reactor Building are shown in Figure 28; the left side is based on horizontal pumps and steam generators; the right side on vertical pumps and steam generators. The vertical arrangement is preferred because the required building diameter is smaller.

The pumps are vertical centrifugal pumps of the type described in Section II, A, 3, b for the 3500-Mwt system, designed to operate at 83,000 gpm at 550 feet of head. The pumps are 30" x 36" x 30" and measure about 8-1/2 feet by 9-1/2 feet over the casing, by 9 feet high. The pumps are driven by 1800-rpm, 15,000-hp motors. The motors are 5-1/2 feet high and weigh about 31 tons each.

The steam generators are vertical U-tube natural recirculation boilers with integral separators similar in design to those described in Section II, A, 3, c. The units are 62 feet long over the heads, with a 15-foot diameter boiler section and 20-foot diameter separating section, and weigh 500 tons each, dry.

The piping is 30 inches ID with 2-3/8-inch wall thickness. The maximum velocity in the piping is 42 ft/sec.

4. SECONDARY COOLING SYSTEM

The secondary cooling system is similar to the 3500-Mwt system described in Section II, A, 4. The operating conditions are shown in Figure 26. The plant requires three turbine-generators of the type and size (TC6F-44") described in Section II, A, 4 to handle a steam flow of 34,000,000 lb/hr. Gross electrical generation is 2490 Mwe. The turbine condensers require 2,360,000 gpm of cooling water at 18°C.

5. REACTOR COMPONENTS HANDLING SYSTEM

The system is functionally similar to the system described in Section II, A, 5. Two operating Fuel Handling Machines will replace a fuel assembly at the required average rate of one every five hours, based on an average exposure of 15,000 megawatt days per metric ton. The machines will shuttle between the reactor position and the transfer positions alternately, with one machine refueling while the other is loading fuel into or out of the transfer tubes.

6. CONTROL AND SAFETY ROD SYSTEM

The reactor has 89 control cluster positions distributed in lattice positions throughout the flat zone. In addition, 80 safety rods are provided at interstitial positions. The rods and drives are as described in Section II, A, 6.

7. REACTOR AUXILIARY FLUID SYSTEMS

The reactor requires the same auxiliary fluid systems as the 3500-Mwt reactor, described in Section II, A, 7. The principal systems are indicated in the flow diagram, Figure 26.

8. REACTOR CONTAINMENT

The reactor enclosure is a 350-foot-diameter spherical shell of 1-1/2-inch-thick A-201 Grade B steel plate, designed for an internal pressure of 19.4 psig at 215°C and an external live load of 60 psf, under the ASME Nuclear Code for Class B vessels. The internal design pressure is based on the same hypothetical incident as that described in Section II, A, 8; however, the shell thickness is based on the external loading rather than internal pressure.

The containment design bases are the same as those described for the 3500-Mwt reactor in Section II, A, 8, including the provision of a spray system and halogen absorbers, appropriately scaled up in capacity.

9. REACTOR BUILDING ARRANGEMENT

No study has been made of the reactor building arrangement for this size. It may be assumed that it will be similar in general to the arrangement described for the 3500-Mwt heavy-water-cooled reactor. As noted in the preceding section, the building is a 350-foot-diameter sphere; as shown in Figure 28, the diameter is based on the space requirements of the primary coolant loops.

D. 8300-Mwt Organic-Cooled Reactor

This plant is similar to the 3500-Mwt organic-cooled reactor plant described in Section B. Major differences are noted below.

1. FUEL AND CORE

The fuel assemblies are the same as for the 3500-Mwt case, as shown in cross section in Figure 22. The reactor core contains 1360 fuel assemblies on a 9-1/4-inch square-pitch pattern. The core is 20 feet long, with a flat-zone diameter of 24 feet and an over-all diameter of 33 feet. It is surrounded radially and axially by heavy-water reflectors which are 20 inches and 24 inches thick, respectively. The operating conditions and limits for the fuel and core are the same as those for the 3500-Mwt reactor, as described in Section II, B, 1.

2. REACTOR STRUCTURE

The reactor structure consists of the same components in the same arrangement as described for the three previous reactors. The calandria is 26 feet long by 36-1/2 feet in diameter. The shields are about the same size and weight as those for the 8300-Mwt heavy-water-cooled reactor, Section II, C, 2. The pressure tube assemblies are the same as those described for the 3500-Mwt organic-cooled reactor in Section II, B, 2 and shown in Figure 21, except that they are 52 feet long.

The carbon steel coolant headers are 42 inches OD with 5/8-inch wall thickness. The piping between the headers and the pressure tube assemblies is 3-inch, schedule 40 carbon steel, the same as for the 3500-Mwt organic-cooled reactor. The design pressure is 400 psig for the headers, piping, and pressure tube assemblies.

3. PRIMARY COOLING SYSTEM

The primary cooling system consists of ten loops, each containing one boiler, economizer, superheater, and pump which circulates 66,000 gpm of "Santowax-R" and removes 837 megawatts (1.6×10^9 pcu/hr) of heat. All equipment and piping is carbon steel. Operating conditions for the system are shown in Figure 27. The system design pressure is 400 psig.

The pumps are horizontal centrifugal pumps of the type described in Section II, B, 3, c, designed for operation at 66,000 gpm and 753 feet of head.

The steam-generator units are similar to those described in Section II, B, 3, d for the 3500-Mwt organic-cooled reactor. The sizes of the units are as follows.

	Superheater	Boiler	Economizer
Length, ft	35	50	16
Diameter, ft	7	15/22[a]	7
Heat Transfer Area, sq ft	29,000	51,800	11,000

(a) Boiler section/separator section.

The piping is 42 inches OD by 5/8-inch wall thickness, with a maximum coolant velocity of 17 ft/sec.

4. SECONDARY COOLING SYSTEM

The secondary cooling system operating conditions are shown in Figure 27. The temperatures and pressures are the same as for the 3500-Mwt organic-cooled plant. Total steam flow is 28,500,000 lb/hr. The plant requires five turbine-generators of the type and size (TC4F-43") described in Section II, B, 4 for the 3500-Mwt organic-cooled plant. Gross electrical generation is 2950 megawatts. The turbine condensers require 2,860,000 gpm of cooling water at 18°C.

5. REACTOR COMPONENTS HANDLING SYSTEM

This system is identical with the system for the 3500-Mwt organic-cooled reactor described in Section II, B, 5. The required rate of replacement of fuel assemblies based on an average exposure of 16,000 megawatt days per metric ton is one every 10 hours. It is judged that one operating machine will maintain this rate of operation.

6. CONTROL AND SAFETY ROD SYSTEMS

This system is the same as the systems for the other three reactors described. The flat zone is provided with 81 control clusters and 40 safety rods.

7. REACTOR AUXILIARY FLUID SYSTEMS

The auxiliary fluid systems are the same as those described in Section II, B, 7 for the 3500-Mwt organic-cooled reactor, suitably scaled up in capacity. The principal organic systems are shown in the flow diagram, Figure 27.

8. REACTOR CONTAINMENT

The reactor enclosure is a 350-foot-diameter spherical shell of 1-1/2-inch-thick A-201 Grade B steel plate, designed for an internal pressure of 15.5 psig at 203°C and an external live load of 60 psf, under the ASME Nuclear Code for Class B vessels. The internal design pressure is calculated in the same way as for the 3500-Mwt organic-cooled reactor in Section II, B, 8, but in the present case the shell thickness is determined by the external loading.

The same design bases are used as for the 3500-Mwt organic-cooled reactor, Section II, B, 8, including the provision of a spray system and halogen absorbers.

9. REACTOR BUILDING ARRANGEMENT

The arrangement of the facilities within the Reactor Building will probably be similar to that for the 3500-Mwt organic-cooled reactor; no studies were made of this. The building size is determined by the space required for the primary cooling loops, as it is for the other plants described.

III. DISCUSSION OF FEASIBILITY

The feasibility of large power reactors having the features described in Section II is discussed in this section, by consideration of the following types of questions:

1. What are the reasons for the particular design choices made, and what alternatives are available?

2. How essential are particular features to the technical feasibility, and how do they affect the costs?

3. What major problems of design and construction will be encountered?

4. What is the status of related technology for the solution of such problems?

5. What type of development work is required to solve such problems and what are the prospects of successful solution?

Since many of the design features and problems to be considered are common to all four reactors, it is convenient to discuss all four together rather than in separate sections.

A. Reactor Type and Arrangement

1. PRESSURE TUBE REACTORS

Heavy-water-moderated reactors are characterized by open lattices of relatively widely spaced massive fuel assemblies, with resultant large core sizes. This pattern, a direct consequence of the neutron-moderating properties of D_2O, contrasts with the compact close-spaced cores of light-water reactors, and suggests the use of a pressure tube reactor instead of the large pressure vessels used to enclose pressurized light-water reactors. Both pressure tube and pressure vessel reactors were considered in earlier studies (2, 3) and advantages were noted for both types; however, for the large reactors now under consideration, the pressure tube design is virtually mandatory. For the heavy-water-cooled reactors at 2000 psig design pressure, particularly, the core sizes are much larger that can be accommodated in a pressure vessel. The largest vessels now offered by manufacturers at this pressure are about 19 feet in diameter, and it is improbable that substantial increases will be made in the foreseeable future. While it is possible that sufficiently large vessels might be built for organic-cooled reactors at around 200 psig design pressure*, the design

*The design pressure required for a pressure vessel reactor is less than for a pressure tube reactor because of lower primary system frictional losses.

problems of segregating the hot organic coolant from the cold heavy-water moderator rule out the pressure vessel design from serious consideration in this case also.

Certain advantages and disadvantages inherent to the pressure tube reactor concept should be noted. Advantageous features are:

- The moderator is cold (around 80°C), which adds to the reactivity of the core, reduces the magnitude of reactivity changes during startups and shutdowns, reduces the hazard from "cold-water accidents", reduces the amount of internal energy the containment building must be designed to hold, and reduces some corrosion problems.

- The moderator is essentially unpressurized, which simplifies insertion of safety and control rods and the use of in-core flux monitors.

- The coolant is completely segregated from the moderator, which permits flexibility in the choice of coolant and its chemical treatment.

- The system lends itself well to monitoring each fuel assembly coolant stream for flow, temperature, and/or fission product activity, if desired.

The disadvantageous features are:

- A fraction (about 4 to 5% in the calandria design) of the fission energy is degraded to low-temperature heat in the moderator, thereby reducing the plant thermal efficiency.

- The large negative temperature coefficient of reactivity of the moderator is not coupled to the main heat removal system and therefore plays no important part in reactor stability.

These two features are not necessarily serious design drawbacks. The first relates to power cost and the second to plant safety; they must be judged in the context of the over-all plant economics and system stability.

2. VERTICAL ARRANGEMENT

A pressure tube reactor may be arranged with the tubes vertical or horizontal; the choice strongly influences the entire design of the reactor and the reactor building. There is no clear-cut basis for decision. AECL used a horizontal arrangement in the CANDU reactor and the NPD, which is probably due in part at least to their fuel-handling scheme, which involves on-power refueling of short elements from both ends of the reactor.

In the present case, refueling of long elements from one end only appears acceptable, and this can be accomplished about as well in a vertical as in a horizontal arrangement. We favor a vertical arrangement because of our experience with the design of the vertical heavy-water production reactors at Savannah River, which, although they differ in many ways from a power reactor, have many structural similarities, particularly in the shields. The vertical arrangement also is better adapted to the reactor control and flux-shaping system we prefer to employ, and is probably better suited to economical arrangement of the reactor building for the large reactors we are considering.

3. CALANDRIA

A basic option in a pressure tube reactor design is the use of a calandria, the tubes of which separate the pressure tube assemblies from the moderator, versus a "noncalandria" moderator tank in which the pressure tubes are immersed in the moderator. In the latter arrangement, the pressure tubes must either be U-tubes inserted from above the calandria, or, if straight-through-flow tubes are used, they must be sealed to the bottom of the tank. Also, the pressure tubes must be thermally insulated to hold heat losses to the moderator to a practical level; the insulation is preferable inside the pressure tube to take advantage of the higher strength of Zircaloy at lower temperatures. The insulated pressure tube design decreases the parasitic absorption of neutrons in the reactor, both by eliminating the calandria tubes and by decreasing the wall thickness of the pressure tube itself (partially offset by the absorption in the insulating material). The insulated pressure-tube design therefore has a strong advantage for a reactor designed as a breeder or for high conversion ratio.

Where electric power cost is the only criterion in design, the cost advantage of the insulated pressure tube must be weighed against the penalties resulting from higher heat losses and the complications of sealing between the pressure tubes and the moderator or adopting the U-tube arrangement. The penalties of the U-tube arrangement in lower fuel specific power are considered to rule it out. At the present time it is not possible to choose between the concept of insulated pressure tubes passing through the bottom of the moderator tank and the calandria concept on the basis of cost; therefore, we have selected the calandria concept for its relative simplicity.

4. ARRANGEMENT OF SHIELDS AND COOLANT CONNECTIONS

A number of arrangements for vertical pressure tube reactors were developed in earlier studies (2, 3), differing chiefly in the method of supplying and discharging coolant and the disposition of the axial shields. The coolant connections are discussed in Section III, B, 5. The arrangement of the axial shields simplifies the problems of supporting the reactor and bracing against the reactions from the inlet and outlet piping, and is judged to be the most suitable arrangement for a calandria-type reactor of any of those considered.

B. Reactor Structure

1. CALANDRIA

a. Materials

Although mild steel is an acceptable material of construction for the primary and moderator coolant systems (see Section III, C, 2), it is judged advisable to make the parts of the reactor structure that are inaccessible for replacement or repair out of corrosion-resistant materials. In the case of the calandria, the need to minimize parasitic absorption of neutrons by the calandria tubes dictates the use of aluminum or Zircaloy tubes. Zircaloy tubes are preferable in this respect because their thermal neutron absorption rate is about 65% of that for aluminum. A less expensive material, such as aluminum or stainless steel, may be used for the shell and end plates.

The materials of construction of the calandria must be considered in conjunction with those of the moderator coolant system, for compatibility with the moderator chemical composition. The principal choices are: (a) an all-aluminum calandria, with aluminum and/or stainless steel equipment and piping for the coolant system, and with the moderator maintained in a neutral or slightly acidic condition; or (b) a calandria with Zircaloy tubes and stainless steel for the remaining parts, a mild-steel coolant system, and the moderator maintained in an alkaline and reducing condition for protection of the mild steel. Either of these combinations is feasible. Considering only the cost of the calandria and cooling system, case (a) is the lower-cost of the two; but if the additional fuel costs resulting from the use of aluminum tubes are included in the comparison, this cost difference will be greatly reduced, and may be in the reverse direction. These costs have not been evaluated in detail.

Aside from cost, the Zircaloy tubes have an advantage over aluminum of greater strength (the tubes, whether aluminum or Zircaloy, require about the same wall thickness because the limiting consideration is external hydraulic pressure), which makes them less susceptible to damage from the causes discussed in Section III, B, 1, c. In the absence of any obvious cost disadvantage, therefore, we have selected the Zircaloy-tube case, (b).

b. Joining Tubes to End Plates

The principal problem in design and construction of a calandria with Zircaloy tubes and stainless steel shell and end plates is the means of joining the tubes to the end plates to secure a reliable leaktight vessel. Several methods are possible. The method used by AECL for the CANDU reactor calandria is to roll the tubes into the end plates, with stainless steel ferrules for strengthening the joints. This work has been completed satisfactorily for the Douglas Point calandria, although the reliability in service has not yet been demonstrated. An alternative method is to provide short stainless steel extensions on both ends of the Zircaloy tubes, by tandem extrusion, and to weld the stainless extensions to the end plates.

The stresses produced by the differential thermal expansion between the Zircaloy tubes and the stainless steel shell are not excessive for the metal temperatures involved.

c. Damage and Repair

The possibilities of damage to the calandria require careful consideration because of its inaccessibility for repair. The shell and end plates and the plenums may be made of heavy plate and strongly constructed so that the possibility of failure is remote in the extreme. The principal problem is with the calandria tubes, including their seals at the end plates.

A calandria tube may fail for a variety of reasons; for example: (a) localized forces and heating resulting from a bowed pressure tube contacting the calandria tube; (b) internal pressure and missiles resulting from failure of a pressure tube; (c) some combination of thermal stress, vibrational fatigue, corrosion, or fabrication defect.

The risk of a failure from some such cause may be reduced by various means, but not completely eliminated. The 1/8-inch gap width between pressure tube and calandria tube was chosen to minimize the first possibility; preliminary calculations indicate that a 10°C temperature difference between opposite sides of a pressure tube is required to produce a bowing of 0.1 inch. If necessary, ribs or other means of maintaining the gap may be provided.

The risk of rupturing a calandria tube by internal pressure may be minimized by providing pressure relief of adequate capacity from the gas space. The large annular gap is advantageous in this respect, and it may also prove desirable to increase the gas space between the calandria and the upper and lower axial shields above that shown on Figures 3 and 19, in order to facilitate venting of steam from the gas space. AECL has performed interesting experimental studies on the consequences to the calandria tubes of a pressure tube failure, using a nine-tube mockup of the NPD reactor calandria (45). Experiments of this type would be valuable with respect to the present reactors at the appropriate stage of design development.

The third possible cause of failure mentioned above also may be minimized by proper design and quality control in fabrication, but can never be completely eliminated. Testing of calandria tubes in a hot loop is desirable to prove the design.

Consequently, damage to the tubes must be anticipated and provision made for repair. This may require, for example, that the shield tubes for the heavy-water-cooled reactors (Figure 5) be made slightly larger than the calandria tubes, as they are for the organic-cooled reactors (Figure 21), and that the joints between the calandria tubes and end plates be designed for remote cutting and replacement through the shield tubes. The art of automatic welding inside tubes has advanced to the point where such a technique appears feasible. Experimental development to prove the applicability of the technique would of course be required.

2. AXIAL SHIELDS

The upper and lower axial shields are the heaviest components in the Reactor Building, except the steam generators. Their height is greater than required by their shielding function and is designed to keep the deformation of the shields under load and the consequent misalignment of the tube axes within acceptable limits. Their weight is minimized by the reactor arrangement employed, which divides the loads about equally between the two shields.

The principal problems in design and fabrication of the shields are to predict and control their deformation under loading and nonuniform heating, and to maintain the required position and alignment accuracy of the tubes after welding. The requirements on positioning and alignment of the tubes depend on the flexibility of the pressure tube assemblies. These problems have been analyzed and solved for shields of similar construction built for the Savannah River Plant. The Savannah River shields are somewhat smaller (about 18 feet diameter by 40 inches high), but the problems are essentially the same.

Figures 29 through 32 show the calculated deflections and edge slopes (which equal the slope of the angle by which the outermost tubes are deflected) for the shields under loading. These graphs are drawn specifically for the heavy-water-cooled reactors, but the results for the organic-cooled reactors are not very different.

Calculations of shield heating, most of which is due to heat transfer from the primary coolant, show that by proper design of the cooling water flow pattern within the shield it is practical to prevent excessive stresses and deformations due to unequal heating.

Thus we are sure that the shields can be designed and fabricated satisfactorily. Their weight and bulk will present transporation and erection problems, but these are not insurmountable. The shields (and calandria) must be installed before the Reactor Building is structurally complete, and they are not intended to be removable. Consequently, they are constructed entirely of stainless steel.

3. RADIAL SHIELDS

The radial shields do not present any unusual design or construction problems. They are similar to shields that have been built for Savannah River Plant.

4. PRESSURE TUBE ASSEMBLIES

a. General

The pressure tube assemblies are regarded as semipermanent components of the reactor, which means that they are to be replaceable without major disassembly of the reactor structure. The large number of assemblies involved, the lack of experience on their service life, and the possibility of their obsolescence within the lifetime of the plant are the principal reasons for this policy. Although replaceable, every effort must be made to ensure their integrity in service because the consequences of a failure include possible damage to the calandria (see Section III, B, 1, c) and loss of fuel coolant, which could result in even more severe damage to the plant. Both of these consequences are potentially more severe for the heavy-water-cooled reactors than for the organic-cooled reactors, because the higher pressure makes the risk of damaging the calandria greater, and the coolant is more easily replaced in the organic system since the surge tank on the pump suction is at atmospheric pressure. However, in the organic-cooled case, a failure would also involve a troublesome problem of removing solidified organic material from the gas space and the calandria.

The pressure tube assemblies are classified as Class A vessels under the ASME Nuclear Code (1963 Edition, paragraph N-131(a)). Although the materials used in the pressure tubes are not covered by the Code, the criteria and procedures prescribed may be applied to the pressure tube design to produce a quality consistent with the Code intent, provided that the mechanical properties of the materials are known and can be met in fabrication.

Specific details of the assemblies that are significant with respect to feasibility are discussed in the following sections. It will be seen that there are many particular aspects of the design that require developmental testing. Full-scale hydraulic and thermal testing of complete assemblies with simulated internal components is also required. These tests are of particular importance with respect to the problem of fretting corrosion mentioned in Section III, B, 4, b.

b. Zircaloy-2 Pressure Tubes - Heavy Water Coolant

Zirconium low alloys are the only known feasible materials for the in-core sections of the pressure tubes of the heavy-water-cooled reactors at the required temperature and pressure, because of the requirements for low neutron absorption. The most highly developed alloy for this purpose is Zircaloy-2 (ASTM B-353 Grade RA-1). For maximum strength, it is desired to use this material in the "Half-Hard" condition, which corresponds to approximately 25% reduction of area by cold working.

The fabrication of Zircaloy-2 tubes to the required dimensions, tolerances, and mechanical properties appears to be feasible with no significant development beyond the present state of the art. This opinion is confirmed by a major manufacturer of Zircaloy tubes. The following tabulation compares the present dimensional requirements with the dimensions of Zircaloy-2 tubes in use or being made for other reactors.

	Length	ID, in.	Wall Thickness, in.
Present Study	18'	4.08	0.27
PRTR	17'-5"	3.25	0.154
CANDU	17'-5"	3.25	0.155
NPD	~15'	3.25	0.163
CVTR	10'-0"*	3.53	0.253
HWCTR	11'-4-1/8"	4.625	0.250

*Straight length. Pairs of tubes are joined mechanically with Zr-2 U-sections at the bottom.

Extensive studies and experiments have been made (and are continuing) regarding the suitability of Zircaloy-2 in reactor service, including mechanical properties, corrosion behavior and the effects of irradiation (27, 28, 29, 30). The results of these investigations are generally favorable, but in certain areas additional test data are needed to provide an adequate degree of confidence. For the most part, such testing is required to prove the acceptability of a specific design rather than the basic feasibility of using this material.

One such area is the creep behavior under stress of Zircaloy-2 at reactor service temperatures, especially under reactor irradiation conditions. Although some tests indicate a pronounced increase in creep rate under such conditions, the results to date are conflicting.

Corrosion behavior is another area of continuing concern. Zircaloy-2 is attacked by high-temperature water through the formation of zirconium hydride, with resultant loss in impact strength and ductility. (See also Section III, B, 4, d.) Two alternative alloys similar to Zircaloy-2 have been investigated by AECL and others with a view toward obtaining improved corrosion resistance. These are "nickel-free" Zircaloy-2, which has the same nominal composition as Zircaloy-2 except that the nickel content is less than 70 ppm; and Zircaloy-4, which is the same as nickel-free Zircaloy-2 except for closer control of the iron content

(0.12 - 0.18%). Zircaloy-4 may prove to be superior to Zircaloy-2 as a pressure tube material because of its lower hydrogen absorption in alkaline water.

Zircaloy-2 is extremely susceptible to fretting corrosion in high temperature water. Conditions under which the protective oxide film on the tube can be continually worn away by rubbing must be avoided in design; and the adequacy of the design in this respect must be confirmed by thorough testing.*

Zircaloy-2 pressure tubes have been in reactor operation satisfactorily for two years in the NPD and for three years in the PRTR.* Zircaloy-4 tubes have not yet been used in reactors.

Design stresses for Zircaloy have been developed on the basis of our review of all available data and application of the criteria of the ASME Nuclear Code. Recommended design stresses (corresponding to the maximum allowable membrane stress, Sm, in the Nuclear Code) are shown in Figure 34. The design temperature of 320°C for the heavy-water-cooled reactors is the temperature of the coolant leaving the hottest fuel assembly, for an average reactor coolant outlet temperature of 304°C.

c. Zirconium-Niobium Alloy Tubes - Heavy-Water Coolant

An alloy of nominal composition 97.5% Zr-2.5% Nb is under extensive development by AECL and others as a potential pressure tube material. It has a tensile strength at 300°C of 80,000 - 90,000 psi as compared with about 52,000 psi for Zircaloy-2 (31). This material is a precipitation-hardening alloy requiring heat treatment after extrusion and drawing of the tubes. The use of this alloy would permit reducing the wall thickness of the pressure tube to about 0.17 inch, as compared with 0.27 inch for Zircaloy-2.

At the present stage of development, much needs to be done to determine the suitability of this material under reactor service conditions. Fabrication of tubes is also still in the development stage. There is no operating experience with such tubes.

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*Reference (51), received after completion of this report, describes results of measurements on PRTR pressure tubes over a period of 2-1/2 years in service. Results confirm the acceptable performance of the tubes to date and emphasize the importance of fretting corrosion.

AECL regards this material as sufficiently promising that they have indicated their intention to use it in their CANDU-type reactor for future plants (subsequent to the Douglas Point plant), and we have no sufficient basis for exercising an independent judgment as to its feasibility. However, it has not yet reached the level of proved technology which characterizes the other components of the 3500-Mwt reactor; consequently, we have not indicated its use in this reactor. We have indicated its use in the 8300-Mwt reactor, which of course is several years in the future.

d. Pressure Tubes for Organic Coolant

In contact with organic coolant, zirconium alloys are subject to the same type of hydrogen corrosion as in water. Since the effects are more severe with organic coolants, the phenomenon is described in some detail here to bring out the factors involved. In high-temperature organic or water, zirconium alloys suffer general surface corrosion at a very low rate, measured in fractions of a mil per year. This type of corrosion is not a problem, but the reactions release hydrogen (or deuterium) at the metal surface and some of the hydrogen dissolves in the metal. While in solution, hydrogen does not affect the mechanical properties of zirconium alloys. But when the solubility limit is exceeded, precipitation of zirconium hydride occurs, which has the effect of drastically decreasing the impact strength of the metal, with some decrease also in ductility. Tensile strength is not seriously affected, but the material is susceptible to brittle fracture.

The solubility limit of hydrogen in zirconium alloys increases with temperature; typical values are as follows:

	Hydrogen Solubility Limit, ppm	
	300°C	400°C
Zircaloy-2	95	270
Zr-2.5% Nb	175	550

The rate of absorption of hydrogen, too, increases with temperature, and is dependent on other factors such as impurities in the liquid. In particular, chlorine is believed to increase the rate. The effect of radiation exposure on the hydrogen absorption rate is uncertain. Various in-reactor experiments to date indicate rates ranging from 1 to 10 times the out-of-reactor rates.

It is important to note that the absorption rate is proportional to metal surface area exposed to coolant, so that, for a tube, the rate of increase in the concentration of dissolved hydrogen in the metal is inversely proportional to wall thickness. Typical absorption rates expressed in terms of hydrogen concentrations, obtained from Reference (35), are as follows:

	Hydrogen Absorption Rate, ppm/(yr)(100 mils)	
	300°C	400°C
Zircaloy-2	5	90
Zr-2.5% Nb	2 - 4	65 - 130

These rates are for water at 300°C and steam at 400°C; the rates for organic liquid are believed to be not markedly different. These data are based on out-of-reactor tests.

From these rates and the solubility limits we can calculate the time required for a given thickness of pressure tube to become saturated with hydrogen; for example:

0.27-in. Zr-2 tube at 300°C	-	50 years
0.17-in. Zr-2.5% Nb tube at 300°C	-	75-150 years
0.062-in. Zr-2 tube at 400°C	-	1.9 years

These examples are given merely to illustrate the problem; they show that hydrogen embrittlement in organic-cooled pressure tubes is a potentially greater problem than in water-cooled tubes because of both the higher temperature and the thinner wall. They should not be taken literally as describing the conditions in the organic-cooled tubes, regarding which little information is available.

In considering the design of pressure tubes for organic- (and water-) cooled reactors to determine their suitability with respect to hydrogen embrittlement, we must do more than compare their service life with the time required to reach saturation. In the first place, when unsaturated tubes are cooled down, hydride precipitation may occur; this is reversible, that is, the hydrogen will redissolve when the tubes are reheated. On the other hand, exceeding the solubility limit does not necessarily mean tube failure; the material becomes progressively embrittled, and consideration must be given to the conditions of pressure, impact loading, etc. that may occur while it is in this state.

The use of zirconium alloys for organic-cooled pressure tubes is under active study by AECL. In addition to studies on the known alloys Zr-2, Zr-4, and Zr-2.5% Nb, AECL suggests that other alloys may possibly be found having lower hydrogen pickup rates. Another approach suggested by AECL is to clad the tube with a thin hydrogen diffusion barrier film, such as aluminum.

The feasibility of hot organic pressure tubes is therefore very much in question at the present time. If no satisfactory solution is found, an alternate approach is to adopt an internally insulated pressure tube design, with a noncalandria moderator tank. This approach has the advantage that the pressure tube is both cooler and is protected from the organic liquid. Insulated pressure tubes of various types are under study by AECL (36). This type of design has not been considered in the present study.

For the present, the wall thicknesses for the pressure tubes have been based on the properties of Zircaloy-2 without consideration of hydrogen embrittlement or creep.

e. Transition Joint

Several methods have been considered for joining the zirconium alloy pressure tubes to the stainless steel extensions. These include a rolled joint, a friction-welded butt joint, and the tandem-extrusion joint. Mechanical joints are ruled out by space limitations.

A rolled-joint design was used by AECL for the Douglas Point reactor. It is not desirable in our reactor arrangement because of the heavy wall required for the stainless steel section into which the Zircaloy tube is rolled. Since this section fits inside the shield tube, the latter would have to be increased in diameter with a corresponding increase in the reactor lattice pitch. Also, one end of the assembly must pass through the calandria tube, which would also have to increase in diameter, making the width of the annular gas space intolerably great.

Friction-welded butt joints and tandem-extrusion joints have the advantage that they may be made to the same inside and outside diameters as the connecting tubing. Also, they are metallurgically-bonded joints, so that they have a presumed advantage for remaining leaktight. Efforts to develop a friction-welded butt joint have so far not been successful with respect to attaining the full strength of the tube, whereas the development of the tandem-extruded joints is proceeding satisfactorily.

The tandem-extruded joint was developed by Nuclear Metals, Inc. (32). The joint is made by coextruding a jacketed composite billet which is stainless steel on the forward end and Zircaloy on the following end. The two metals are fitted together on a taper, with the Zircaloy as the male member. The joint can be cold worked by tube reducing to at least 40% reduction in area.

As-extruded and 20% cold-worked joints have been tested to failure at Savannah River by pressurization and axial loading; in all cases failure occurred in the adjacent Zircaloy tubing rather than in the interface section. Joints were also hydrostatically tested to hoop stresses of more than twice normal design stresses, and subjected to temperature and pressure cycling up to 116 cycles. The joints were examined for dimensional stability, dye-penetration, helium leakage, and metallographic condition; no significant defects were found. These tests were made on seven specimens, mostly about 1-1/2-inches ID. All joints were Zircaloy-2 to type 347 stainless steel, except one joint which was type 304L stainless steel (33).

Six joints have also been tested by irradiation to neutron exposures corresponding to 20 years of service, while under axial stress, with no measurable loss in strength or serious bond deterioration (34).

Preliminary results of stress analysis of the tandem-extruded joint design indicate some problems in meeting certain design criteria of the ASME Nuclear Code with respect to thermal stresses, but these problems may be overcome by fatigue testing of the joints as permitted by the Code under Paragraph I-1080.

The four reactors under consideration have three different designs of tandem-extruded joints, as follows:

3500-Mwt D_2O-Cooled:	Zircaloy-2 to type 316 Steel, 4.08" ID x 4.62" OD
8300-Mwt D_2O-Cooled:	Zr-2.5% Nb to type 347 Steel, 4.08" ID x 4.42" OD
3500- or 8300-Mwt Organic:	Zr-2 (?) to type 347 Steel, 3.55" ID x 3.67" OD

Any of these will require testing under reactor conditions of temperature and irradiation to establish its suitability.

f. Pressure Tube Extensions and End Fittings

Because of the considerable length required for the pressure tube assemblies to pass through the axial shields and primary coolant pipe banks, they are made from a less expensive material than zirconium. Because of the requirement for tight sealing on the end closures that must be removed for refueling, they are made from a corrosion-resistant alloy. Type 316 is preferred for its good fabrication properties, but type 347 is used when the higher strength is needed to more closely approach that of the zirconium-niobium alloy. Consideration is also being given to Inconel because of problems experienced with chloride stress cracking of austenitic stainless steels.

There are a number of problems with respect to these parts of the pressure tube assemblies that must be worked out by mockups and testing. These include:

- The design of the end closures to achieve tight, reliable sealing when they are removed and replaced repeatedly;

- Methods for making, cutting, and testing the welds of the end fittings to the piping, the internal tube weld of the end fitting to the pressure tube extension, and the seal weld for the gas space.

We expect that satisfactory solutions to these problems can be found through an appropriate development program.

5. COOLANT HEADERS AND PIPING

The means of supplying coolant to and discharging it from the reactor pressure tube assemblies is one of the major design problems of the reactor complex. The principal problems arise from the large number of closely spaced tubes to be served and from the differential axial and radial movement of the hot pressurized system relative to the cold calandria and shield system. The pressure tube assemblies lengthen by amounts ranging from 1-3/4 inches for the 3500-Mwt heavy-water-cooled reactor to 3 inches for the 8300-Mwt organic-cooled reactor. The maximum radial differential movement that must be accommodated is equivalent to the thermal expansion occurring over the distance from the central tube to the outermost tube. This movement ranges from 0.7 inch for the 3500-Mwt heavy-water-cooled reactor to 1.3 inches for the 8300-Mwt organic-cooled reactor.

Methods of supplying and discharging coolant suggested from previous studies (2, 3) included various arrangements of plenums, cross-headers, and individual piping from a main header. In any of the schemes using plenums or cross-headers, the outer pressure tubes would have to be allowed to move radially, by the distances given above, with respect to the calandria and shield tubes, requiring an excessively large gas gap between the calandria tubes and pressure tubes and between the shield tubes and pressure tubes. In addition, the gas seals around the pressure tubes would have to accommodate this amount of radial expansion plus the axial expansion. Because of these and related design problems, the use of individual flexible supply and discharge piping is the most feasible scheme for a large reactor.

The 3-inch pipe size selected for the runs between the headers and the pressure tube assemblies is the most suitable size, considering the factors of pressure drop, heavy-water inventory, pipe flexibility, and space available between the shield tube extensions for pipe runs.

In order to avoid the problems of radial motion of the pressure tube assemblies relative to the calandria, it is necessary to anchor the ends of the 3-inch pipes where they join the pressure tube assemblies. This is accomplished by having the pressure tube end fittings fitted snugly in the shield tube extensions, so that the pipe reactions are transferred to the very heavy rigid shield structure, which is kept cold and subject to only slight thermal expansions.

Essentially all of the differential thermal expansion of the piping and header must now be accommodated by the flexibility of the piping between the pressure tube and the header, and the piping must be arranged to provide sufficient flexibility to keep stresses and reactions on the pressure tube assemblies to acceptable levels. The use of carbon steel rather than stainless steel pipe simplifies this problem because of its lesser thermal expansion.

The flexibility problems of the supply piping differ somewhat from those of the discharge piping because the upper ends of the pressure tube assemblies are anchored to the upper axial shield, while the lower ends grow downward by the amounts mentioned above. Also, it is desirable to make the axis diameter of the lower header as small as practicable to minimize the overhang of the concrete shield around the reactor. For these reasons it is necessary to mount the lower header on spring supports so that it can move up and down in response to temperature changes.

The piping must cross the face of the shields in banks of up to 13 pipes high for the 3500-Mwt reactors, and 20 pipes high for the 8300-Mwt reactors, running in lanes between the shield tubes, in order to keep the ends of the pressure tube assemblies accessible for installation and replacement and for refueling. This requirement determines the lattice pitch of the reactor. In the 3500-Mwt heavy-water-cooled reactor, for example, the minimum practical outside diameter of the shield tube extensions is 5.87 inches, and the minimum practical outside diameter of the coolant pipes, from the standpoint of pressure drop, is 3.5 inches. On a 10-inch lattice pitch, this leaves only an average clearance of about 0.3 inch to accommodate pipe movement under thermal expansion. In the 8300-Mwt heavy-water-cooled reactor, the corresponding clearance is 0.4 inch. In the organic-cooled reactor, the fuel assemblies are smaller in diameter, so that the shield tubes may also be smaller and a closer lattice spacing is possible. Some slight increase in the lattice pitch (not exceeding 1 inch) is likely to be required when these and related design problems are considered in greater detail. This amount of increase is estimated not to have a significant effect on energy cost.

C. Primary Cooling System

1. COOLANTS

Heavy water and organic coolants were selected for this study by direction of the AEC on the basis of economic studies which showed these two to be the most favorable of all the coolants considered for heavy-water-moderated reactors (5). The technology of heavy water as a reactor coolant is well established and requires no discussion here; the organic coolants are newer and are less well known, so some information pertinent to their feasibility is appropriaate.

"Santowax-R" or similar mixed terphenyl isomers judged to be the most suitable organic coolants known, and most of the experimental work on organic coolants has been devoted to these materials. [See References (36, 37, 38, 39).] Santowax is readily available at low cost, has a low vapor pressure at high temperature, and is reasonably resistant to pyrolytic and radiolytic decomposition. As a reactor coolant, it is inferior to heavy water, having lower heat transfer rates and greater pressure drops (at the same conditions of temperature and flow rate), mostly because of its higher viscosity.

The commercial availability of the material is of importance because of the large quantities that would be required if organic-cooled power reactors become a reality on a substantial scale. Although it is a byproduct of diphenyl production, there is no apparent reason why it cannot be made in virtually unlimited quantities from benzene at substantially the same cost.

Fouling of heat transfer surfaces as a result of decomposition of the organic material and because of its tendency to pick up iron and other elements in the cooler parts of the loop and deposit them on fuel surfaces has been a major problem. AECL found that chlorine promotes this action. Fortunately, the vendor (Monsanto) is able to supply low-chlorine Santowax, which has greatly reduced the fouling rate. In the judgment of AECL, fouling can now be kept under control, principally by quality control on the raw Santowax and by adequate continuous purification to maintain low HBR concentration in the coolant.

Any water in the organic coolant has a strong corrosive action on steel. However, the degasifier in the purification system will remove any water.

Because Santowax is solid at room temperature, careful attention will be required in design to provide adequate heat tracing and drains, and to avoid cool spots and stagnant areas where crystallization could occur.

2. MATERIALS OF CONSTRUCTION

The use of mild steel as the principal material of construction in a heavy-water coolant system has been tested in the HWCTR with excellent results. The heavy water is maintained in an alkaline condition (pD 10.2 to 11.2) with lithium hydroxide, and oxygen is suppressed by addition of deuterium or hydrazine. The water is also purified, at a turnover rate of about 10 hours, with ion exchangers and filters. Over a two-year period of operation the concentration of iron in the water and the turbidity have both been negligible.

In the HWCTR, most of the piping, and the steam generators, including the tubes, are carbon steel. Heavy-water velocities in the pipe are about 22 ft/sec and in the steam generator tubes are about 12 ft/sec. Some problems were encountered during construction because of rusting and pitting of pipe and tubes during storage prior to installation. However, the surfaces were chemically cleaned and protected with a film of water-displacable oil, which was removed by flushing with light water after the system was completed. When the system was heated, a black adhering magnetic film was formed on the steel surfaces and protects it. There is no evidence, from visual examination of steam generator tubes, that any further corrosion has taken place since the system was put in service. Two failures of steam generator tubes have occurred, but the cause is not believed to be due to corrosion.

On the basis of experience from the HWCTR, we conclude that, with proper water chemistry control, and proper precautions during construction, the use of carbon steel for heavy-wall pipe and equipment is entirely satisfactory.

For steam-generator tubes, there is not sufficient experience to advocate carbon-steel tubes unreservedly in heavy-water service. Consideration should be given to corrosion-resistant alloys, depending on the cost differential, accessibility for repair, and costs incurred by shutdowns for repair. Austenitic stainless steels are not highly regarded for this purpose, because of the troubles from cracking due to chloride stress corrosion that have been experienced at Savannah River Plant, and elsewhere. Inconel is considered to be the best choice, aside from cost.

For the organic-cooled systems, carbon steel is completely satisfactory insofar as system integrity is concerned.

3. OPERATING AND DESIGN CONDITIONS

Reactor inlet and outlet temperatures, and the corresponding primary coolant flow rates, were determined with the SRL computer program on the basis of the fuel and core design considerations and associated costs as described in Sections II, A, 1 and II, B, 1. The maximum temperature T_m of coolant leaving a fuel assembly was obtained by adding $0.4\Delta T$ to the reactor average outlet temperature, where ΔT is the difference between the reactor outlet and inlet temperatures. The operating pressure at the reactor outlet was taken to be the saturation pressure corresponding to T_m. The corresponding operating pressure at the reactor inlet was found by adding the pressure drop through the reactor. The system design pressure was then taken as 15% higher than the reactor inlet pressure, to allow for normal pressure fluctuations, margins for relief valve settings, etc.

The temperature on the secondary side of the steam generators was obtained by an economic balance of the associated costs, including the costs of the steam generators and of the turbine generators. This optimization was also performed with the SRL computer program on the basis of the data then available. In the case of the heavy-water-cooled reactors, the resultant conditions gave a very close approach of the primary coolant temperature to the boiling temperature in the steam generators. This, of course, affects markedly the amount of steam generator surface area required. Reviewing the results, it is evident that the temperature on the secondary side of the steam generators should be reduced to the extent possible without requiring the addition of another turbine, in order to economize on the steam generators.

4. PUMPS

a. Size Limits

Major American manufacturers of centrifugal mechanical-seal pumps for nuclear service are confident that pumps of the type described in Section II are entirely feasible up to at least 90,000 gpm for vertical pumps and up to about 60,000 gpm for horizontal pumps. The horizontal pumps are larger, heavier, and more expensive than vertical pumps of the same capacity. Vertical pumps up to 200,000 gpm and horizontal pumps up to 87,000 gpm have also been suggested by manufacturers in some cases, but these represent a much greater extrapolation of known design and a greater capacity than we consider necessary or wise to consider for this study.

The pumps proposed in this study, on the other hand, represent reasonable advances beyond present designs. For example, the 83,000 gpm vertical pumps for the 8300-Mwt heavy-water-cooled reactor are compared below with the vertical mechanical-seal pumps now being built for the Pacific Gas and Electric Company's Bodega Bay nuclear power plant in California, which are the largest pumps of this type built to date:

	Bodega Bay Plant	8300-Mwt D_2O-Cooled Plant
Size (discharge x suction x impeller diameter, in.)	26 x 28 x 26	30 x 36 x 30
Capacity, gpm	29,000	83,000
Total dynamic head, feet	100	550
Suction pressure, psig	1,089	1,570
Suction temperature, °C	282	266
Speed, rpm	880	1,800
Case weight, lb	18,000	35,000

This tabulation shows that although the capacity and head of the largest proposed pumps are much greater than for the pumps now being built, the physical size is only slightly greater; most of the increase in capacity and head is obtained by increasing the speed.

b. Net Positive Suction Head (NPSH)

The net positive suction head required to prevent cavitation is normally quite high for these large pumps, 120 feet to 200 feet in most cases. This is no particular problem with the heavy-water pumps during hot operation, because the overpressure that must be applied to prevent boiling in the hottest fuel channels ensures more than adequate NPSH at the pump suction. This is true even if the pump is placed upstream of the steam generators; the available NPSH is about 300 feet in this case, and about 800 feet on the downstream side of the generators. However, it does prevent the pumps from operating at full speed when the reactor is depressurized, regardless of where the pumps are located in the loop relative to the steam generators and regardless of where the pumps are physically located in the building (within reasonable limits).

The NPSH situation is of greater importance, probably, for the organic pumps, because in the organic systems the NPSH requirement adds directly to the system operating and design pressure. To minimize system design pressure it is desirable to locate the pumps at the lowest physical point and highest overpressure point in the loop and to design the pumps for as low an NPSH as is practical. The manufacturers have advised that it is feasible to design for an NPSH of 50 to 60 feet, and perhaps as low as 45 feet. However, the size and cost of the pumps increase, because the reduction in NPSH requirement is achieved by lowering the speed and increasing the impeller size. In the example of the 83,000 gpm pumps cited above, the impeller size would increase from 30 inches to 40 inches, or more.

c. Fabrication Problems

Fabrication problems for these large pumps include the casing castings and the shafts. From a corrosion standpoint, carbon steel is satisfactory for the casing, as noted in Section III, C, 2; however, for maximum strength with minimum weight, and for better casting quality, a low-alloy steel is recommended. The shafts are made of a high-alloy high-strength steel.

d. Shaft Seals

The shaft seal is an important design problem for the heavy-water pumps, but the outlook for a successful design is very encouraging. Considerable progress has been made in design of high pressure rotating face seals by pump manufacturers in recent years. Experience with such seals on heavy-water pumps at Savannah River Plant has been very satisfactory both for the relatively low pressure, high capacity pumps for the production reactors and the high pressure, relatively small (1500 psig, 5000 gpm) HWCTR pumps.

The 4.64-inch-diameter shaft seals on the HWCTR pumps are of a special design developed jointly by the manufacturer and Du Pont to minimize the effect of pressure on deformation. A larger version of this seal, having a midface diameter of 10-1/4 inches, has been built by the same manufacturer and is currently being tested by him at 1700 psig and 1200 rpm. This is believed to be the largest high-pressure, face-type seal built to date.

Shaft seals for the organic pumps are of conventional stuffing-box design because the pump suction pressure is low. It is necessary to steam jacket or otherwise heat the stuffing box, and some precautions are necessary to prevent water from getting into the organic coolant from this source.

5. STEAM GENERATORS

a. Types and Sizes

Manufacturers of steam generating equipment in the United States have capabilities for building generators with capacities to 900 thermal megawatts, and generating up to 3-1/2 million pounds of steam per hour. Capabilities of course vary among the manufacturers, depending on their specific designs and shop facilities. Only those shops which are able to ship by water would be able to handle the size mentioned above; fabricators who must ship by rail would be limited to about two-thirds of this capacity, or less, for some designs.

Four types of units suggested by the vendors have been considered. Three of these are natural-circulation boilers; the fourth is a once-through forced-circulation boiler. The natural-circulation units include: (1) a horizontal straight-tube boiler with moisture-separating equipment in the upper part of the boiler drum; (2) a horizontal straight-tube boiler with a separate steam drum above the boiler drum and connected to it by riser and downcomer pipes; (3) a vertical

U-tube boiler with moisture-separating equipment in an enlarged upper section of the same shell.

The last-mentioned type was selected for this study, primarily because it permits a better arrangement of equipment in the reactor building than any of the others, with respect to minimizing the size of the building and the length of the primary coolant pipe lines. This vertical U-tube type of generator is made by several manufacturers, and many have been made for nuclear power plants.

For this study we sized three units of this type for the operating conditions of the heavy-water-cooled reactors. These are listed in the following table; corresponding information is given for the largest units of this type built to date - three units now under construction for the Southern California Edison Company's Camp Pendleton plant.

	Units Considered in Current Study			Units for So. Calif. Edison Co.
Capacity, Mwt	450	600	900	~500
Steam rate, 10^6 lb/hr	1.8	2.4	3.6	2
Design press., psig - Pri.	2,000	2,000	2,000	2,500
- Sec.	750	750	750	675
Surface area, sq. ft.	53,000	70,000	106,000	28,000
Tube size, in.	1/2	1/2	1/2	3/4
Tube pitch, in.	3/4	3/4	3/4	1-1/16
No. of tubes	9,000	12,000	18,000	3,800
Height, ft.	53	56	62	45
I.D., ft. Boiler	10.2	11.6	14	10
Separator	13.5	15.5	19	12
Weight, tons	250	330	500	215
Tube material	Carbon steel or Inconel		-	Inconel

b. Fabrication Problems

The principal problems in fabrication which limit the capabilities of the various manufacturers are as follows:

Size: Shells larger than 12 to 14 feet in diameter cannot be handled by rail and would have to be shipped by water. This would apply to all three units considered in this study except perhaps the smallest. The sizing of the separator section is somewhat uncertain.

Weight: Most shops are limited to a handling capacity of 250 tons, or less. One manufacturer currently has a capacity of at least 500 tons, and a second is enlarging his facilities to have this capacity. Water shipment is probably required for weights over 250 tons, and the largest shops referred to above are located and equipped for barge loading.

Tube Sheet Thickness: One of the advantages of the U-tube design of boiler is that it eliminates problems of differential thermal expansion between the tubes and the shell. The penalty for this, however, is that the tube-sheet thickness is some 70% greater than the tube sheets in a straight-tube boiler. By TEMA standards, the thicknesses required for the 450, 600, and 900-Mwt units tabulated above are about 25, 29, and 35 inches, respectively. These are only approximations, since the units will actually be designed under the ASME Nuclear Code.

The tube sheets for the units being built for Southern California Edison are 24 inches thick, which is probably the maximum that has ever been built. The fabrication problems with tube sheets of this thickness are the problem of making the forgings of satisfactory quality and the problem of drilling the tube holes accurately enough to meet the requirements for minimum ligament between holes. Tolerances of less than 10 mils per foot on runout must be maintained. With the small size holes required for the units considered in this study, meeting these tolerances is even more difficult. Nevertheless, fabricators have expressed the opinion that 1/2-inch drilling in tube sheets up to 27 or 28 inches thick is probably feasible.

We judge that the 450 and 600-Mwt units are feasible in this regard, and very likely the 900-Mwt unit too, making allowance for the possibilities of reducing the thickness by use of a high strength alloy and careful stress analysis, and some advances in the art over a period of time.

Tube Size: The economical tube size for steam generators with heavy water in the tubes is smaller than is customary for light-water-heated units, because of the high cost of the heavy water. For example, in 1/2-inch 18-gage tubes, the value of the heavy water is $9 per square foot of heat transfer area, whereas in 3/4-inch tubes it is $16 per square foot. An additional penalty for the larger tubes is incurred because of the heavy water held in the channel. For an ellipsoidal head, this can amount to as much as $3/sq ft differential. A total penalty of about $10/sq ft for heavy water costs is therefore incurred by the use of 3/4-inch tubes, as compared with 1/2-inch tubes. Therefore there is a considerable justification for paying for the extra fabrication costs of 1/2-inch tubes. Manufacturers generally consider that 1/2-inch tubes are the smallest practical size for fabrication (because of the tube drilling problem mentioned above, welding problems, and others); and not all manufacturers are willing to make generators with this small a tube.

c. Design Problems

The following design considerations are pertinent to these generators.

<u>Natural Circulation Rate</u>: The recirculation rate in the boiler must be sufficiently high to keep all of the tubes well wetted in order for the unit to function properly and achieve its design heat transfer rate. A recirculation rate of at least 5 to 10 pounds of water per pound of steam generated is normally desirable. Since units as large as those under consideration have not been operated, we have no basis from experience to determine their performance. Neither are there any reliable procedures for estimating the performance by extrapolation from smaller units. However, it is evident in a qualitative way that the larger the units (in terms of heat transfer surface) and the more closely spaced the tubes, the lower will be the recirculation rate, other things being equal. Consequently, there is reason for concern regarding the units under consideration.

By calculation of the frictional resistance of the tube bundle, the recirculation rate is found to be adequate, but such calculations are not very reliable as applied to this situation. It appears that it will be necessary to make a hydraulic mockup of the boiler to obtain reliable information for design purposes. If the circulation rate turns out to be inadequate, it can be remedied in the design by increasing the tube pitch. This of course aggravates some fabrication problems mentioned above.

<u>Velocity in Tubes</u>: The generators should be designed for the maximum practical velocity of primary coolant in the tubes, in order to minimize the diameter of the boiler. In general, velocities may be limited by pressure drop or by erosion-corrosion. In the present case, the pressure drops are very low and not limiting. With respect to erosion-corrosion, velocities of 10 to 15 ft/sec are frequently recommended for carbon steel tubes, with higher velocities permissible for alloys such as Inconel. The generators tabulated above for this study are all based on a velocity of 14 ft/sec. The use of Inconel tubes and a higher velocity is probably justifiable.

Economizer: An economizer is a heat exchanger in which boiler feed water is heated to or near the boiler temperature by countercurrent heat exchange with the primary coolant leaving the boiler. When the boiler is of the U-tube design, the economizer is necessarily a separate shell. We have not used an economizer for the heavy-water-cooled cases because of the additional piping and heavy-water holdup. However, an economizer effects a savings in the total heat transfer surface area required; and, for the particular operating conditions considered in this study for the heavy-water-cooled reactors, this saving is abnormally great because of the close approach of the heavy-water temperature to the boiler-water temperature. For any of the three units being considered, the addition of an external economizer would decrease the total surface area required (boiler plus economizer) by 17% and would decrease the surface area required in the boiler itself by 27%. If the operating temperatures are adjusted to provide a greater approach, as suggested in Section III, C, 3, the savings an economizer will effect will be less; nevertheless, the question of its economic justification requires design consideration. A drawback that has been suggested to the use of an economizer is that it has the effect of decreasing the natural circulation rate in the boiler, because the feed water entering the boiler drum (which is added in such a way that it mixes with the recirculating water) is at a higher temperature. This effect, however, has been considered and seems to be of negligible practical significance.

Tube Welding: The integrity of the tube to tube-sheet welds on steam generators having heavy water in the tubes is a question of great concern. Careful attention must be given to this question in the stress analysis, qualification of welders and welding procedures, and inspection and testing. Despite all such precautions, only the operation of a mockup or prototype unit will satisfactorily resolve this question.

Separating Section: The steam disengaging area and moisture separating space in vertical U-tube steam generators is limited as compared with horizontal boilers, and care must be exercised to provide efficient moisture separation.

d. Organic Coolant

The same types of steam generators may be considered for the organic-cooled reactors as for the heavy-water-cooled reactors. Horizontal units were ruled out for the same reasons of building space, so only the vertical U-tube and once-through forced-circulation units were considered in any detail. Although the U-tube type was selected for this study, the once-through design also appears to be feasible. In this design, the lower-pressure organic liquid is on the shell side and the water and steam on the tube side. The economizer, boiler, and superheater sections are all in the same shell. Based on a design concept suggested by one manufacturer, two once-through forced circulation units were roughly sized, as follows:

600-Mwt - 9 ft diameter x 70 ft long

900-Mwt - 11 ft diameter x 70 ft long

The units may also be shortened somewhat at the expense of an increase in diameter.

In the case of the U-tube boiler design, the high cost of heavy water is not involved and it is economical to provide an economizer and a superheater in separate shells. The economizer and superheater units are of conventional shell and tube design and have no unusual design or fabricating problems. The problems in connection with the boilers are essentially the same as those for the steam generators for heavy-water reactors, but are in general less severe because of the lower design pressures involved.

6. PIPING

The economical pipe velocity for heavy water is high because of the high value of heavy water. Optimization is difficult because of ramified effects on system pressure, pump design, etc., in the velocity range of interest, and because of the somewhat indefinite limitations imposed by factors such as vibration, cavitation, erosion-corrosion, etc. A maximum velocity of 46 ft/sec was selected by judgment.

Information has been received from Hanford of flow loop tests with water in carbon-steel pipe at velocities of 18, 42, and 86 ft/sec, which showed no effect of velocity on corrosion. The tests were made with deionized and deoxygenated water at pH 10, at 295°C, and were run for a duration of 1500 hours.

7. EQUIPMENT ARRANGEMENT

a. Number of Loops

The most important question to be decided with respect to the arrangement of primary coolant loop equipment is the number of loops to be provided. This involves questions of reliability of cooling, feasible equipment sizes, costs of the loops, and economical use of building space. For the 3500-Mwt heavy-water-cooled reactor we considered three cases, each involving one pump and steam generator per loop, as follows:

Number of loops	8	6	4
Pump size, gpm	43,700	58,300	87,500
Steam generator size, Mwt	450	600	900

We decided that four loops is the minimum number we would consider safe from a reliability standpoint, for a plant of this large size. As it turned out, this also happens to be the minimum from the standpoint of the maximum feasible sizes of pumps and steam generators, as brought out in the discussion in Sections III, C, 4 and III, C, 5.

The total costs of the loops, including equipment, piping, and heavy-water inventory, should not differ very much for four, six, or eight loops. Very preliminary information indicates that costs for pumps and steam generators will not decrease significantly as their size increases. The cost of pipe will probably increase somewhat as the number of loops decreases, and the heavy-water volume will remain constant or increase slightly. The only offsetting costs which might decrease as the number of loops decreases are costs for instrumentation and valves, and piping installation costs.

The optimum number of loops, four or more, then depends mainly on arrangement considerations. Studies showed that, with allowance for space requirements for fuel handling, eight loops become quite crowded unless they are moved radially outward, which increases the building size and heavy-water inventory. Six loops, on the other hand, can be accommodated around the reactor satisfactorily.

The questions and problems associated with the large steam generators required for a four loop system makes this system undesirable in the absence of any cost advantage, especially since it is not required from an arrangement standpoint. In fact, a four loop system is less satisfactory than a six loop system, because of piping flexibility problems.

A six loop system is therefore the preferred arrangement for the 3500-Mwt heavy-water-cooled reactor. For the 8300-Mwt heavy-water-cooled reactor, arrangement considerations unquestionably dictate the use of the largest feasible equipment, and at the present time this leads to a requirement for nine or ten loops, each of which is about the same size as the loops in the four loop system for the 3500-Mwt reactor. Any larger number of loops would create serious arrangement problems.

For the organic-cooled reactors, these considerations turn out in about the same way as for the heavy-water-cooled reactors, and the same number of loops have been specified in the reference designs.

b. Design Problems

A number of design problems which affect the arrangement of the loops are mentioned here, not that they affect the feasibility of the system, but to point out some of the possible alternatives. Most of these questions have not been pursued to firm conclusions; so the loop arrangements described for the reference plants are by no means thought to be the best possible.

The paramount considerations in respect to loop arrangements are: (1) the loops should be as close to the reactor and as compact as possible to minimize the size of the containment shell and, especially in the case of the heavy-water coolant, to minimize the inventory; (2) the arrangement must provide adequate flexibility of the piping to keep stresses and reactions within acceptable levels; (3) the arrangement should provide adequate access for equipment maintenance and replacement.

These objectives conflict to some extent, and compromises must be made. Other secondary considerations will also be mentioned.

As previously noted, the first consideration led to the selection of vertical steam generators. Vertical pumps also are preferred, from a cost and weight standpoint, but piping considerations lead to consideration of horizontal pumps in some cases.

Piping flexibility is a difficult problem in all cases, and the arrangements shown for the reference designs are only marginally adequate in this respect. The problem can be alleviated considerably by providing pump and steam generator supports which permit these components to move; but this is not readily accomplished because of the weights involved. It is not practical to support the pumps from the piping, as is sometimes done in other systems, because of the weight of the motors.

Consideration was given to making provision for maintenance on a loop while the reactor is operating at partial power on the other loops. Insofar as providing the necessary shielding around each loop is concerned, this is practical; however, it is questionable whether the expense of the shielding is justified. It is not shown in our reference designs. Block valves are provided which permit a loop to be taken out of service while the others are operating.

We consider it necessary to be able to remove a steam generator from the building and replace it, and provisions for doing this were made in reference designs. This is also a question requiring further study, particularly for the heavy-water-cooled reactors, where a separate shield door and a panel in the building shell is required for each unit.

Following our practice for Savannah River Plant reactors, we considered making the pump motors accessible for inspection during plant operation, as shown in the reference designs. For the vertical pumps this makes a problem of access to the pump for disassembly; and for horizontal pumps it necessitates a long shaft between the pump and motor. At present it appears preferable to avoid these problems by putting the motors in the process rooms with the pumps and performing any necessary surveillance by instrumentation.

Relocating the steam generators at a higher elevation relative to the reactor in order to provide natural circulation of the primary coolant in the event of pump failure has advantages and should be considered.

The location of the pumps physically in the building and relative to the steam generators in the flow circuit is determined mainly by NPSH considerations. In the case of the heavy-water-cooled reactors, it is quite difficult to provide enough NPSH so that the pumps can be operated at full speed when the reactor is depressurized; consequently, a low-speed drive or separate auxiliary pumps must be provided for shutdown cooling. We have chosen to use auxiliary pumps for this purpose. Under these conditions, the elevation of the main pumps is not of great importance, because more than sufficient NPSH is available during operation.

On the other hand, consideration ought to be given to making it feasible to operate the pumps at normal speed under any conditions of reactor pressure. To do this it is necessary to use larger low-NPSH pumps, to locate them about 100 feet below the pressurizer, and to make the pressurizer connection to the loops in the pump suction lines. The last requirement means that either the pumps must be relocated upstream of the steam generators, or cross-connections must be made between the suction lines, or separate connections must be made from the pressurizer to the individual loops. The last two schemes increase the heavy-water inventory.

In the case of the organic-cooled system it appears necessary for several reasons to be able to maintain full flow with the main pumps and eliminate any external pressurization. To accomplish this, the pumps are located low in the building on the downstream side of the steam generators, with atmospheric pressure surge tanks at a higher elevation connected directly to the pump suction lines.

D. Secondary Cooling System

Within the scope of this study we did not consider it necessary to consider any problems outside the reactor area except the availability of large-size turbine-generators, which is a matter not only of direct importance to the economics of the plant but which has a strong bearing on the operating conditions selected for the reactor and primary cooling system.

The largest units currently offered by two major United States manufacturers of such equipment are a tandem-compound, four-flow, 43" last-stage blade length turbine (TC4F-43") by one, and a tandem-compound, six-flow, 44" last-stage blade-length (LSB) turbine (TC6F-44") by the other. For 650-psia saturated steam (from the heavy-water-cooled reactor plants), the first unit has a maximum capability of about 675 Mwe when the exhaust loading is approximately 10,000 lb/(hr)(ft^2). This loading constitutes a present-day limit for exhaust steam in the "wet" region. The largest generator size available from this manufacturer is 832 MVA. The machine offered by the second manufacturer has a nominal rating of 1000 Mwe, driving a 1000-Mwe generator.

Thus, one TC6F-44" machine suffices for the 3500-Mwt heavy-water cooled reactor, and three for the 8300-Mwt case. A slightly higher gross thermal efficiency (about 1.5%) could be obtained by using two and four of the smaller TC4F-43" machines for these cases, but this gain is not likely to outweigh the higher investment which would accompany the additional machine.

It is estimated that one 1000-Mwe machine would suffice, for the 3500-Mwt heavy-water-cooled reactor, for steam pressures down to roughly 600 psia; below this pressure, two machines would be required.

For the organic-cooled reactors at 492-psia superheated steam, the TC6F-44" machines are not recommended because of blade strength limitations due to the high temperature and low pressure of the steam. For these cases the TC4F-43" machines are used. A possible alternative for the 8300-Mwt case is to use six TC4F-38" units instead of five TC4F-43" units to obtain a slightly higher efficiency. Here again, however, the cost of the additional machine probably does not justify this choice.

In all cases, the ratings of the machines as described in the reference designs have been adjusted by Sargent & Lundy, Engineers to fit the specific plant conditions.

A machine identical to the TC4F-43" machine, except that only one two-flow low pressure element is used, is now being built for the Bodega Bay nuclear power plant. The first machine using 43"-long last stage blades by the first manufacturer will come into operation this year in a conventional steam plant, and the first nuclear application, ie, saturated steam, using 43" LSB will be in the NPR at Hanford.

The first 44" LSB machines by the second manufacturer will be used in the connecticut Yankee and San Onofre nuclear power plants. Both the Yankee and the Shippingsport nuclear power plants use turbines with 40" LSB by this manufacturer, with no major problems to date.

The manufacturers are working on development of larger machines having 50" and 52" last-stage blade lengths, so an increase in capabilities within the foreseeable future can be expected.

E. Fuel Handling

NOTE: The fuel handling concepts described in this report involve handling the fuel in two or three segments, to shorten the fuel handling machine. However, it is likely that core nuclear considerations will require the use of full-length fuel assemblies. This affects the fuel handling machine and the head room required for its operation.

Even the most cursory examination of the problems of designing and building a safe and reliable on-power refueling machine would take us beyond the scope of this study, and we have therefore limited ourselves to suggesting a conceptual method for fuel handling based on an adaptation of the type of fuel handling machine developed by AECL for the Douglas Point nuclear power plant.

The Douglas Point on-power fuel handling machine is essentially completed and has been undergoing tests. An earlier on-power refueling machine from which the Douglas Point machine evolved is in operation on the NPD. In the first on-power testing of the NPD machine in December 1962, a heavy-water spill occurred as a result of a snout seal failure. Design improvements have since been made to prevent this type of accident. The first successful on-power refueling at NPD occurred in November 1963. Since then, on-power refueling has been performed routinely. As of March 1964, at least thirty fuel transfers have been accomplished without incident. The only problem has been that it is judged necessary to replace the "Viton B" O-ring seal on the snout every two or three cycles, which affects the fuel management program to some degree.

Another version of an on-power refueling machine designed specifically for the Douglas Point reactor has been developed by American Machine & Foundry Company under the AEC-AECL Cooperative Program. Out-of-reactor operation of components of this machine has been demonstrated at Greenwich, Connecticut.

The satisfactory progress in development of on-power refueling by AECL and AM&F is very encouraging with respect to the feasibility of applying on-line refueling to the reactors under study. The differences involved, such as the vertical arrangement, the use of long fuel assemblies, and the consideration of organic-cooling, do not necessarily make the feasibility look better or worse. Nevertheless, the differences are there, and despite whatever advantage may be taken of the accomplishments of AECL and AM&F, the development of an on-power refueling machine will be the major development effort required for any of the reactors considered in this report.

F. Reactor Control

Reactivity control problems of large-size reactors differ from those for smaller reactors because of flux flattening requirements and spatial xenon effects. Satisfactory methods of control of large heavy-water-moderated reactors have been worked out for the Savannah River production reactors and are applicable to the power reactors under consideration. The method depends on providing an instrument system that measures spatial power distribution and a control system capable of adjusting the relative spatial power distribution. This system may be under the central control of a human operator at a console or of a process control computer which receives and analyzes the data from the instrument system and feeds back the appropriate signals to the control system. Either method is practical; the process control computer may be economically justifiable for plants of the sizes under consideration.

1. CONTROL SYSTEM

The control system of the reactor consists of control clusters placed uniformly throughout the central region of the reactor core. Each cluster contains four or more individual rods that are used to control the power level and the power distribution. In addition to these, there is a set of individual rods for safety and shim control. The control clusters are located on regular lattice positions, a distance sufficiently far from adjacent fuel positions such that the local flux perturbations in adjacent fuel are not severe. Thus, with a relatively large number of control positions spread uniformly over the flat zone, the control poison can be distributed uniformly without introducing large local perturbations.

2. POWER LEVEL CONTROL

As in all power reactors, control must be maintained over the reactor power on a minute-by-minute basis. Due to the negative prompt coefficient associated with heating of the fuel, the reactor will tend to maintain a constant power in the absence of control-rod movement. To follow changes in the turbine load it is necessary to move control rods to match the reactor power to the turbine demand and also to maintain a constant reactor inlet temperature.

3. CONTROL OF THE POWER DISTRIBUTION

In addition to control of the power level, it is desirable to control the power distribution in the core. In this way the average power density of the core can be increased and the reactor size reduced. To achieve these savings it is necessary to (1) have instruments that measure the power distribution and (2) have a control system capable of adjusting the relative power distribution.

Thermocouples in the outlet lines of each pressure tube measure the radial power distribution. Gamma ray detectors in instrument thimbles in the reactor core measure the axial power distribution at three or more points along the length of the reactor core. These axial power monitors are placed in three or four interstitial positions in the flat zone.

With control rods placed uniformly throughout the flat zone of the reactor core, positive control of the power distribution can be maintained at all times. The control-rod complement during reactor operation consists of partial length rods located approximately at the reactor midplane and other full-length rods either fully inserted or dipping in from the top. The axial power distribution is maintained by slight movements of the partial length rods. The radial distribution is controlled by the relative movement of the control rods moved either individually to trim out local perturbations or moved in gangs to adjust the flux distribution over larger areas.

4. EFFECT OF XENON ON THE POWER DISTRIBUTION

The power distribution of a reactor can be perturbed by "wandering" xenon, and the size of this perturbation increases (1) as the thermal flux increases, (2) as the size (in units of the migration area) of the reactor increases, (3) as the temperature coefficient increases (becomes more positive or less negative), and (4) as the magnitude of the initiating perturbation increases. There is a threshold of instability that can be defined in terms of Items 1, 2, and 3 (see References 46 and 47). When the reactor flux is above the threshold value, the power distribution, if left uncontrolled, will oscillate with diverging amplitude; below the threshold the amplitude is damped. In practice, however, the damped as well as the undamped oscillations must be controlled. All four of the reactors under consideration are above the threshold for spatial oscillations and can be controlled through the use of the detection system for measuring the power distribution and the control system for adjusting the power distribution.

Experience at Savannah River has shown that the relatively simple methods described in Reference (46) are adequate for predicting the threshold and the period of oscillation for xenon instabilities and that the reactor instrument and control systems for measuring and controlling the power distributions are adequate.

Tabulated below are threshold fluxes for the four reactors. These calculations are based on the "worst" conditions, ie, the maximum end-of-life flux and a flat power distribution. All of the reactors are unstable with respect to axial, radial, and azimuthal oscillations.

Coolant	D_2O	D_2O	Organic	Organic
Power, Mwt	3500	8300	3500	8300
Operating flux, 10^{14}n/(cm^2)(sec)	2.5	2.5	1.1	1.1
Threshold Flux, 10^{14}n/(cm^2)(sec)				
Axial	0.35	0.35	0.17	0.17
Radial	0.8	0.13	0.4	0.12
Azimuthal	0.21	0.07	0.17	0.07
Doubling time, min				
Axial	230	230	23	23
Radial	1200	30	150	16
Azimuthal	100	16	24	8

Xenon oscillations are set off by reactor noise that is present in the reactors at all times. In heavy-water-moderated reactors this noise is of the order of 1-2% in the spatial power distribution. This noise will set off xenon oscillations, which, if left uncontrolled, will continue to grow. The rate at which this perturbation in the power distribution grows is given in the table in terms of the doubling time for the perturbation. These doubling times range all the way from infinite, when the reactor is at the threshold, to values of the order of about 5-10 minutes. For example, in the most severe case, which is represented by the 8300-Mwt organic-cooled D_2O reactor, a 2% perturbation in the power distribution, if uncorrected, would grow to a 4% perturbation in 8 minutes. These time periods are long compared with the response time of the reactor control loop, including the human or computer operator. Thus, we conclude that the control of spatial oscillations due to xenon depends primarily upon the existence of adequate instrument and control systems rather than upon the proximity of the operating flux to the threshold flux.

G. Pressure–Volume Control

The primary cooling system for the heavy-water-cooled reactors requires a pressure-volume control system to protect it against pressure changes resulting from changes in the volume of water in the system. The volume of water in the system may change due to addition of water by the makeup pumps or removal through the letdown valves or leakage. More importantly, the volume may change due to temperature changes resulting either from a change in reactor power output or a change in the secondary cooling system which affects the heat transfer rate in the steam generators.

Control of pressure-volume fluctuations incidental to normal operation is accomplished with a pressurizer vessel, which acts in the first instance as a surge tank to accommodate the change in volume. Two types of pressurizers are in use: vaporizers and gas pressurizers. Vaporizers apply pressure to the system by means of the vapor pressure of the coolant, which is heated in the vessel with electric heaters to the saturation temperature corresponding to the desired pressure. Gas pressurizers use an inert gas to apply pressure to the system.

A vaporizer is used not only to apply the system pressure initially, but also to control it. This is accomplished by means of the electric heaters and cold-water sprays in the vapor space, which raise or lower the temperature in response to a pressure signal from the vaporizer.

A gas pressurizer, on the other hand, is used to apply the pressure but it does not control pressure directly. Instead, it maintains constant pressure through control of the liquid volume. This is accomplished by adding or withdrawing water to or from the system in response to a liquid-level signal from the pressurizer.

Liquid level must also be controlled in a vaporizer to keep the level within the functional limits of the unit, but this is incidental to the principal function of the vessel. Changes in water level within the functional limits do not affect the pressure. In fact, in a system with a vaporizer, pressure fluctuations can only be caused by the action of the vaporizer itself and its control loop.

In a gas pressurizer, the pressure changes as the level changes, but, since the cause of the disturbance is a change in system volume (the pressure change is an effect), the control is only on the liquid level, not on the gas pressure. The gas pressure is only adjusted manually when it is desired to change the pressure level of the system or to adjust for accumulation of gas from decomposition or loss by leakage. If the pressurizer and its control system are designed correctly, the small pressure fluctuations incidental to its operation will be of no consequence.

Either type of pressurizer may be used on a pressurized water system. The merits and drawbacks of each follow from the foregoing discussion. The gas pressurizer responds more directly to the original source of the disturbance, but the vaporizer probably functions more smoothly. The vaporizer requires a spray supply system but the gas pressurizer requires a gas supply system and, if the gas is helium, probably a gas recovery system. The vaporizer requires steam safety valves which are a possible source of loss of coolant whereas a gas pressurizer can be designed so as not to require gas relief valves. A vaporizer is considerably hotter than the remainder of the primary coolant system. A gas pressurizer tends to saturate the system with gas, which may form gas pockets in other parts of the system on cool-down.

Probably the steam safety valves are the principal drawback for a vaporizer, and the gas handling system for a gas pressurizer.

To protect the system against volume-pressure fluctuations too severe for the pressurizer to handle, pressure relief devices must be provided. Whichever type of pressurizer is used, liquid relief valves should be used for this purpose, because: (a) they are more effective than gas safety valves in counteracting the coolant volume increase which causes the pressure rise; (b) they do not produce a hazardous boiling condition in the primary system, as the opening of a gas safety valve may; and (c) it is simpler to cool and collect liquid discharge than steam discharge.

H. Heavy-Water and organic losses

1. HEAVY-WATER LOSSES

The magnitude of heavy-water losses is a question of considerable concern with respect to economic feasibility of the plants. For example, in the 3500-Mwt reactor plants, a loss rate of 100 lb/day corresponds to a cost penalty on the power produced of about 0.1 mill/kw hr; in the 8300-Mwt plants the same penalty is incurred by a loss rate of about 230 lb/day*. At these levels, the losses are tolerable, but losses much in excess of these rates would significantly penalize the plants; consequently, it is necessary to assess the problems of achieving low loss rates. The problems are of course more serious for the heavy-water-cooled reactors than for the organic-cooled reactors, in which all of the heavy water is at low temperature and pressure.

Losses of heavy water due to radiolytic decomposition, carry-out with spent fuel, and isotopic degradation are easily controlled and of negligible concern as compared with losses due to leakage from seals and joints of various types.

Achievement of the loss-rate goals must begin with development and testing of joints and seals and must be a primary consideration in all stages of design, equipment fabrication, plant construction, and operation. The

*Computed at a plant load factor of 80% and a projected heavy-water cost of $20/lb.

development and testing program should be directed towards selection of suitable designs and testing them for reliability under cyclic temperature and pressure conditions; in some instances new designs must be developed for specific purposes. Means for quantitative measurement of very small rates of leakage are essential for this work. See, for example, References (41) and (42).

In design, unusual weight should be given to simplifying systems, minimizing joints, providing for leakage collection from packing, flanges, etc., using welded connections and seals as much as possible, and providing vapor recovery facilities for certain areas. Provision should also be made in design for accurate methods of inventory control, for leak detectors, and for measurement of heavy-water concentrations in water and air streams. In fabrication and field construction, stringent requirements must be set for quality control, inspection and testing of joints and seals, including the steam generator tube-sheet welds. Some operational precautions, many of which can be simplified and improved by advance design planning, are noted in Reference (44).

Any assessment of expected plant losses at this time is very difficult and risky. Leakage tests on valves, seals, etc., cannot be extrapolated to over-all plant losses. More useful information may be obtained from actual operating experience of heavy-water reactors. The principal sources of such information at the present time are the Nuclear Power Demonstration Reactor (NPD), the Plutonium Recycle Test Reactor (PRTR), the Heavy Water Components Test Reactor (HWCTR), and the Savannah River production reactors.

The NPD contains 132 pressure tube assemblies, operates at 1040 psig, and at 82 Mwt. Heavy-water inventory is 28,000 lb in the primary system and 136,000 lb in low pressure systems. It has been in operation since January 1962. Losses from then to October 1963 averaged 40 lb/day. From May through August 1963, during which period the reactor was in operation, monthly average losses ranged from 23 to 53 lb/day. The major losses are from numerous small leaks into the boiler-room air, which is discharged to the stack.

The PRTR is a 70-Mwt reactor with 85 pressure tubes, operating at 1050 psig. During initial power operation of the plant in late 1961, unrecovered heavy-water loss rates were reduced from 200 lb/day to about 50 lb/day (43). During April and May 1962, losses were still reportedly averaging 54 lb/day (44).

The HWCTR is a pressure vessel reactor, rated at 61 Mwt, which operates at about 1000 psig. System inventory, mostly hot and pressurized, is about 70,000 lb. Unrecovered heavy-water losses during the first year of operation (1962-1963) averaged 25 lb/day. In late 1963 a systematic campaign of leakage reduction was put into operation. Losses are calculated by very detailed inventory procedures and checked against losses calculated from stack air and boiler water concentrations. Loss rates for a 23-day period in January 1964, averaged 14 lb/day, which included 5.5 lb/day through leaks at steam generator and purge cooler tube to tube-sheet joints. The unrecovered and unaccounted for losses were therefore 8.5 lb/day.

Attempts to repair the HWCTR steam generator leaks by peening and tube rolling have had only limited success. The leaks are apparently distributed over numerous tubes and shift from place to place. For instance, in March 1964, soap bubble tests on one of the two steam generators at 400 psig showed 63 leaking welds, which included 20 which had not leaked on a previous similar examination, and 14 which had been repaired at the time of the previous examination.

The reductions achieved in the HWCTR loss rates by the efforts over the past few months are attributed partially to tightening of valves, flanges, and other seals where leakage was detected, and partially to improvised collection systems. Leakage is now collected from the monitor pin joints, the purge cooler head flange, and the packed-stem valves (which were provided with lantern rings and leak-off connections). This collected leakage is not included in the loss rate figures.

The experience of these plants illustrates well the problems involved; at the same time, they are encouraging as to the improvements that can be achieved by suitable efforts, particularly in the HWCTR.

Higher losses are to be expected in large plants than in small plants, but not in proportion to reactor size or system inventory. In many respects the large power reactors will be more favorable for low loss rates; for example these small reactors which are operated for research and testing purposes are shut down and opened more frequently, and have relatively more connections, sampling points, valves, etc. On the other hand, in some respects the large-size reactors compare less favorably; the operating pressure is 70% higher, for example, and the pressure-tube end closures, which must be removed and replaced mechanically for on-power refueling, are a potential source of leakage. Consequently, no predictions should be made from the experience to date of these reactors.

The only reactors at all comparable in size to the power reactors under consideration, and for which loss rates are available, are the heavy-water production reactors at Savannah River Plant. Loss rates per reactor during the past year are in the range of 60 to 80 lb/day. Although these are low-pressure reactors, much of the water is near the boiling point and the reactors are more open to the atmosphere than a power reactor would be. The production reactors are also shut down and opened for refueling much more frequently than would be the case for power reactors.

When the Douglas Point plant is in operation, it may be possible to arrive at a reasonable idea of the loss rates to be expected for large power reactors. A much better basis for prediction would be obtained from operation of a reasonable-size prototype of the reactors under consideration.

2. ORGANIC LOSSES

Organic losses in the organic-cooled reactors are almost entirely due to degradation from heat and radiation. Loss rates are estimated at 500 to 1000 lb/hr in the 3500-Mwt reactor and proportionately higher in the 8300-Mwt reactor. At a price of $0.17/lb, the high-side figure adds about 0.2 mill/kw hr to the electric-power cost.* These figures are only approximate, but data are available to determine the costs reasonably accurately for a specific design. If the losses are kept to or below this level, the cost penalty is not severe. Also, some study has been made for Piqua of the economics of reclaiming usable material from the high boiler residue by catalytic hydrocracking, with encouraging results (40). In large-size power plants, the economics of recovery should be more favorable.

I. Reactor Containment

1. DESIGN CRITERIA

The problems of reactor containment depend on the plant site, which is not specified for this study. Therefore, we can consider this question only in a general way. For this purpose we have assumed a maximum allowable leakage rate from the Containment Building of 0.1% of the contents per day. On this basis, the following site requirements are estimated to meet the standards of Title 10, Code of Federal Regulations, Part 100, 1962, for normal site conditions:

*It may be noted that this cost increment equals the cost associated with a heavy-water loss of 200 lb/day.

Reactor power, Mwt	3500	8300
Minimum exclusion area radius, miles	0.5	0.75
Minimum radius of low population zone, miles	7.5	11
Minimum distance to nearest population center, miles	10	15

These figures are based on consideration of the data from reference (48) with the application of what we consider to be a conservative factor, 10, for the reduction in the escape of iodine from the building as a result of the action of the spray system and the filter-absorber units. In the absence of any specific site information, such criteria appear to be reasonable.

A maximum permissible leak rate of 0.1% per day permits the use of a "standard" steel containment shell design, which proved to be capable of achieving leak rates less than this; for example, Yankee Atomic Power Plant (125-foot-diameter sphere, 34.5-psig design pressure), Indian Point (160-foot-diameter sphere, 25-psig design pressure), and Dresden (190-foot-diameter sphere, 29.5-psig design pressure) have all achieved lower rates. In principle, at least, a given leak rate expressed as percent per day should be more easily achievable the larger the vessel.

The design pressure and volume of the containment shell depend on the quantity of stored energy which is assumed to be released to the building for design purposes. For this purpose, in the case of the heavy-water-cooled reactors, it is assumed that the entire contents of the primary cooling system and the contents of the secondary side of one steam generator are released to the building with steam flashing to equilibrium. In the case of the organic-cooled reactors, the same event is assumed to take place, and, additionally, it is assumed that just enough water is available (from rupture of the shield or moderator systems, for example) to produce saturated steam by contact with the hot organic material, with no liquid water left over.

It is assumed that the cause of this accident may be a complete break of a main primary cooling pipe. In such a case, the pressure builds up so rapidly that loss of heat to the building structure or to the environment is negligible before the maximum pressure is reached; therefore, such losses are not considered in computing the design pressure. It is also assumed that in this short time (less than a minute) the spray system has not been actuated.

The inclusion of the contents of the secondary side of a steam generator in the incident is based on the supposition that the primary system failure may result in rupture of a steam generator or steam piping by missiles or equipment dislocation, but that this could not happen to more than one unit. Another possible cause leading to the same result is a failure of one or more steam generator tubes, causing excessive pressure on the secondary side which is then discharged into the building through the safety valves.

2. SHELL DESIGN

On these assumptions, the internal design pressure for containment buildings for heavy-water-cooled reactors is:

$$p = \frac{65.5}{1.06(V/w) + 1.86}$$

and for organic-cooled reactors is:

$$p = \frac{u' - 39 + 0.95\ V/w_o}{1.2(V/w_o) + 0.83}$$

In these formulas, p is the building internal design pressure in psig; V is the free volume of the building in cubic feet; w is the mass of hot water in the primary system and the secondary side of one steam generator, in pounds; u is the weighted initial internal energy of this water, in pcu/lb; w_o is the mass of hot organic liquid in the system, in pounds; $u' = u_o + (w_b/w_o)\ (u_b - u_h)$; u_o is the initial internal energy of the hot organic liquid, in pcu/lb; w_b is the mass of water on the secondary side of one steam generator, in pounds; u_b is its initial energy in pcu/lb; and u_h is the initial internal energy of the additional water flashed by the organic material, in pcu/lb. The formulas are valid from about 10 to 25 psig.

Any combination of building volume and design pressure corresponding to these formulas will satisfy the assumed containment requirements; as the building size is decreased, the pressure and shell thickness both increase. The minimum practical building size is the most economical, except that if the shell thickness exceeds 1-1/2 inches (for the materials used for this construction), the cost increases because of ASME Nuclear Code requirements for stress relief of field welds in such shells. Consequently, the size selected is either the size dictated by this shell thickness or the minimum size permitted by the equipment arrangement, whichever is larger. Since spherical shells of a given size and pressure require a lesser shell thickness than cylindrical shells, a spherical shell may be smaller and more economical than a cylindrical shell if the equipment arrangement permits. Generally speaking, as the power rating of a reactor increases, the economical containment size (ie, the size for which the shell thickness is 1-1/2 inches) increases faster than the size required to house the equipment; for small reactors the building size is generally dictated by equipment arrangement and only for very large reactors by the containment requirements.

Cylindrical shells, with an ellipsoidal base and hemispherical dome, and spherical shells were considered for reactor containment. In all four cases the size of a cylindrical shell dictated by the 1-1/2-inch shell thickness exceeds the size required to house the equipment satisfactorily; consequently, a spherical shell is smaller and more economical. The spherical shell does not appear to create serious arrangement problems. With the present equipment arrangement, the spherical shell diameters required to house the equipment (250 ft and 350 ft for 3500 and 8300-Mwt reactors, respectively) are larger than the economical containment sizes, which are 200 ft for the 3500-Mwt reactors, and about 310 ft for the 8300-Mwt reactors. Consequently, more efficient equipment arrangements will in all four cases permit economies to be made in the size and cost of the containment shells.

External live loadings as well as internal pressure must be considered in the design of large containment spheres. For design purposes we assumed maximum wind loads and snow loads of 30 lb/sq ft each, superimposed. On this basis, the shell thickness for the 3500-Mwt reactors is controlled by internal pressure and for the 8300-Mwt reactors by external loading, the cross-over point being about 6000 Mwt. The external loading requires a shell thickness of 1-1/2 inches for a 350-foot diameter sphere; consequently, this is about the largest size (for the assumed design loading) that can be built with an unstiffened shell, if the 1-1/2-inch plate thickness limit is adhered to. However, larger spheres can be built by providing stiffening rings for additional support, and we do not see any practical limit to the size of sphere that can be built or the size of reactor that can be accommodated in this type of structure.

No unusual design or construction problems are foreseen with respect to the containment shells.

3. SPRAY AND HEAT REMOVAL SYSTEM

After the maximum pressure due to release of coolant to the building has been reached, the pressure will tend to rise slowly because of release of decay heat from fuel. However, the flow of heat into the building structure will more than compensate for this, so that actually the pressure will fall very slowly after the initial incident, until an equilibrium is reached. The spray and heat removal system, which is sized for a heat removal capacity equal to the rate of decay heat 10 minutes after reactor shutdown, permits decreasing the pressure more rapidly; roughly, it will reduce the gage pressure by half in 30 minutes.

4. ORGANIC FIRES

Organic fires are not taken into account in calculating the design pressure of the containment building because it is not credible that a fire could generate pressure at such a rate that the spray system could not cope with it. In the 3500-Mwt case, a fire that consumed all of the oxygen in the containment building would burn about 33,000 lb of the organic material, or 2.3% of the hot inventory. The heat from this fire added to the sensible heat initially in the organic would theoretically generate a total building pressure of 59 psig, or 2.4 times design pressure, if all the heat were applied to flashing water to steam. Actually, because of the relatively slow rate at which this heat is generated, the peak pressure would be less than this, and if the spray system were operating, the pressure would not rise above the design pressure unless the entire conflagration took place within 15 minutes, ie, a combustion rate in excess of 2000 lb/min. For example, if the entire 33,000 lb of organic were consumed in 10 minutes, a maximum pressure of 31 psig (25% over design pressure) would be reached (with the sprays operating).

The heat from an organic fire could cause a failure of the containment shell if applied directly to it. However, the shell is lined on the inside with at least 3 ft of concrete up to 30 ft above the equator (for structural reasons). All of the hot process equipment is below this level and covered with a thick concrete shielding floor. Therefore, any major organic fire would be confined to the lower part of the building, where the shell is protected from it by the concrete. Under these conditions it is highly unlikely that the steel shell would be damaged by the heat from the fire.

The containment buildings for the organic-cooled cases will be provided with a pressurized fog-nozzle system in appropriate areas for quenching organic fires. Special precautions will be required with respect to pipes and conduits which penetrate the containment building (eg, cooling water lines and electrical conduits) to prevent the possibility of a containment breach resulting from rupture of such a line by heat from an organic fire.

J. Conclusions

1. CAPACITY LIMITS

It is technically feasible to construct heavy-water-moderated pressure tube reactors up to at least 8300-Mwt capacity, based on either heavy-water or organic coolants.

The reactor structure itself has no well-defined size limitation, although of course the problems of fabricating and installing huge calandria and shield structures and the numerous coolant inlet and outlet pipes become progressively more difficult as the size increases. These problems would eventually impose a practical limit at some size beyond the range considered in this study.

The significant components that have capacity limitations and require multiple units to achieve the plant capacity are the steam generators and pumps in the primary cooling loops, the turbine-generators, and the fuel handling equipment. Current technology probably limits the size of the primary cooling loops to about 600 to 900 Mwt, and 900 Mwt appears to be an upper limit for the foreseeable future. Turbine-generators are currently limited to 1000 Mwe, but increases in capabilities during the next decade are foreseeable. On-power refueling machines are probably limited by their speed of operation to a capacity (very roughly) of 5000 Mwt per machine.

The principal limitation on the capacity of a reactor system is the space problem of arranging the cooling loops and refueling equipment around a central reactor in an efficient manner. This problem probably sets a practical limit to the capacity of heavy-water-cooled reactors on the order of 10,000 Mwt, based on ten or eleven loops and two fuel handling machines. Organic-cooled reactors should have a somewhat higher limit.

2. TECHNOLOGICAL STATUS

Nearly all parts of any of the four plants could be built with only rather modest engineering development efforts to advance current manufacturing and construction capabilities. The areas requiring the greatest amount of development work are the fuel handling machine, the pressure tube assemblies, and leaktight joints and seals. Of these three items, the first involves the greatest expenditure of time and money, but the latter two are no less important to the success of the plant. Seal development is somewhat less critical for organic-cooled than for heavy-water-cooled reactors because of the lower cost, lower pressure, and lower vapor pressure of the organic coolant and the lower pressure and temperature of the heavy water in these plants.

Of the three major areas of development work, only the fuel handling machine is associated particularly with large reactors. Work in the other two areas is required for any reasonable size of heavy-water power reactor that might be built, of the type we have proposed. The first such reactor to be built would presumably be smaller than those considered in this study, so that these areas of technology would be considerably advanced over the present state before design and construction of a 3500-Mwt or larger reactor would be undertaken.

To compare the technological status of heavy-water-cooled and organic-cooled reactors, the former are substantially ahead with respect to fuel development, pressure tube development, and coolant technology. In all other important respects, however, the technology required to design and build an organic-cooled reactor plant is as far advanced as for a heavy-water-cooled reactor plant, and in many respects, the problems are simpler because of the lower design pressure.

Although fuel design is outside the scope of this study, it should be noted that one of the most important technical factors which affect the design and cost of the plant is the maximum allowable surface temperature of the fuel cladding used for design purposes. This is true both for the heavy-water-cooled and the organic-cooled reactors. Any increase in this temperature limit permits a more economical plant design.

3. SAFETY

The large sizes of the reactors under consideration impose no unusual safety problems except in respect to on-power fuel handling. There is adequate background of experience to ensure reactor controllability. Standard steel-shell containment is feasible for any size reactor that might be built, and additional containment protection could be provided, if required.

Pressure tube reactors have their specific safety problems as do other types; a major one in this case is the integrity of the pressure tube assemblies.

The hazards of organic fires in this type of reactor are not judged to be particularly severe, and can be coped with by normal industrial methods.

4. COSTS

Although no attempt was made in this study to evaluate costs, certain preliminary conclusions regarding cost trends may be drawn. Insofar as the reactor system is concerned (the reactor and its auxiliary equipment and facilities including the containment building), unit capital costs will decline only slightly as the plant size increases in the range we are considering, because, for most of the major equipment, the cost will be almost directly proportional to power level. This appears to be true for the reactor structure, including the pressure tubes, for the primary coolant loops, as noted in Section III, C, 7a, and, in a stepwise manner, for fuel handling equipment. The costs of the reactor building shell and interior structures, and the heavy-water inventory, also seem to increase in direct proportion to the power level.

Therefore, contributions to decreased unit capital cost as the design power level increases must come mainly from minor auxiliary facilities or from whatever construction economies are obtainable by a larger scale of construction activity.

This conclusion is not incompatible with reductions in unit costs which might occur because the larger plants are built after the smaller plants and can take advantage of technological advances.

5. FEASIBILITY QUESTIONS

The major areas of uncertainty with respect to over-all plant feasibility are the following:

- Capital costs
- Heavy-water losses
- Pressure tube reliability and safety
- On-power refueling reliability and safety.

As noted, cost evaluation was outside the scope of this study. On the other three items, firm conclusions will not be possible until experience is obtained from the operation of prototype plants; however, the limited experience available to date is encouraging with respect to all three items.

In a somewhat different category are questions which relate to the plant location, which is not specified. The most important questions are the isolation and containment requirements and the ability to transport large components to the plant site by water. The answer to the latter question will very likely be affirmative because of the large requirements for condenser water. It should also be noted that the need for on-power refueling will depend considerably on the utility network into which the plant is incorporated.

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TABLE 1. DESIGN PARAMETERS FOR HEAVY-WATER-MODERATED REACTORS

Primary Coolant	D_2O	D_2O	ORGANIC	ORGANIC
Thermal Power to Coolant, Mwt	3500	8300	3500	8300
Fuel				
Material	UO_2	UO_2	UC	UC
Initial enrichment, % ^{235}U	1.2	1.2	1.2	1.2
Fuel density, % of theoretical	92	92		
Wt of fuel, lb/ft UO_2 or UC	20	20	27	27
Cladding	Zr-2	Zr-2	SAP	SAP
Cladding thickness, inches	0.020	0.020	0.020	0.020
Geometry (see fig. no.)	6	33	22	22
Heat rating limit $\int k d\theta$, watts/cm	40	40		
Cladding surface temp limit, °C	330	330	470	470
Max $\int k d\theta$, tube 1, watts/cm	30	30		
Max $\int k d\theta$, tubes 2 and 3, watts/cm	40	40		
Max fuel temp, °C	~1750	~1750		
Max cladding temp, °C	330	330	470	470
Max heat flux, pcu/(hr)(ft^2)	500,000	500,000	300,000	300,000
Max coolant velocity, ft/sec	49	49	58.8	58
Minimum burnout safety factor	1.7	1.7	1.7	1.7
Core Geometry				
Number of fuel positions	516	1258	604	1360
Number of control rod positions	37	89	37	81
Number of interstitial safety rod positions	40	80	40	80
Dia, core and reflector, inches	303	454	305	436
Core diameter, inches	266	414	265	396
Flat zone diameter, inches	150	240	168	286
Core length, feet	15	15	20	20
Lattice pitch (square), inches	10	10	9.25	9.25
Radial reflector (D_2O), inches	20	20	20	20
Axial reflector (D_2O), inches	24	24	24	24
Core, Thermal and Hydraulic				
Power to coolant, Mwt	3500	8300	3500	8300
Power to moderator, Mwt	142	337	143	340
Total fission power, Mwt	3642	8637	3643	8640
Fuel inventory, metric tons U	68	151	142	317
Avg specific power, Mwt/metric ton U	54	54	25	27
Avg power of flat zone assembly, Mwt	7.5	7.5		
Max assembly power, Mwt	9.5	9.5	7.77	7.66
Average fuel exposure, MWD/metric ton U	15,000	15,000	15,800	16,700
Inlet temperature, °C	267	267	280	280
Mixed outlet temperature, °C	304	304	380	380
Maximum outlet temperature, °C	320	320		
Coolant inlet flow, gpm	350,000	830,000	280,000	660,000
Average flow per assembly, gpm	680	660	462	485
Maximum flow per assembly, gpm	713	713	616	607
Primary Coolant System				
Coolant	D_2O	D_2O	Santowax-R[b]	Santowax-R[b]
Design pressure, psig	2,000	2,000	350	400
Reactor inlet temperature, °C	267	267	280	280
Reactor outlet temperature, °C	304	304	380	380
Total flow, gpm (at reactor inlet)	350,000	830,000	280,000	660,000
Number of loops	6	10	6	10
Number of pumps	6	10	6	10
Number of steam generators	6	10	6	10
Pressure drops, ft				
Fuel	110		394	
Pressure tube extensions and muffs	50		50	
Inlet and outlet piping and headers	173		130	
Main piping	144		54	
Steam generators	53		59	
Total	530	550	687	753
Piping material	C/Stl	C/Stl	C/Stl	C/Stl
Main piping, OD, inches	28	35	36	42
Inlet and outlet piping OD, inches	3-1/2	3-1/2	3-1/2	3-1/2

TABLE I. DESIGN PARAMETERS FOR HEAVY-WATER-MODERATED REACTORS (Con't)

Primary Coolant Thermal Power to Coolant, Mwt	D_2O 3500	D_2O 8300	ORGANIC 3500	ORGANIC 8300
Primary Coolant Pumps (each)				
Flow, gpm	58,300	83,000	46,700	66,000
Total dynamic head, ft	530	550	687	753
Brake horsepower	10,000	15,000	9,000	14,000
Pressure Tubes				
Material	Zr-2 25% C.W.	Zr-Nb	Zr-2(?)	Zr-2(?)
Geometry (see fig. No.)	5, 6	5, 33	21, 22	21, 22
Design temperature, °C	320	320	380	380
Design pressure, psig	2,000	2,000	350	400
Design stress[a], psi	16,400	26,000	14,300	14,300
Steam Generators (each)				
Heat load, 10^9 pcu/hr superheater	-	-	0.16	0.22
boiler	1.12	1.59	0.84	1.19
economizer	-	-	0.12	0.18
total	1.12	1.59	1.12	1.59
Heat transfer surface, sq ft				
superheater	-	-	20,400	29,000
boiler	70,600	100,000	35,600	51,800
economizer	-	-	7,700	11,000
total	70,600	100,000	63,700	91,800
Primary coolant temp, °C inlet	304	304	380	380
outlet	266	266	279	279
Secondary side temp, °C inlet	198	198	195	195
boiler	260	260	243	243
exit	260	260	359	359
exit condition	Sat.	Sat.	S.H.	S.H.
Secondary side boiler pressure, psig	665	665	503	503
Materials - primary side	Inconel(?)	Inconel(?)	Steel	Steel
secondary side	Steel	Steel	Steel	Steel
Turbine Generators				
Number of units	1	3	2	5
Type	TC6F-44"	TC6F-44"	TC4F-43"	TC4F-43"
Total steam flow, 10^6 lb/hr	14.3	34.0	13.1	28.5
Throttle pressure, psig	635	635	477	477
Throttle temperature, °C	257	257	358	358
Condenser pressure, in. Hg abs	1.5	1.5	1.5	1.5
Gross electric generation, Mwe	1054	2490	1240	2950
Containment Vessel				
Shape	Sphere	Sphere	Sphere	Sphere
Diameter, feet	250	350	250	350
Plate thickness, inches	1-1/4	1-1/2	1-1/4	1-1/2
Material	A201B	A201B	A201B	A201B
Internal design pressure, psig	25.0	19.4	24.8	15.5
External loading, psf	60	60	60	60
Design temperature, °C	110	102	110	95
Total volume, 10^6 cu ft	8.19	22.5	8.19	22.5
Free volume, 10^6 cu ft	6.1	18	6.1	18
Inventories, Tons				
Heavy water - hot pressurized	412	905	-	-
cold	351	765	384	738
total	763	1670	384	738
Organic	-	-	720	1250

(a)Design Stress Intensity Sm per ASME Nuclear Code.
(b)Trademark of the Monsanto Chemical Company.

TABLE 2. PROPERTIES OF "SANTOWAX-R"*

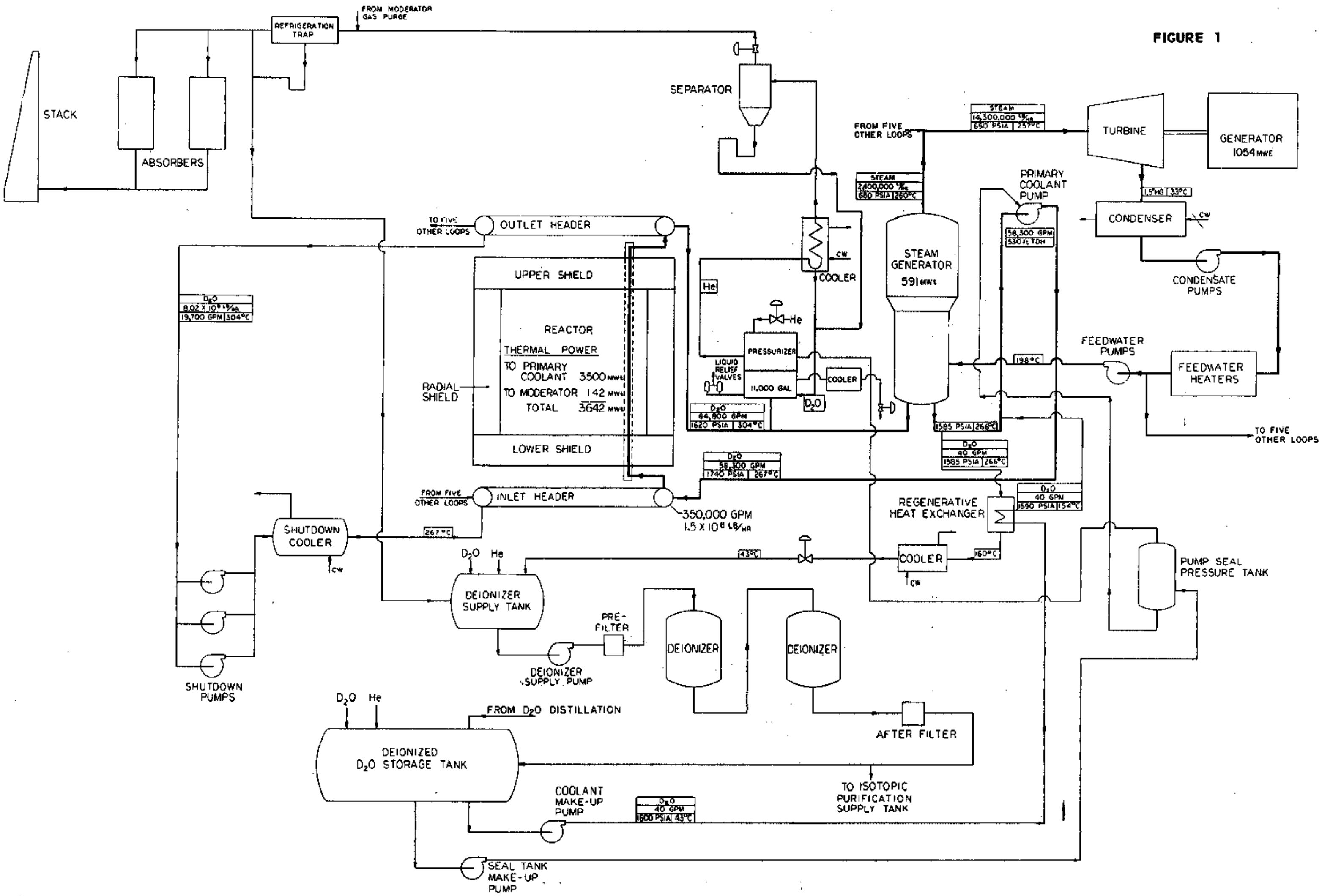
(Taken from Atomics International Report NAA-SR-Memo-3223, "The Properties of Santowax-R (Mixed Terphenyl Isomers) as Organic Moderator-Coolant").

Temperature, °C	280	280	380	380
High boiler residue (HBR) %	0	30	0	30
Density, g/cm^3	0.914	0.934	0.819	0.857
Viscosity, cp	0.40	0.95	0.25	0.47
Heat capacity, pcu/(lb)(°C)	0.521	0.500	0.544	0.516
Thermal conductivity, pcu/(hr)(ft)(°C)	0.0677	0.0677	0.063	0.063
Vapor pressure, psia	2	2	15	16

High boiler residue (HBR) %	0	30
Melting point - initial, °C	109	112
- final, °C	155	136
Heat of combustion, pcu/lb	9600	9600
Ignition temp of dust cloud, °C	620	
Minimum explosive concentration in air, oz/ft^3	0.035	
Radiolytic decomposition rates:		
G_{gas}, molecules of gas per 100 ev at 350°C	0.011	0.003
G_p, molecules of HBR formed per 100 ev in temperature range of 280 to 380°C	0.14	0.08
Pyrolytic decomposition rate in absence of air at 380°C, wt % HBR formed per hr	2×10^{-3}	

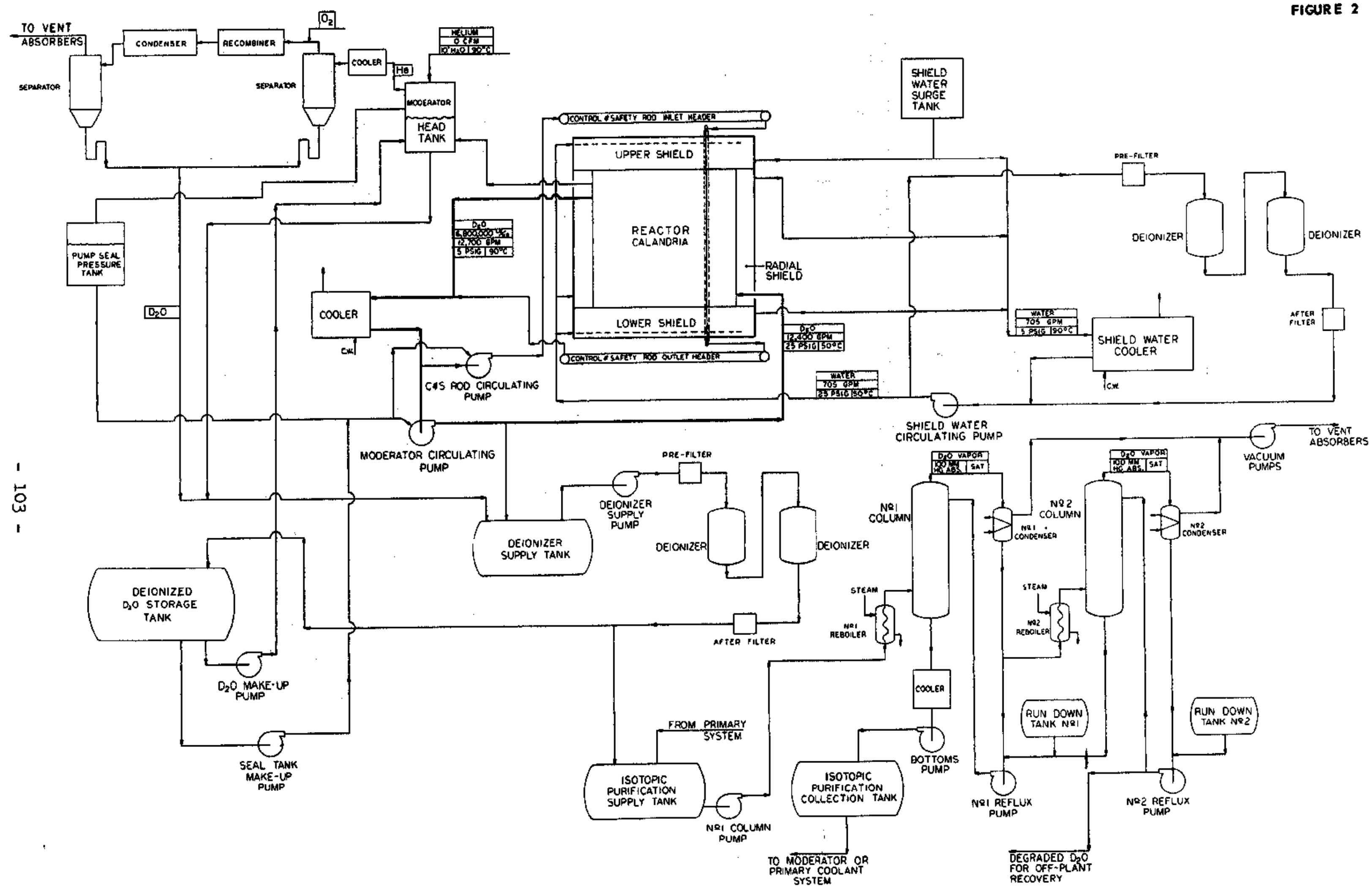
*Trademark of the Monsanto Chemical Company.

FIGURE 1



3500 MWT D_2O COOLED REACTOR
FLOW DIAGRAM
MAIN COOLING SYSTEM

FIGURE 2



3500 MWT D_2O COOLED REACTOR
FLOW DIAGRAM
MODERATOR & SHIELD COOLING SYSTEMS

3500 MWT D_2O COOLED REACTOR
REACTOR ARRANGEMENT
CROSS SECTION

FIGURE 3

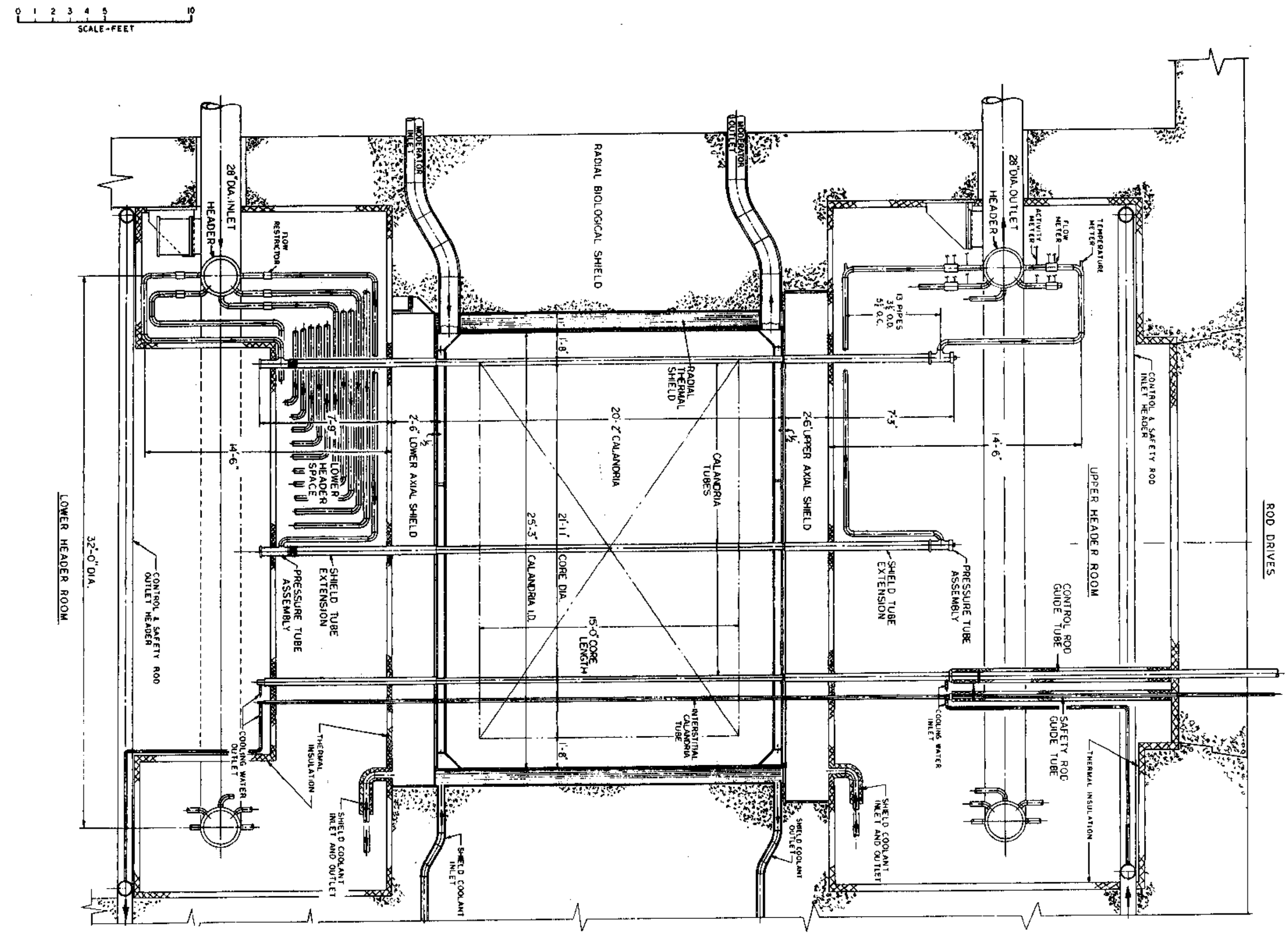
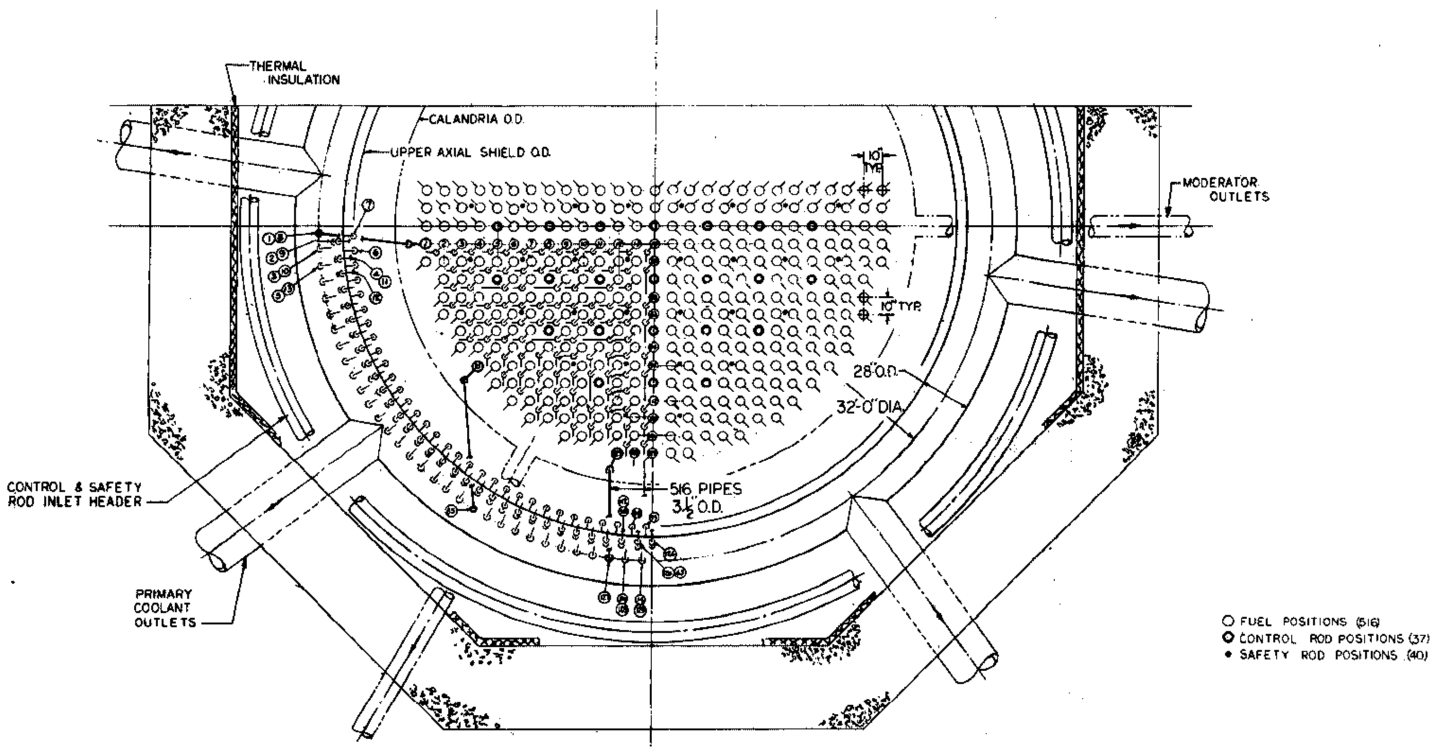


FIGURE 4



3500 MWT D_2O COOLED REACTOR
REACTOR ARRANGEMENT
PLAN

FIGURE 5

3500 MWT D_2O COOLED REACTOR PRESSURE TUBE ASSEMBLY

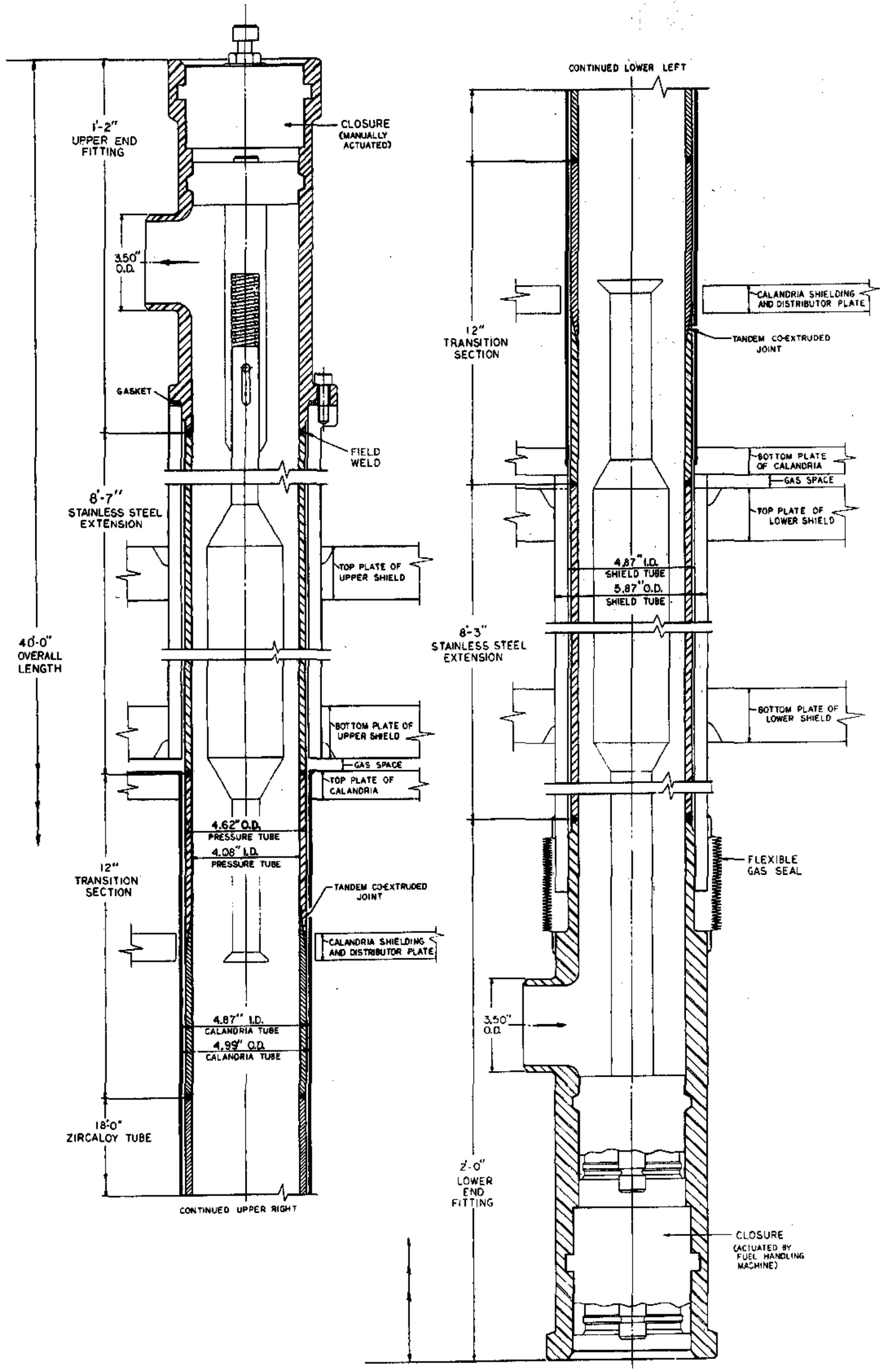
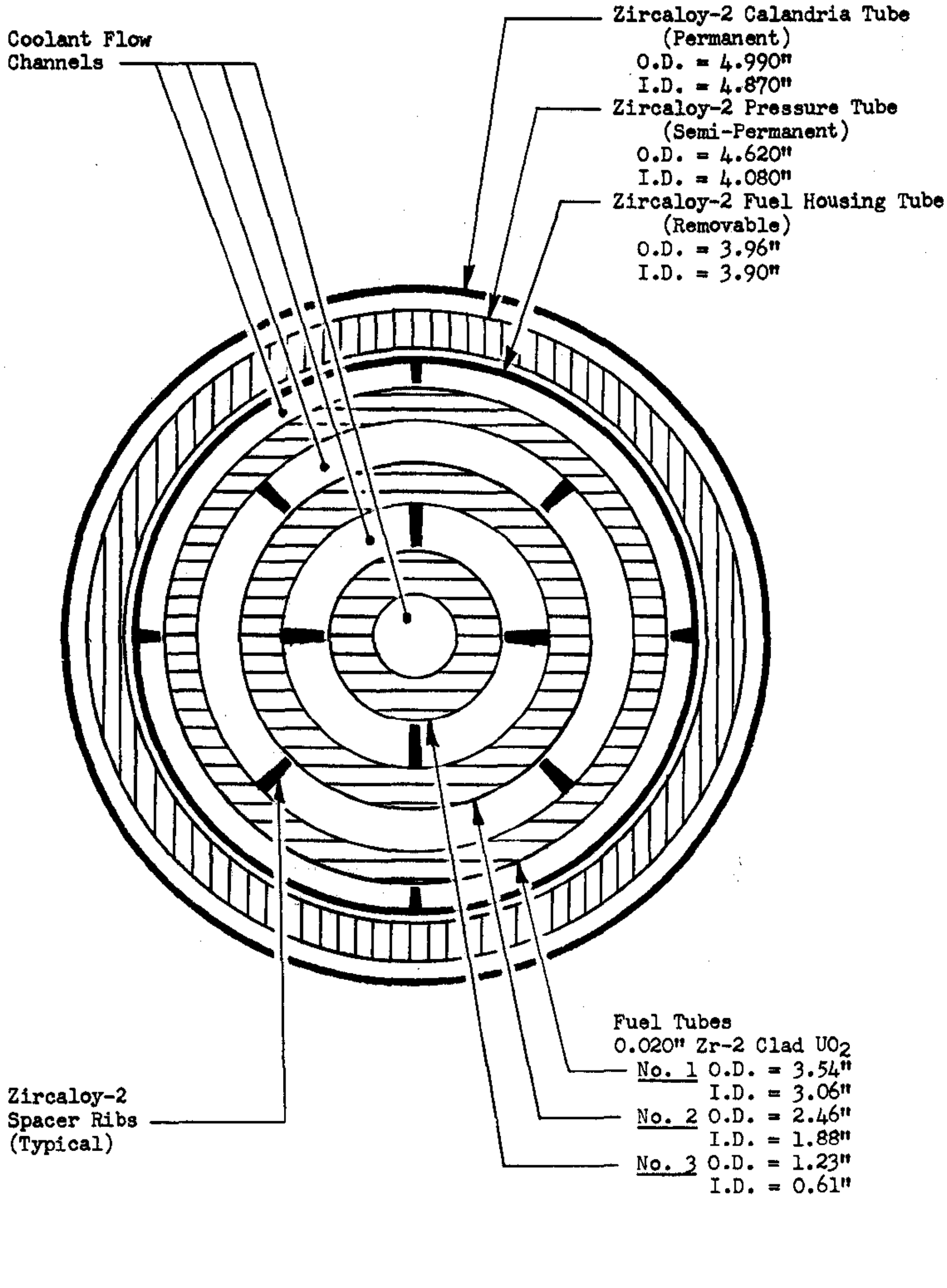
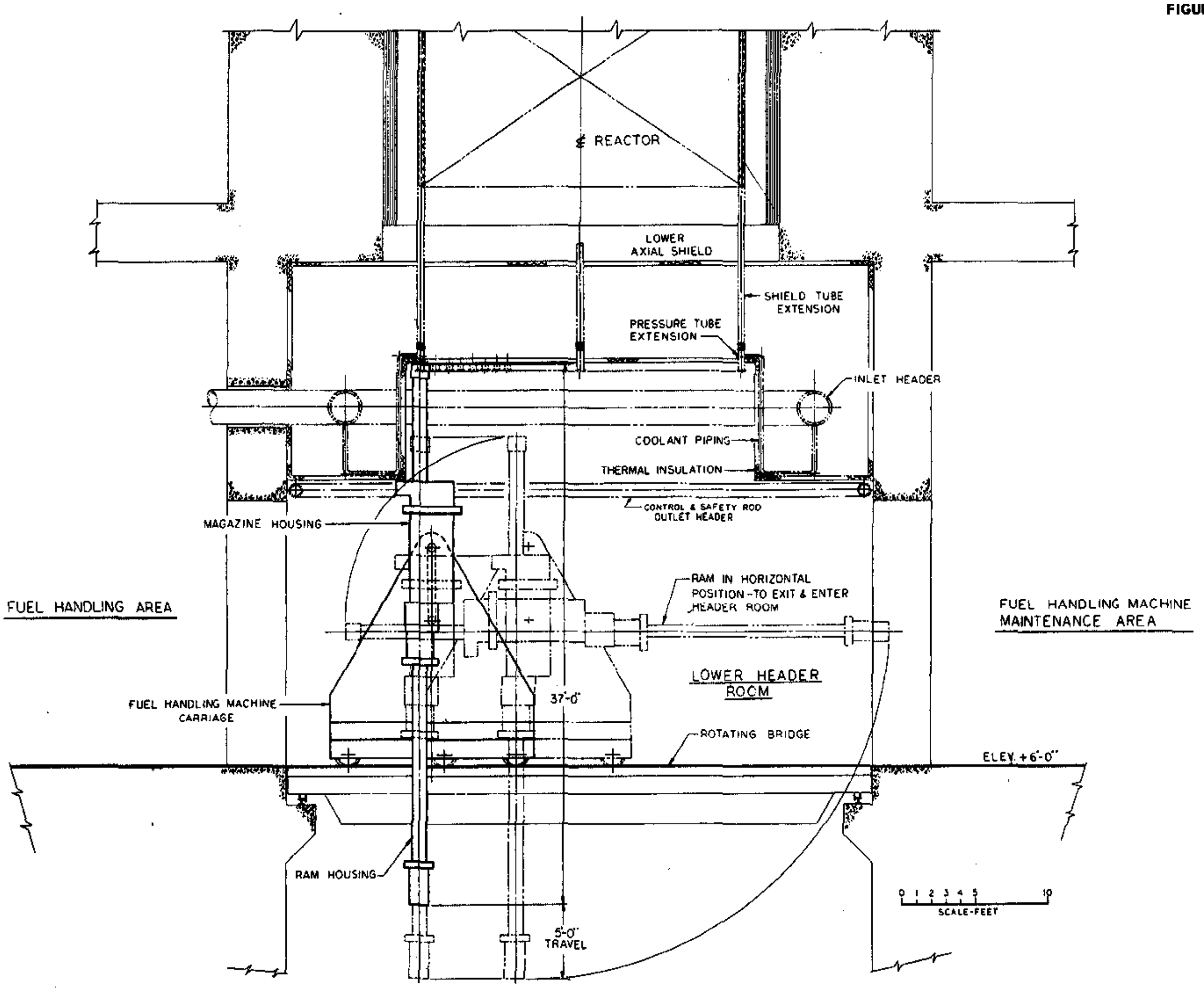


FIGURE 6



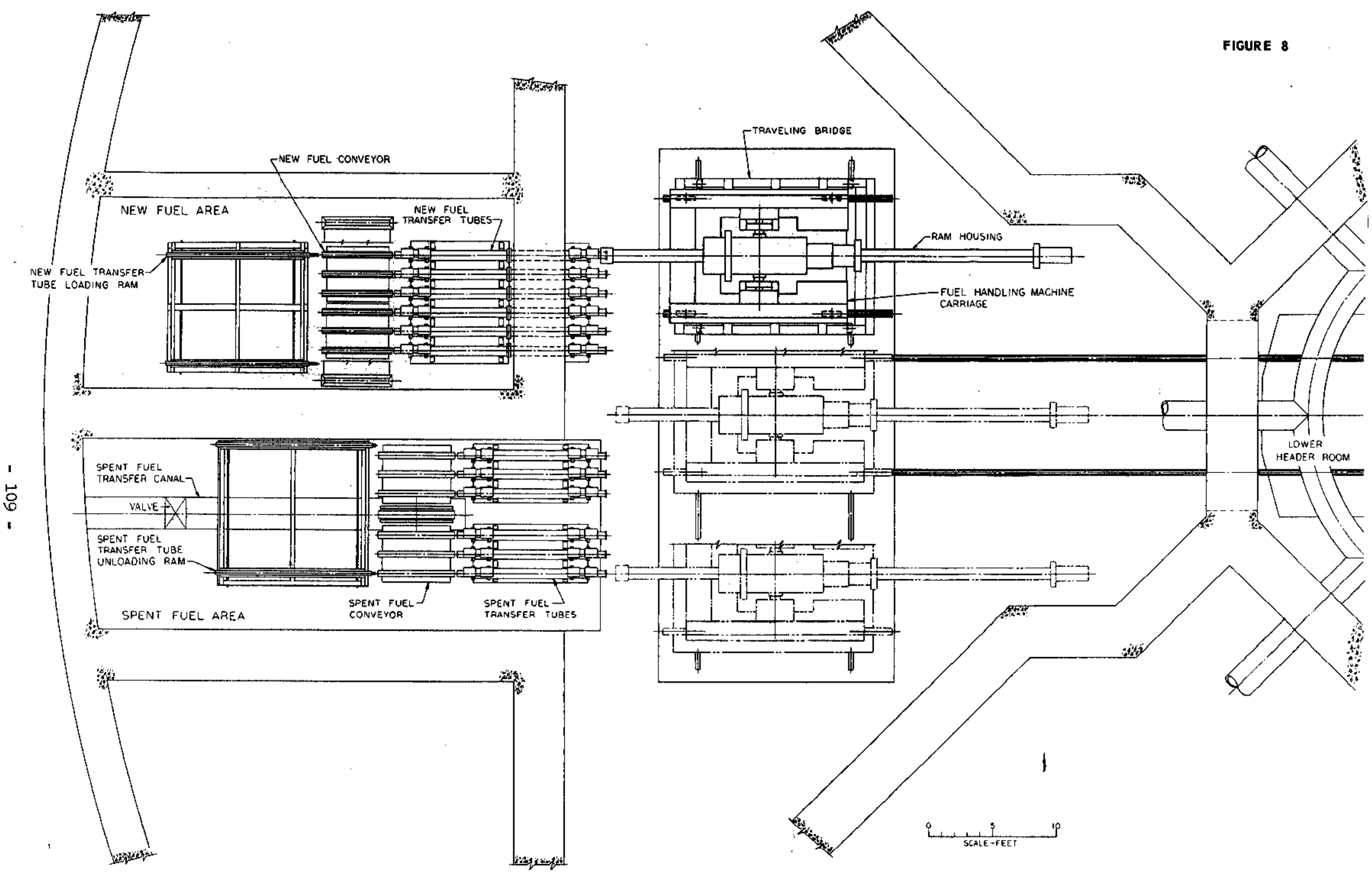
3500 MWT D_2O COOLED REACTOR
FUEL CROSS SECTION

FIGURE 7



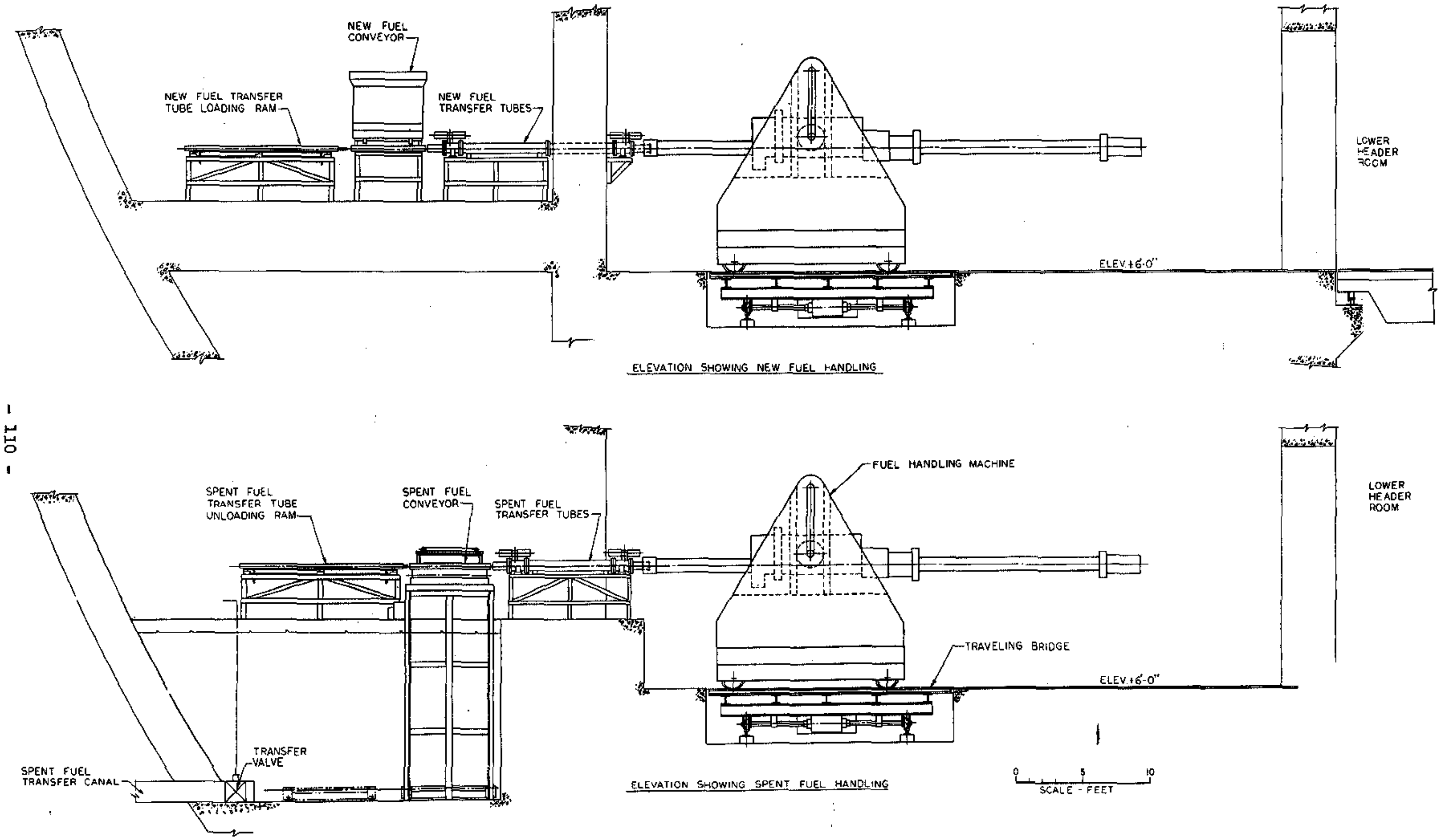
3500 MWT D_2O COOLED REACTOR
FUEL HANDLING MACHINE

FIGURE 8



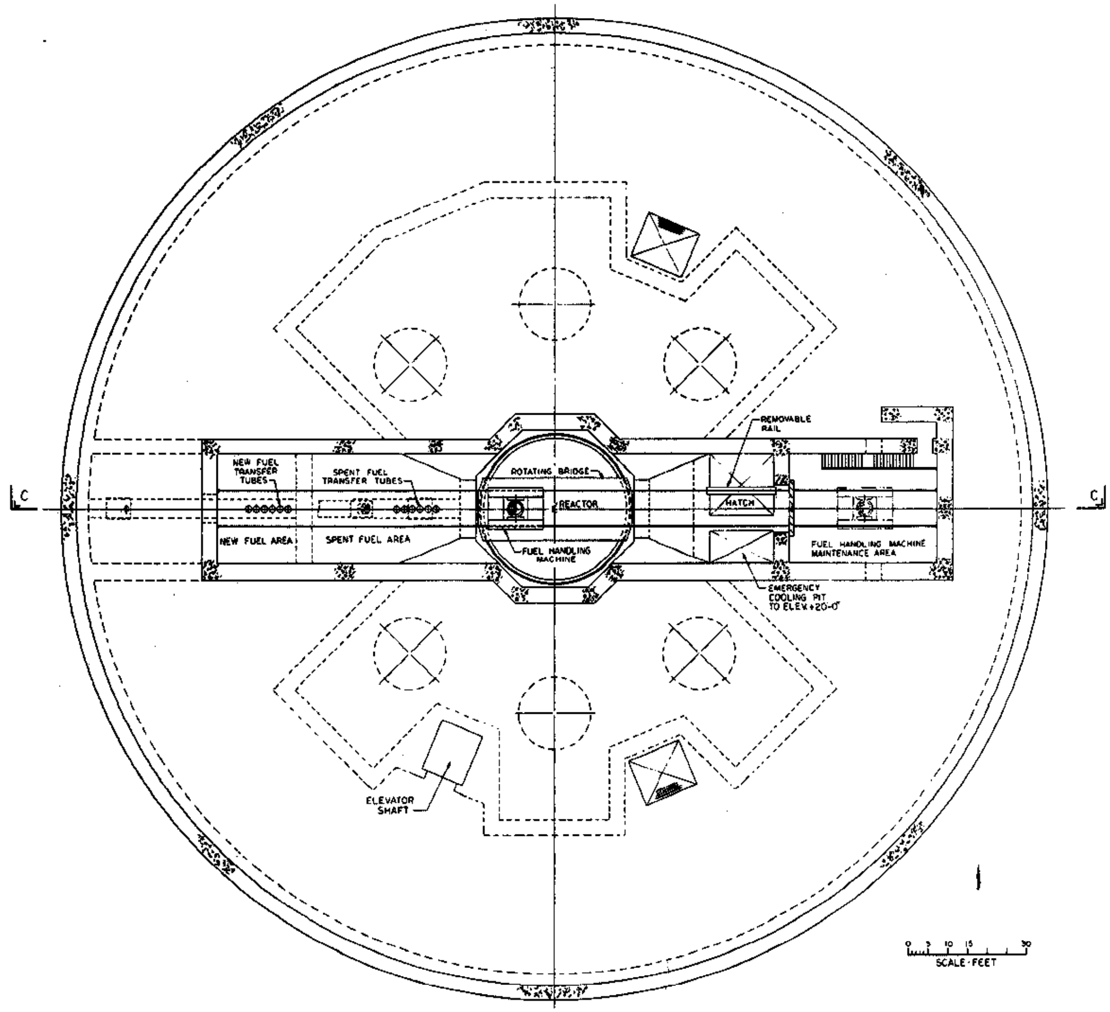
3500 MWT D_2O COOLED REACTOR
FUEL HANDLING AREA
ARRANGEMENT - PLAN

FIGURE 9



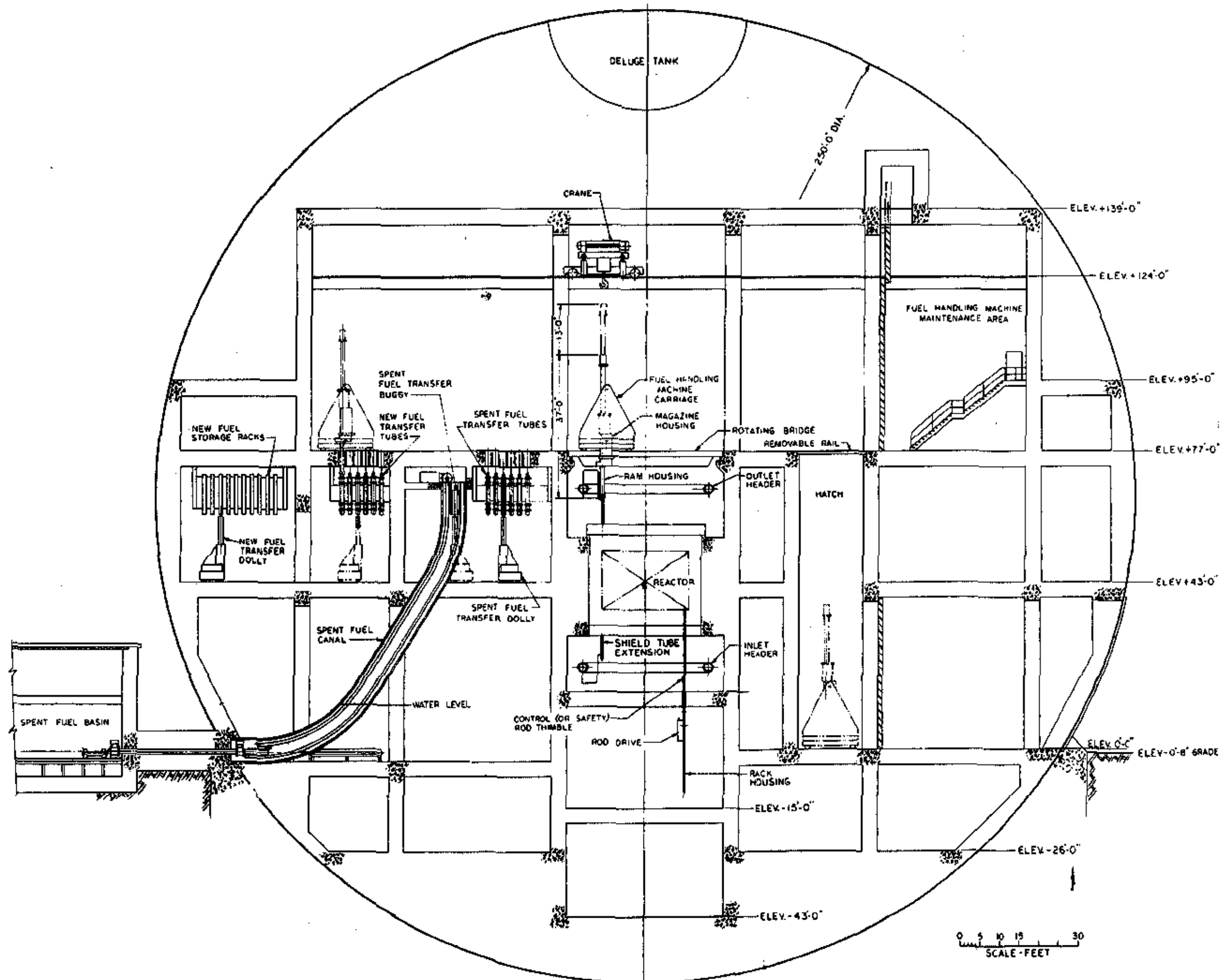
3500 MWT D_2O COOLED REACTOR
FUEL HANDLING AREA
ARRANGEMENT – ELEVATIONS

FIGURE 10



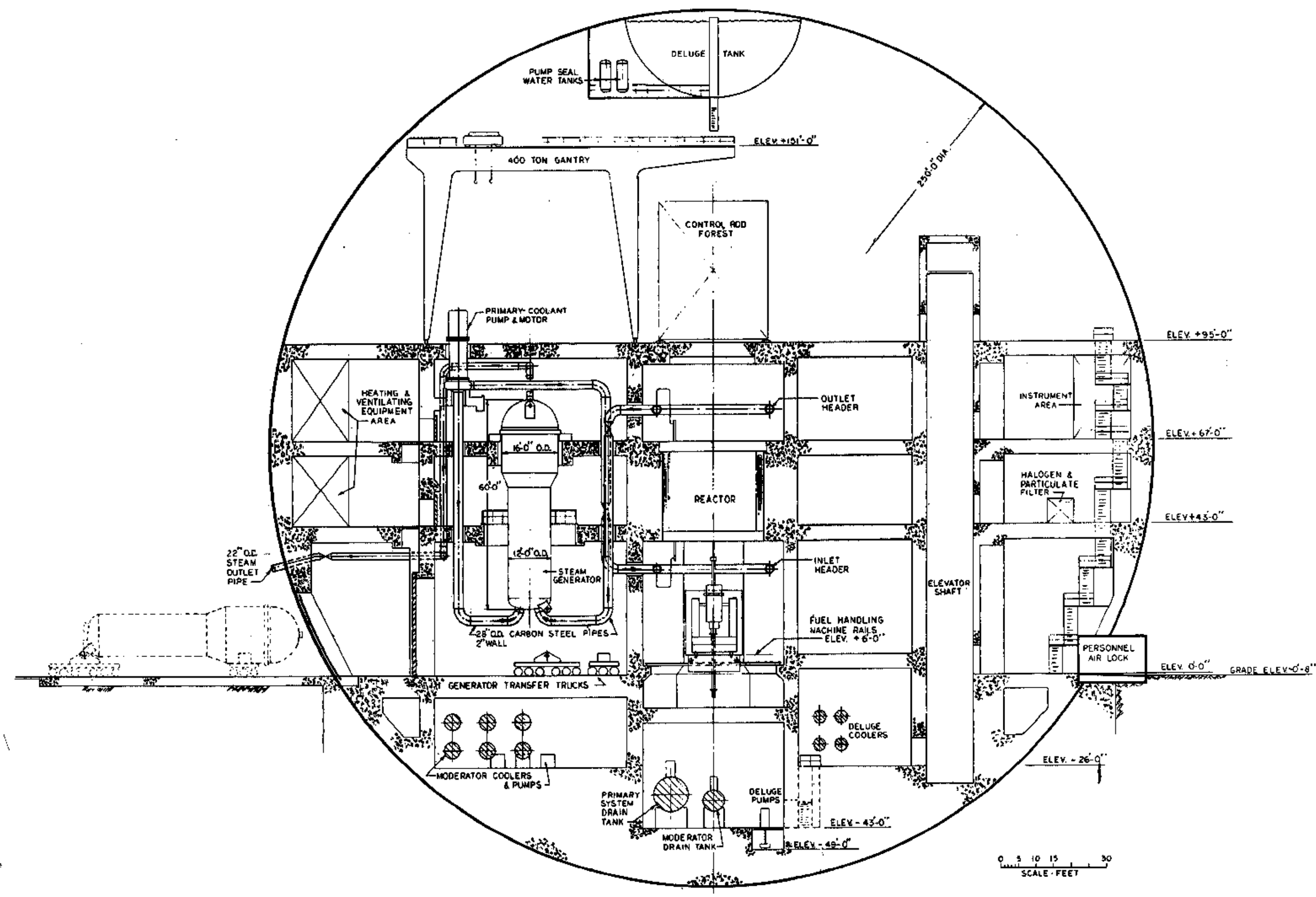
3500 MWT D_2O COOLED REACTOR
TOP FUEL HANDLING AREA
ARRANGEMENT - PLAN

FIGURE 11



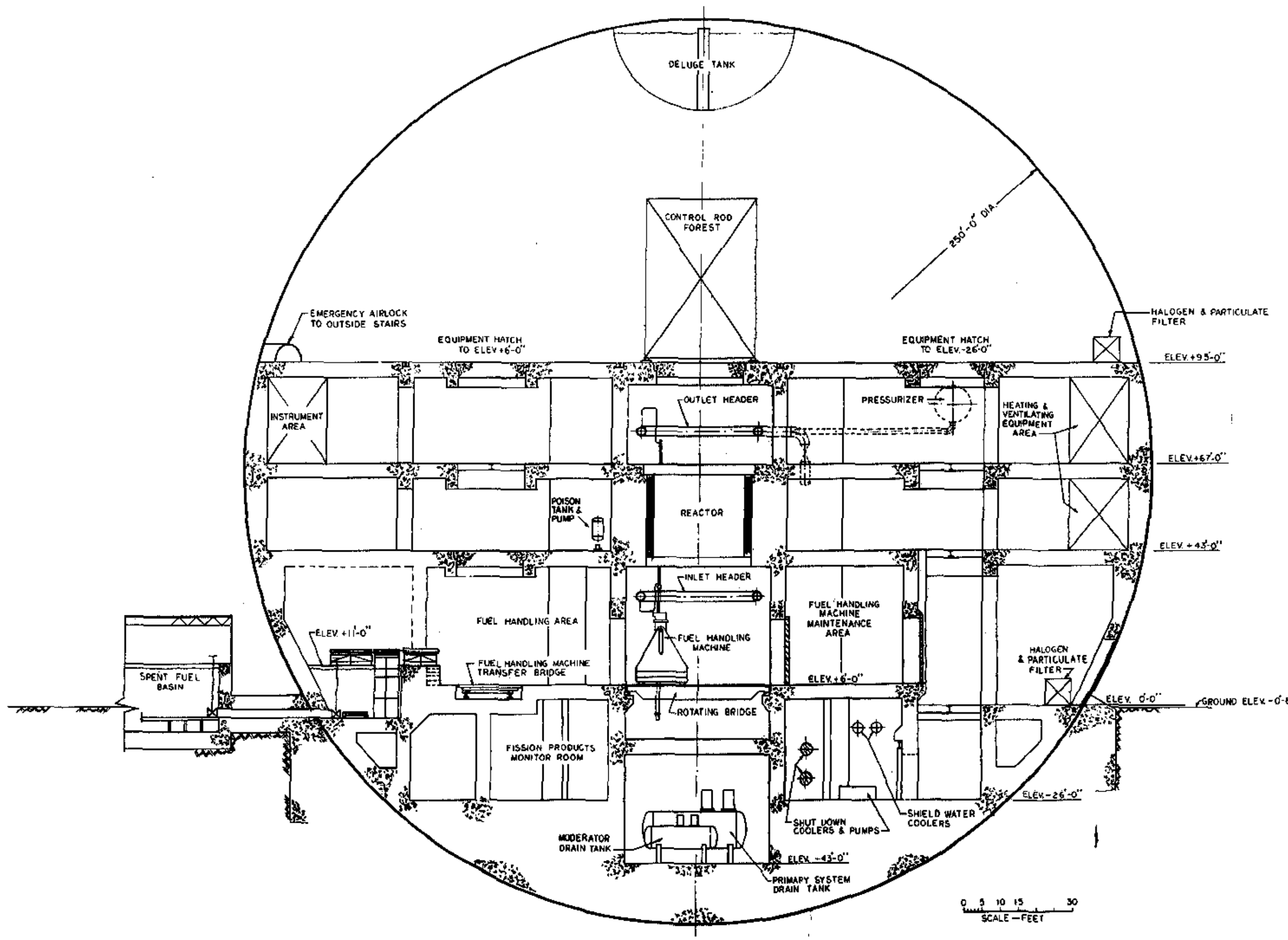
3500 MWT D_2O COOLED REACTOR
TOP FUEL HANDLING AREA
ARRANGEMENT – ELEVATION

FIGURE 12



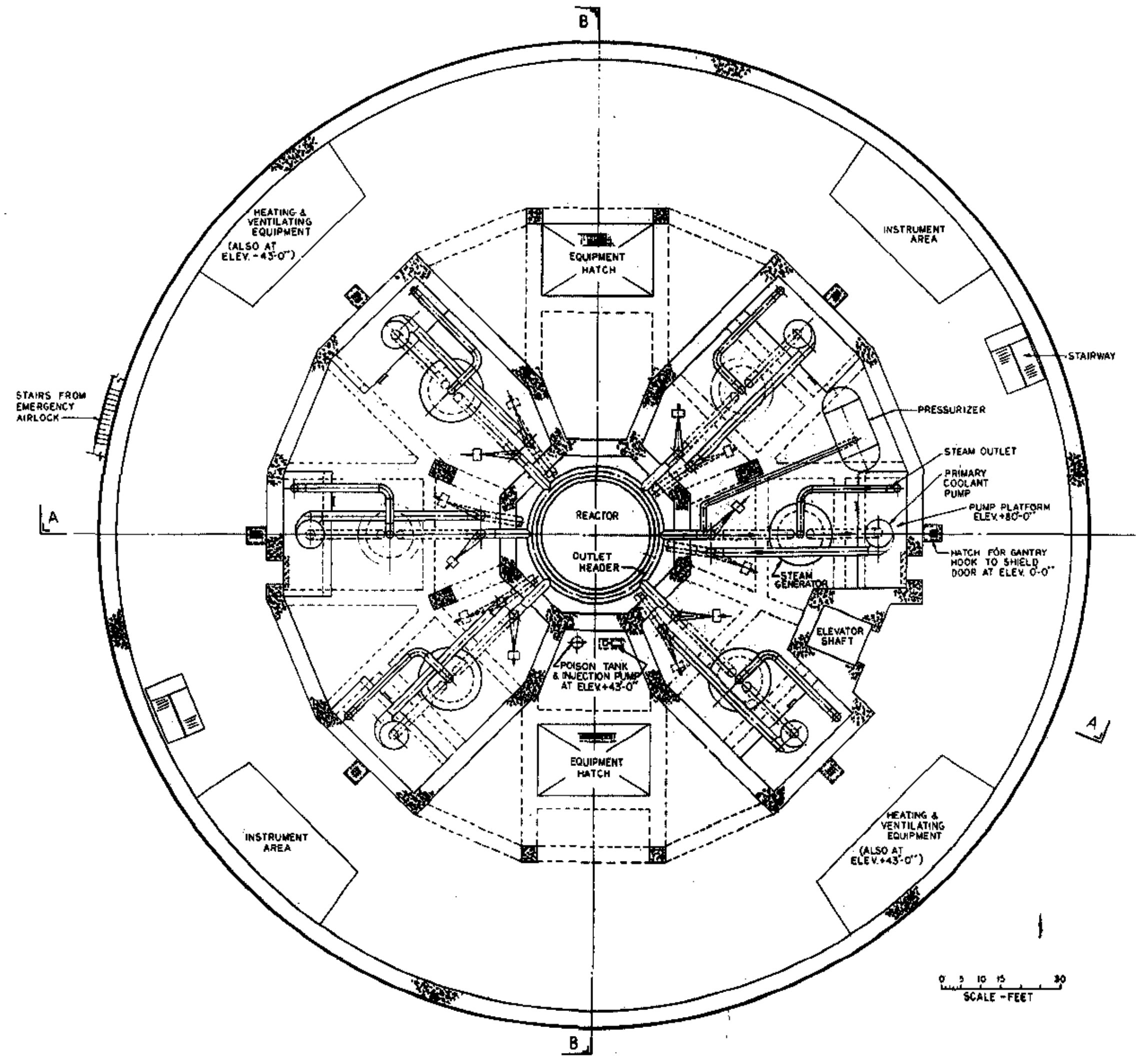
3500 MWT D_2O COOLED REACTOR
REACTOR BUILDING ARRANGEMENT
SECTION A–A

FIGURE 13



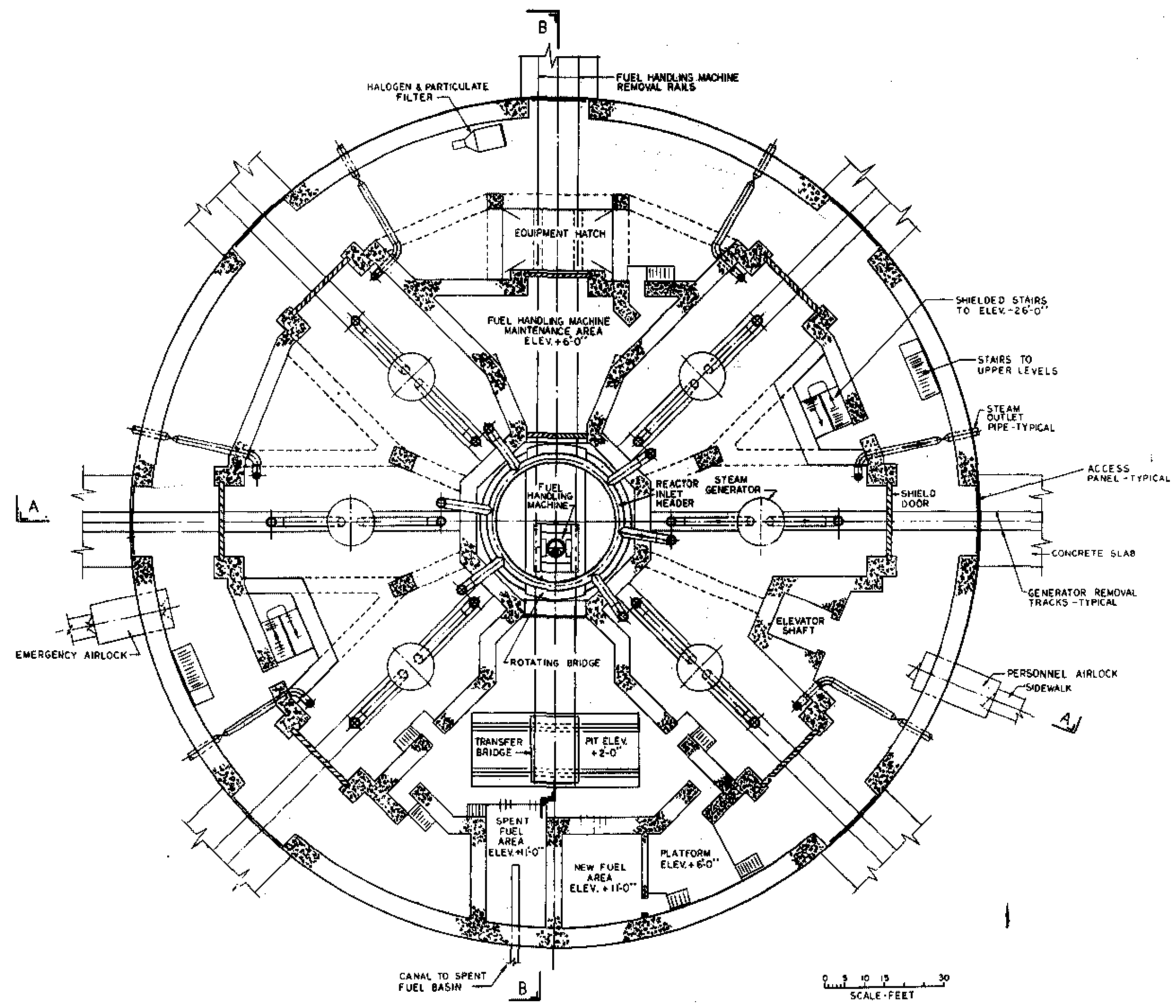
3500 MWT D_2O COOLED REACTOR
REACTOR BUILDING ARRANGEMENT
SECTION B–B

FIGURE 14



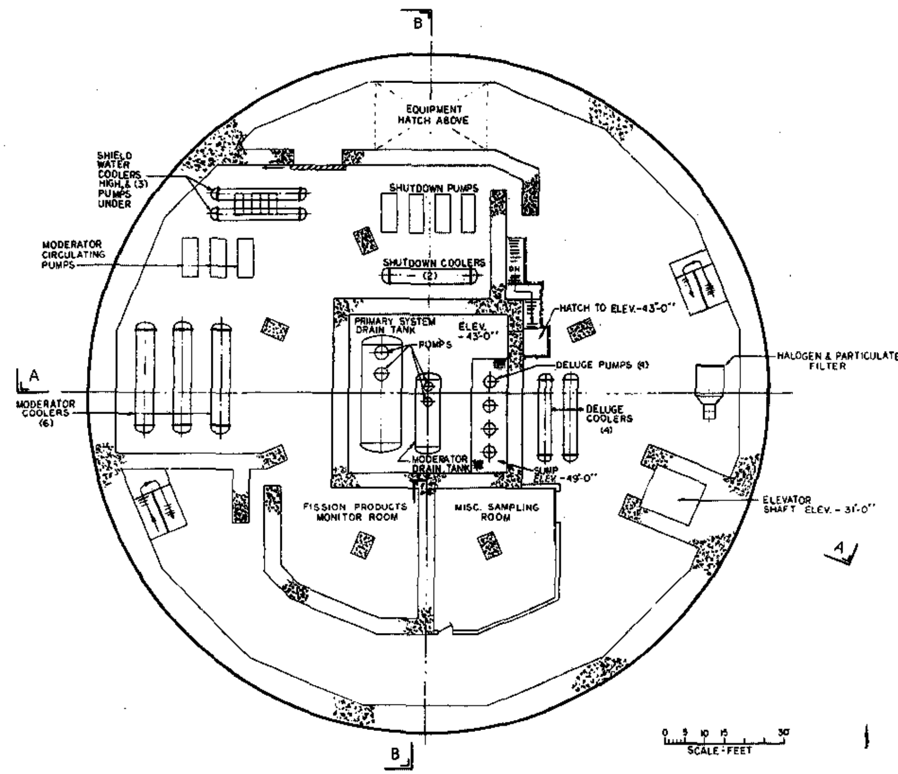
3500 MWT D_2O COOLED REACTOR REACTOR BUILDING ARRANGEMENT PLAN AT ELEVATION +67'-0"

FIGURE 15



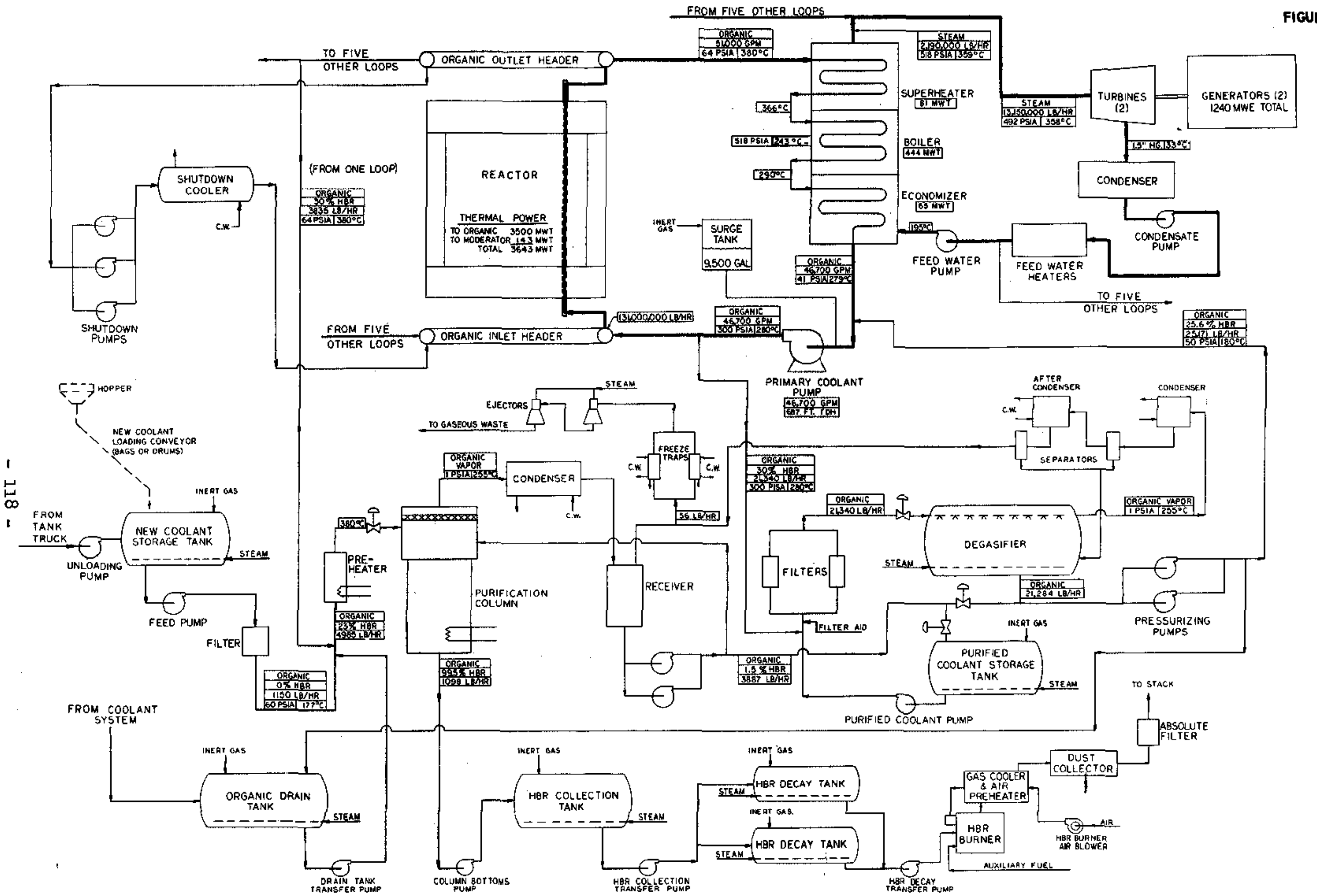
3500 MWT D_2O COOLED REACTOR
REACTOR BUILDING ARRANGEMENT
PLAN AT ELEVATION 0' – 0"

FIGURE 16



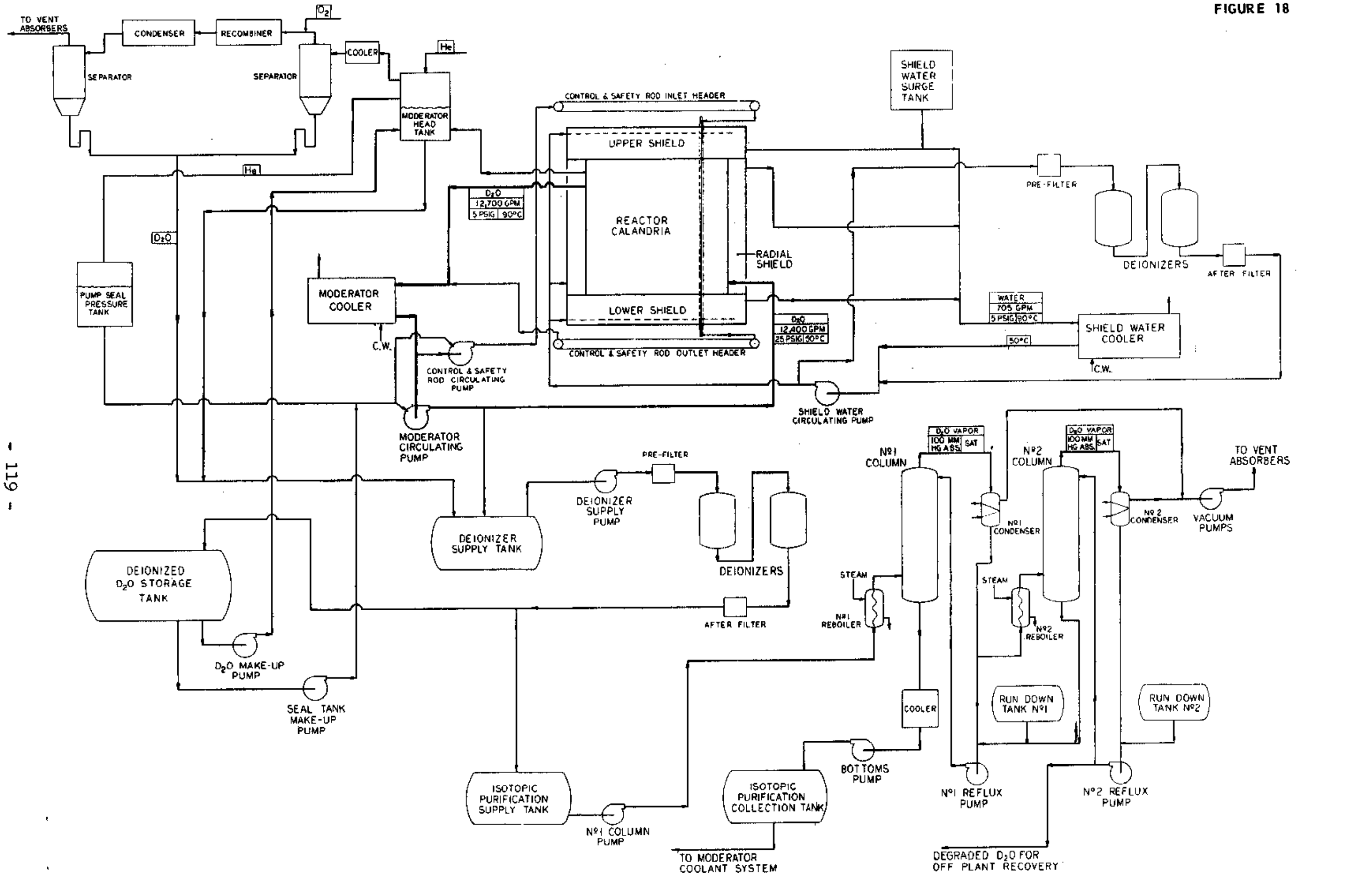
3500 MWT D_2O COOLED REACTOR
REACTOR BUILDING ARRANGEMENT
PLAN AT ELEVATION -26'-0"

FIGURE 17



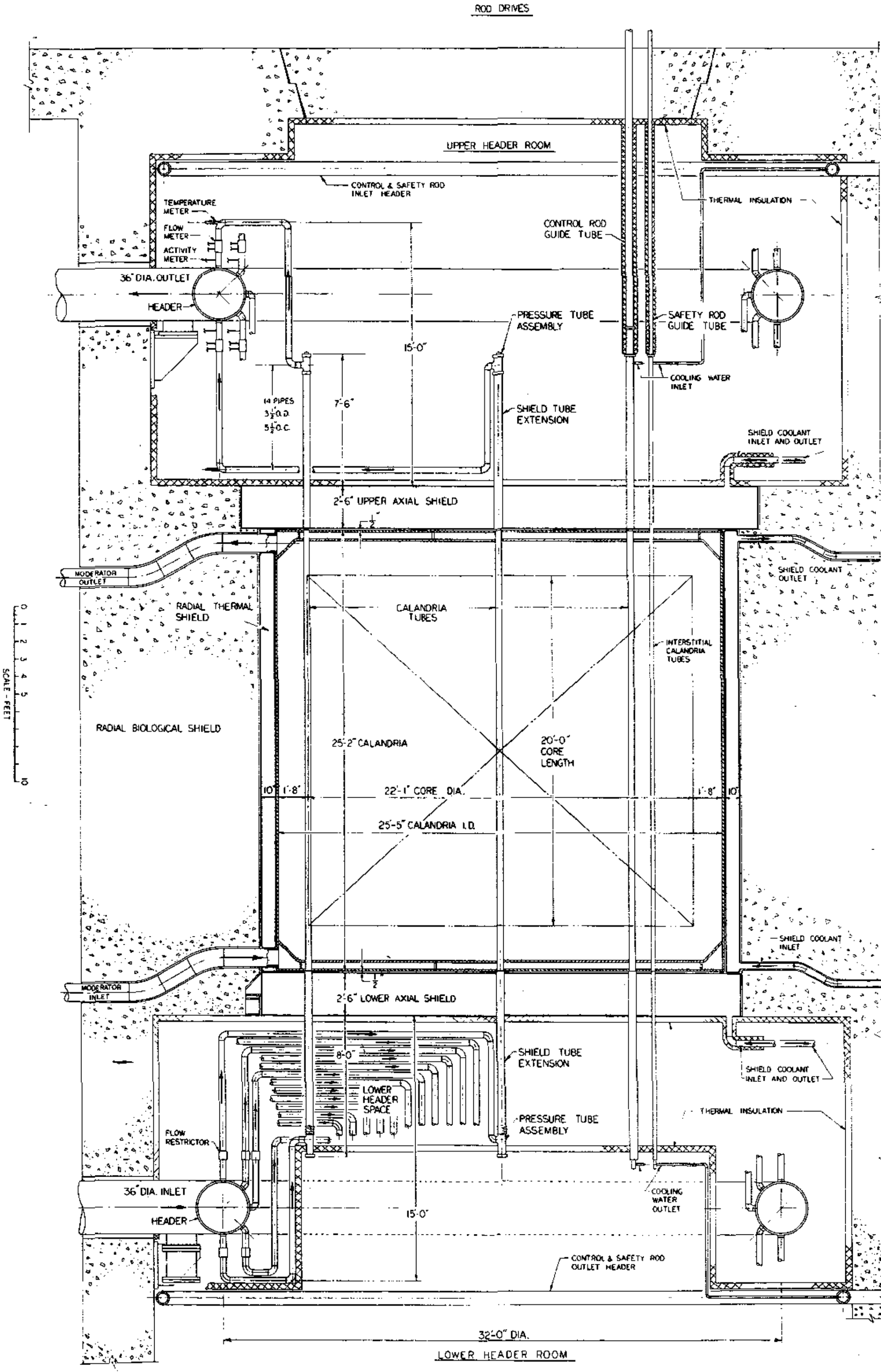
3500 MWT ORGANIC COOLED REACTOR
FLOW DIAGRAM
ORGANIC COOLANT SYSTEM

FIGURE 18



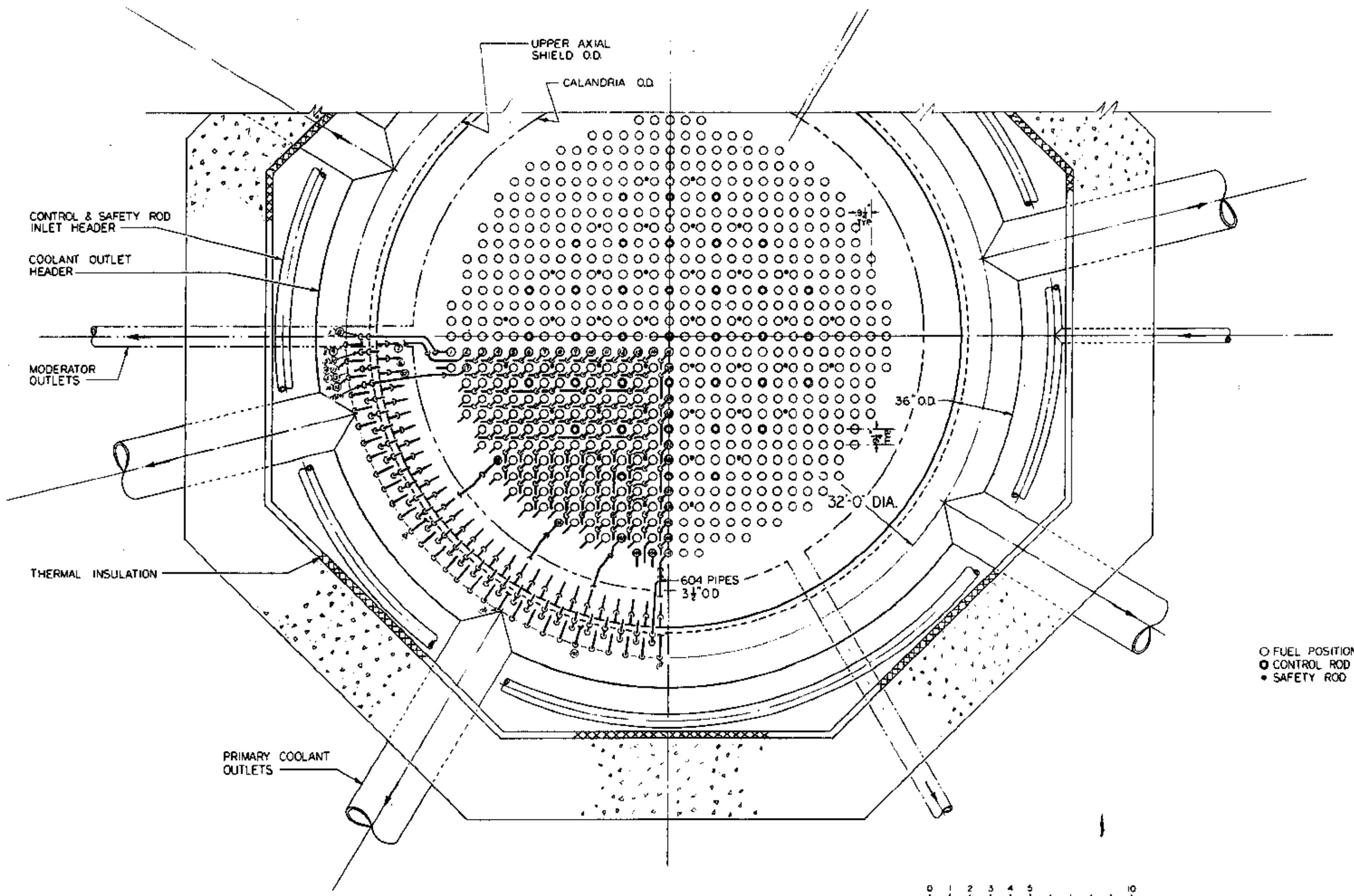
3500 MWT ORGANIC COOLED REACTOR
FLOW DIAGRAM
MODERATOR & SHIELD COOLING SYSTEMS

FIGURE 19



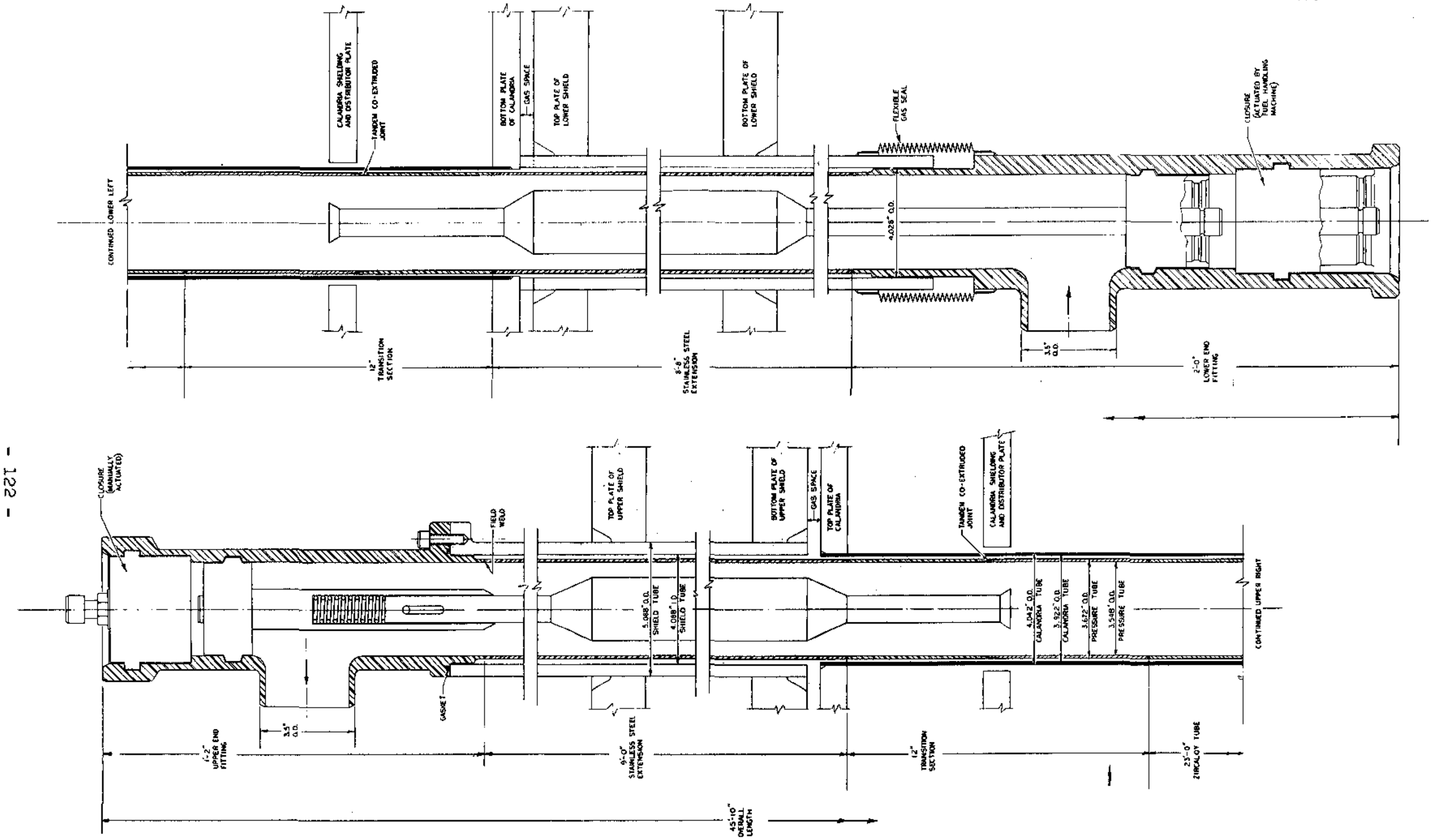
3500 MWT ORGANIC COOLED REACTOR
REACTOR ARRANGEMENT
CROSS SECTION

FIGURE 20



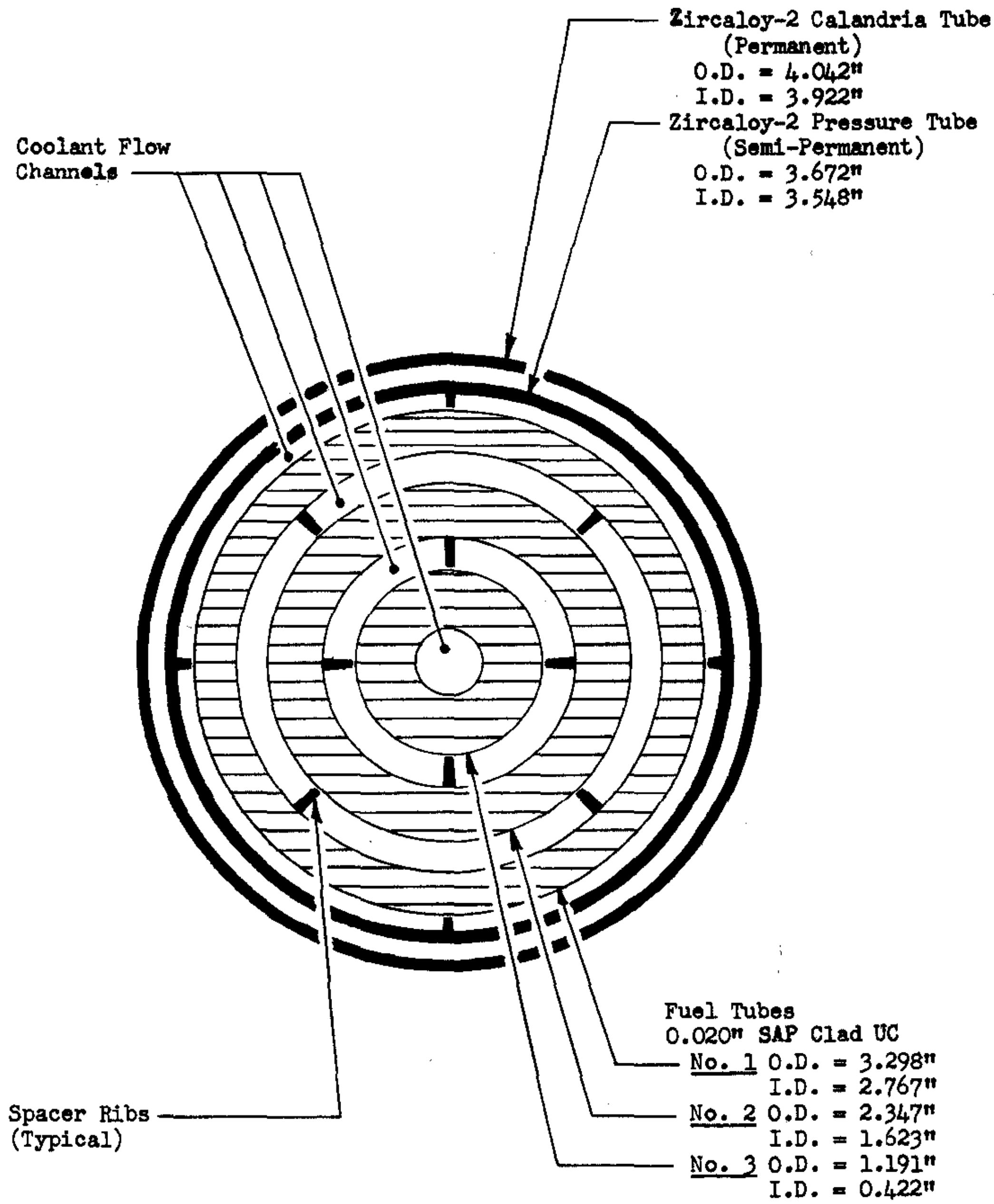
3500 MWT ORGANIC COOLED REACTOR
REACTOR ARRANGEMENT
PLAN

FIGURE 21



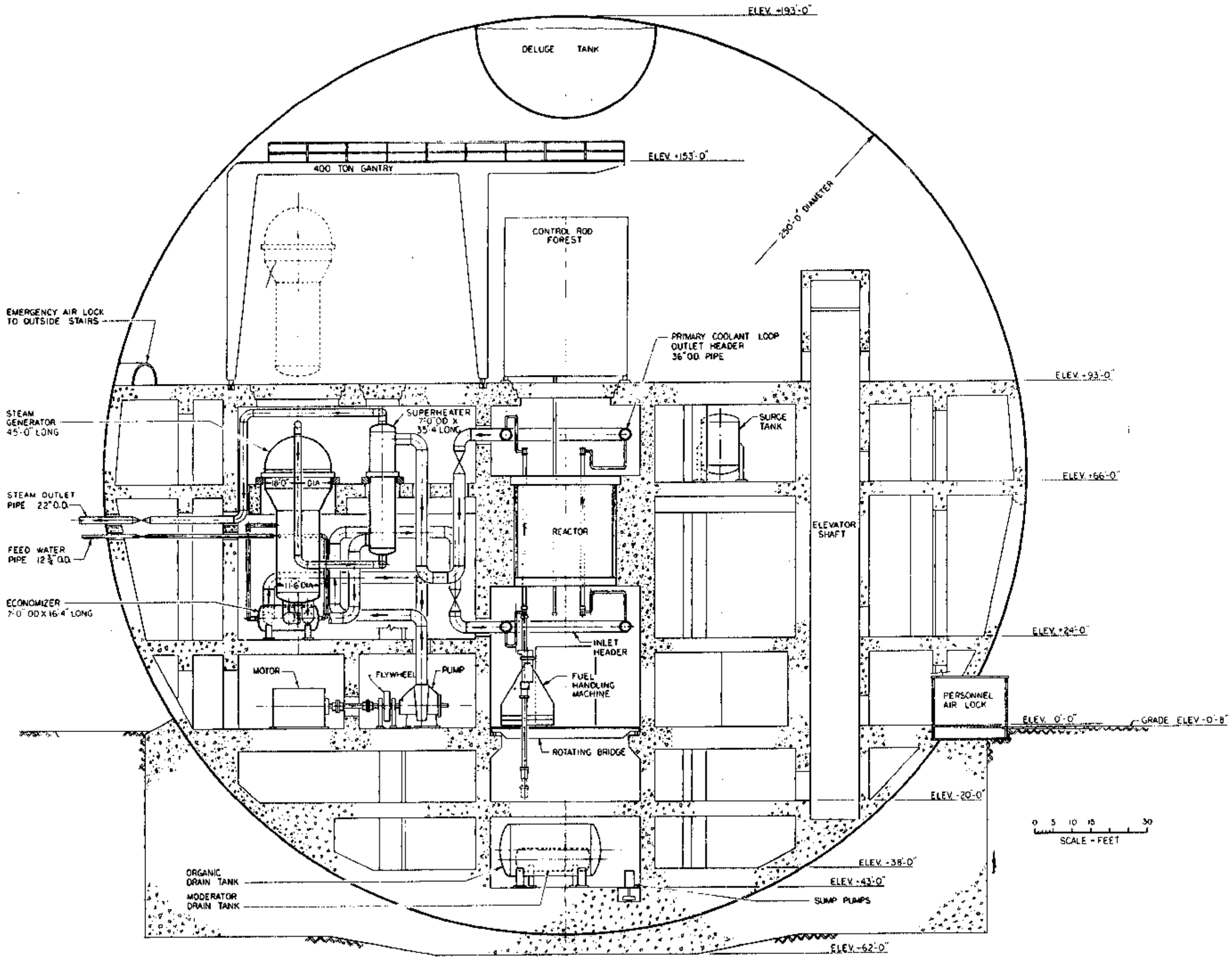
3500 MWT ORGANIC COOLED REACTOR
PRESSURE TUBE ASSEMBLY

FIGURE 22



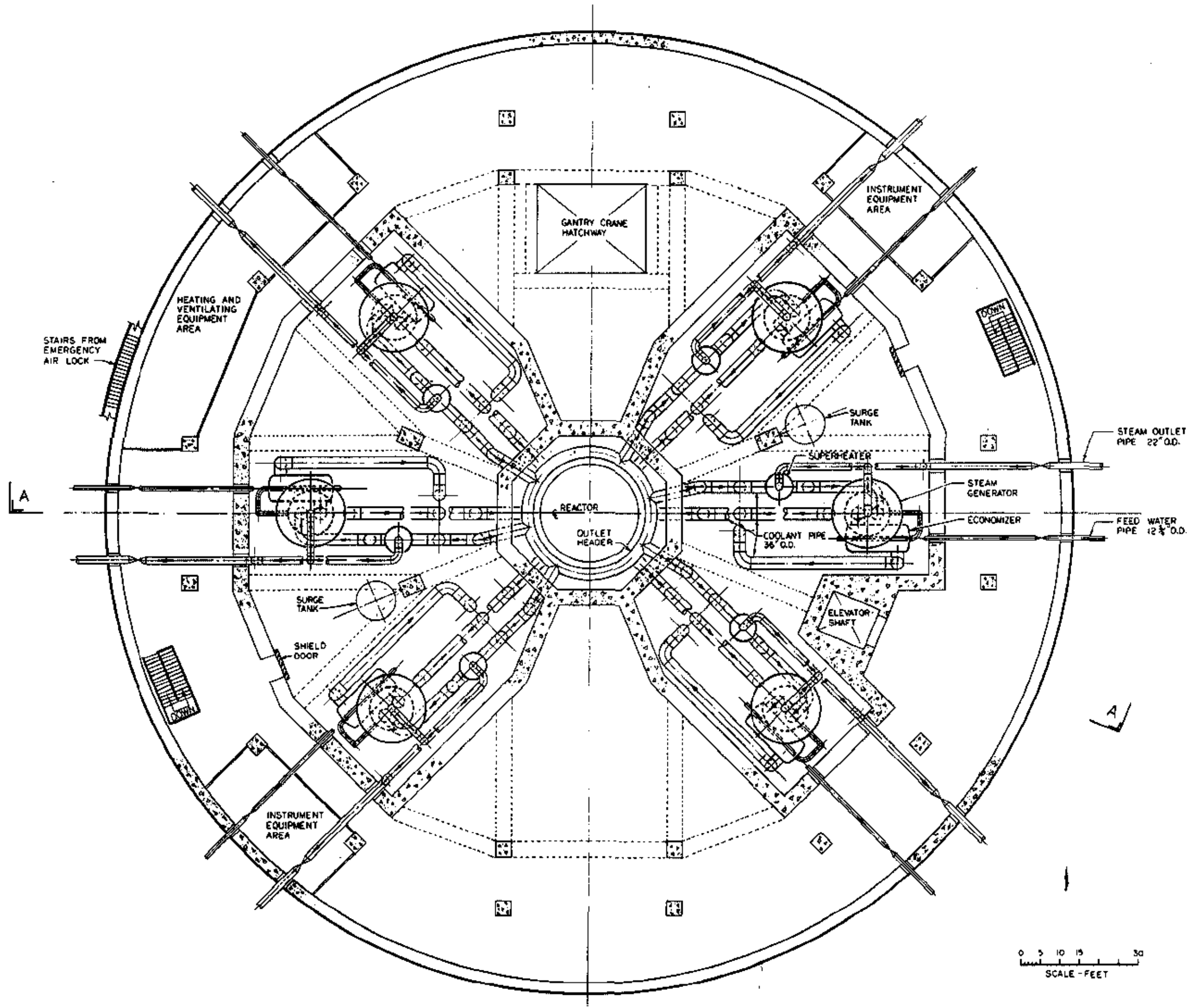
8300 & 3500 MWT ORGANIC COOLED REACTORS
FUEL CROSS SECTION

FIGURE 23

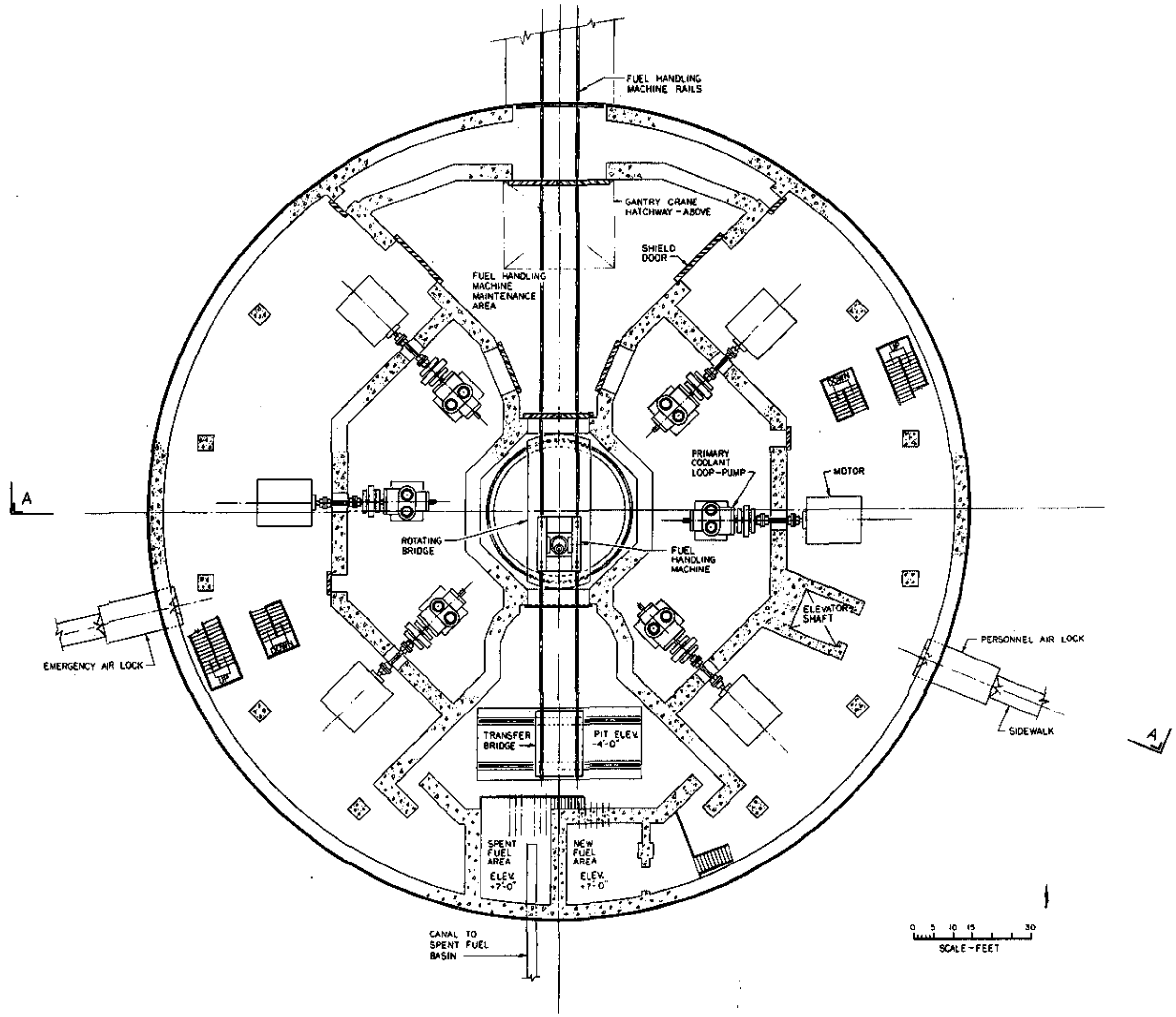


3500 MWT ORGANIC COOLED REACTOR
REACTOR BUILDING ARRANGEMENT
SECTION A–A

FIGURE 24

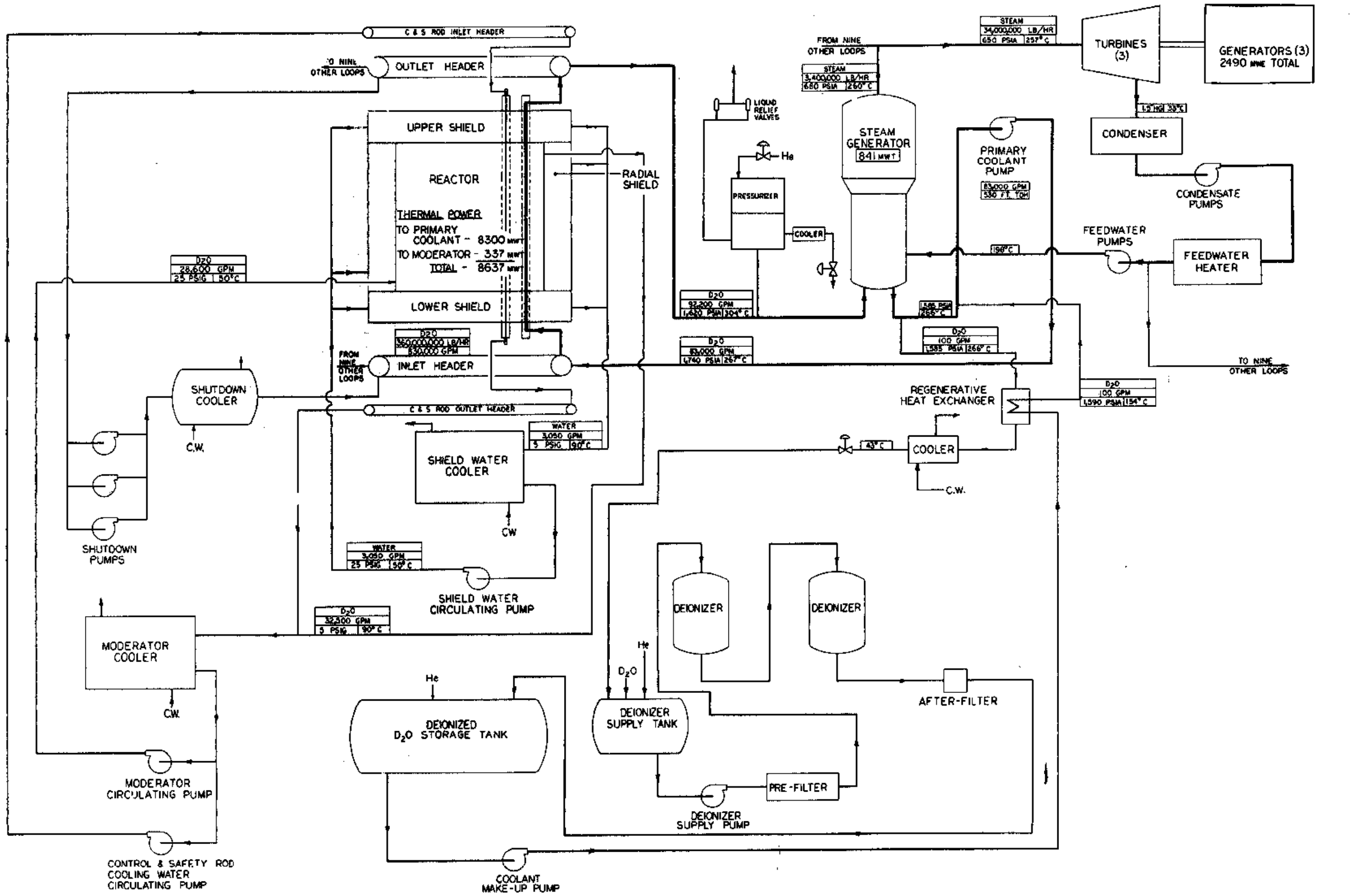


3500 MWT ORGANIC COOLED REACTOR
REACTOR BUILDING ARRANGEMENT
PLAN AT ELEVATION + 66' – 0"



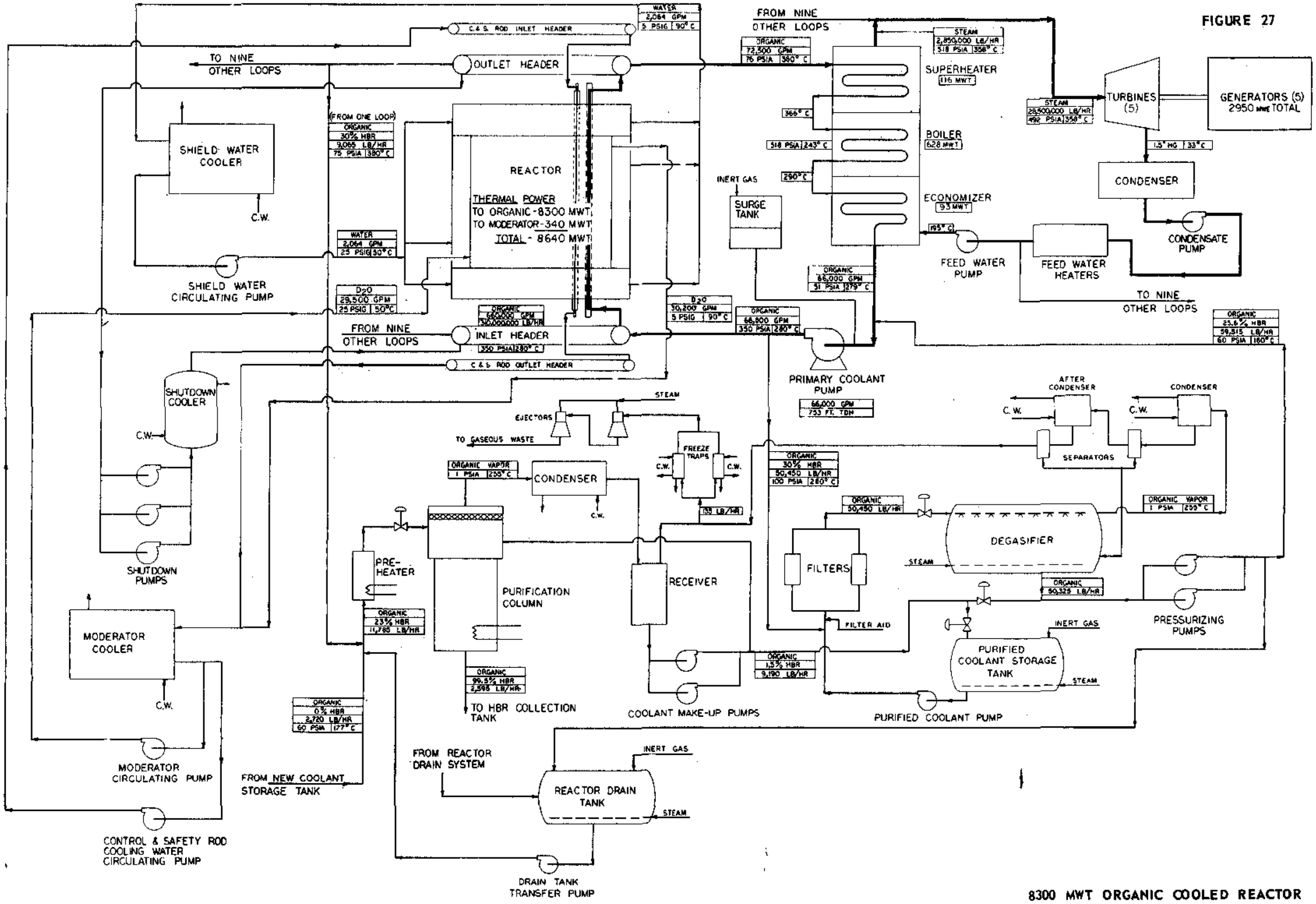
3500 MWT ORGANIC COOLED REACTOR
REACTOR BUILDING ARRANGEMENT
PLAN AT ELEVATION 0' - 0"

FIGURE 26



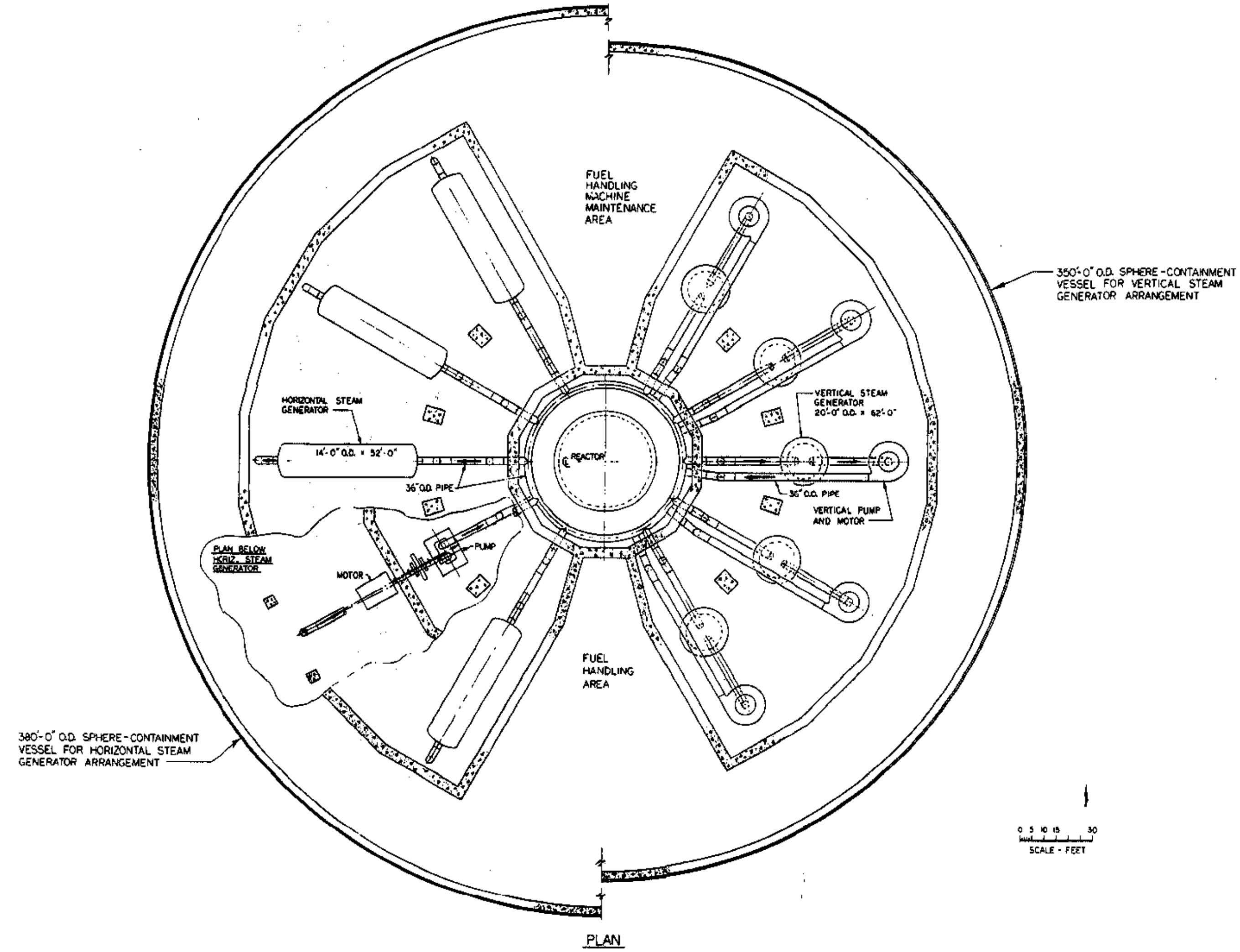
8300 MWT D_2O COOLED REACTOR FLOW DIAGRAM REACTOR COOLING SYSTEM

FIGURE 27



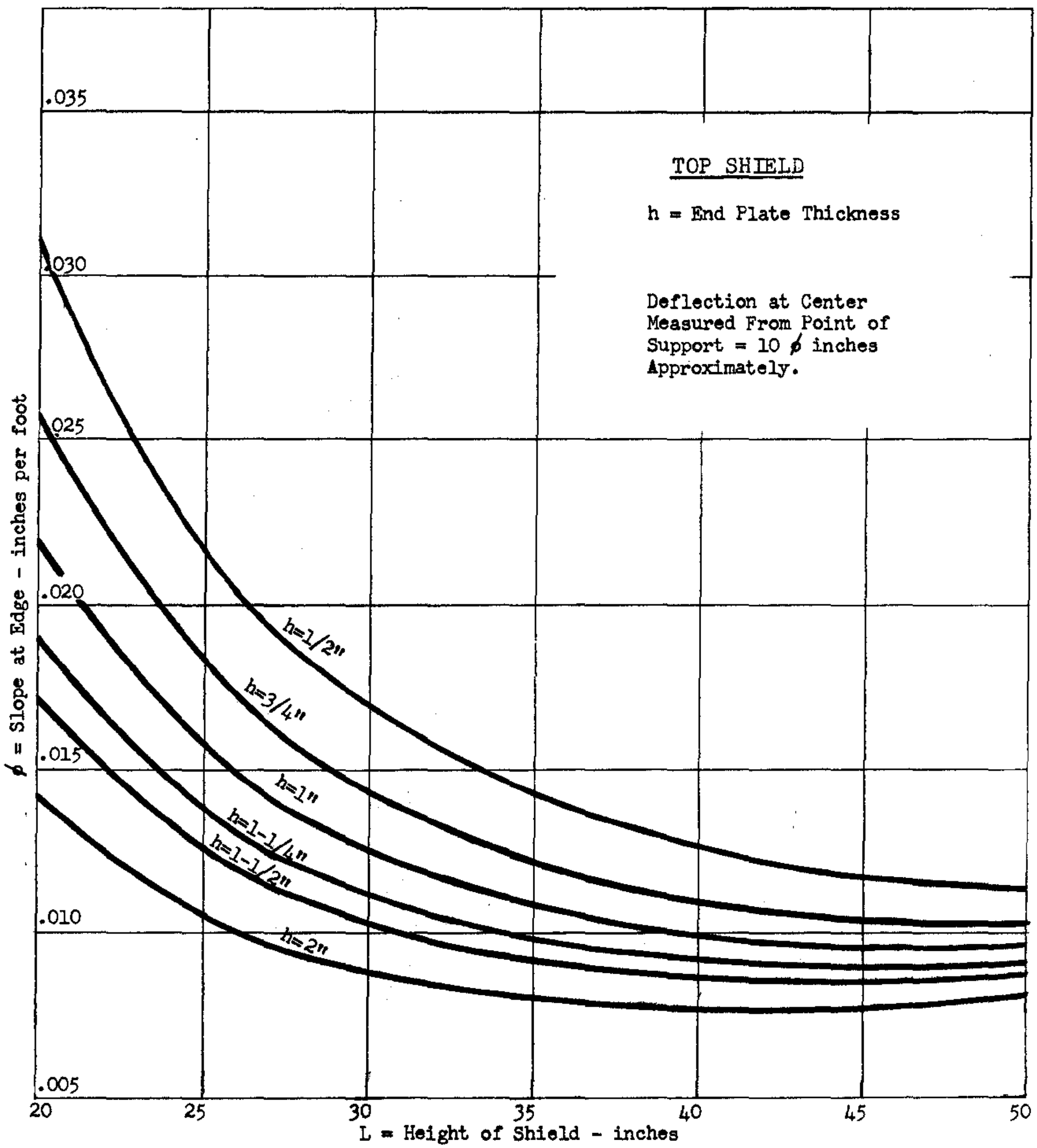
8300 MWT ORGANIC COOLED REACTOR
FLOW DIAGRAM
REACTOR COOLING SYSTEM

FIGURE 28



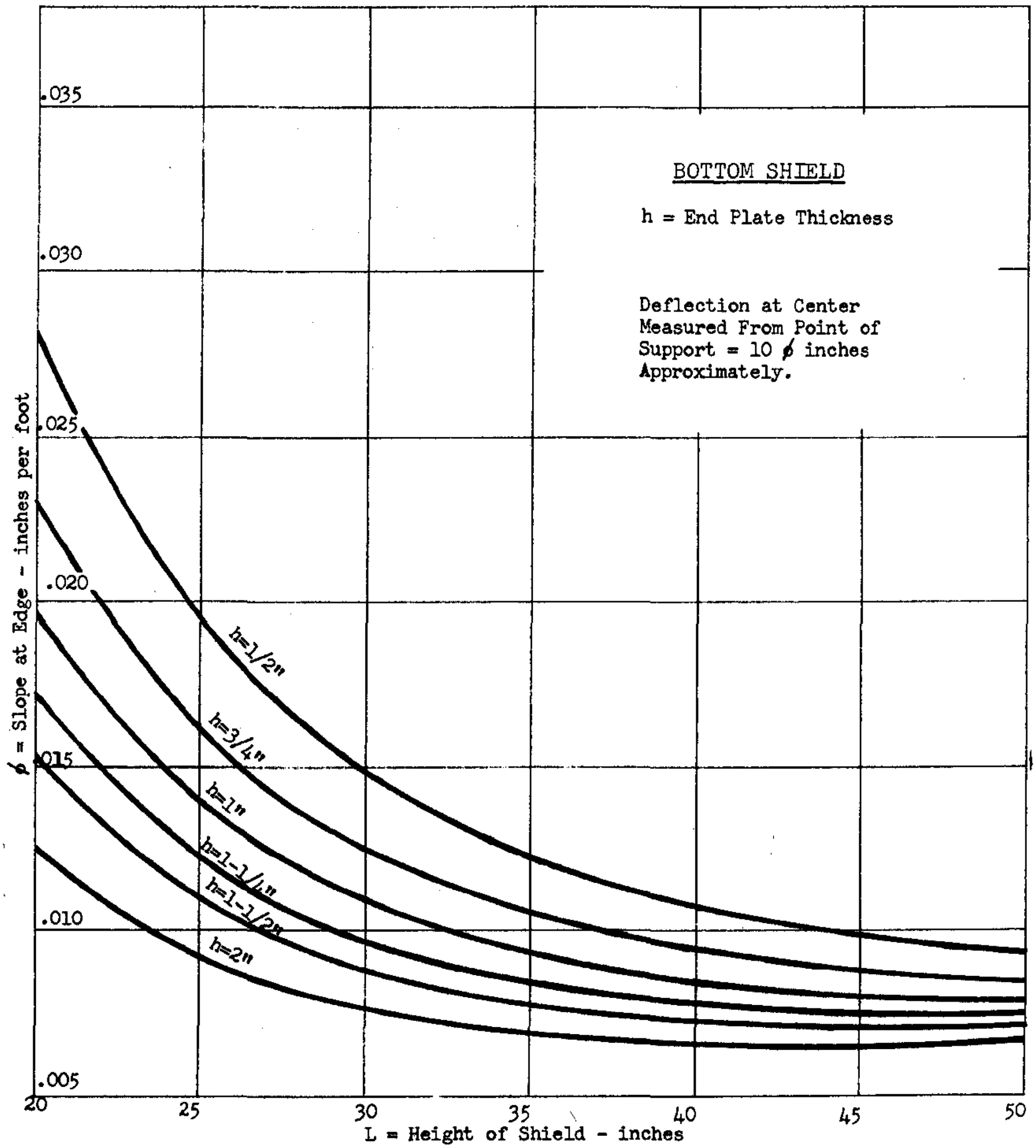
8300 MWT D_2O COOLED REACTOR
REACTOR BUILDING ARRANGEMENTS
PLAN

FIGURE 29

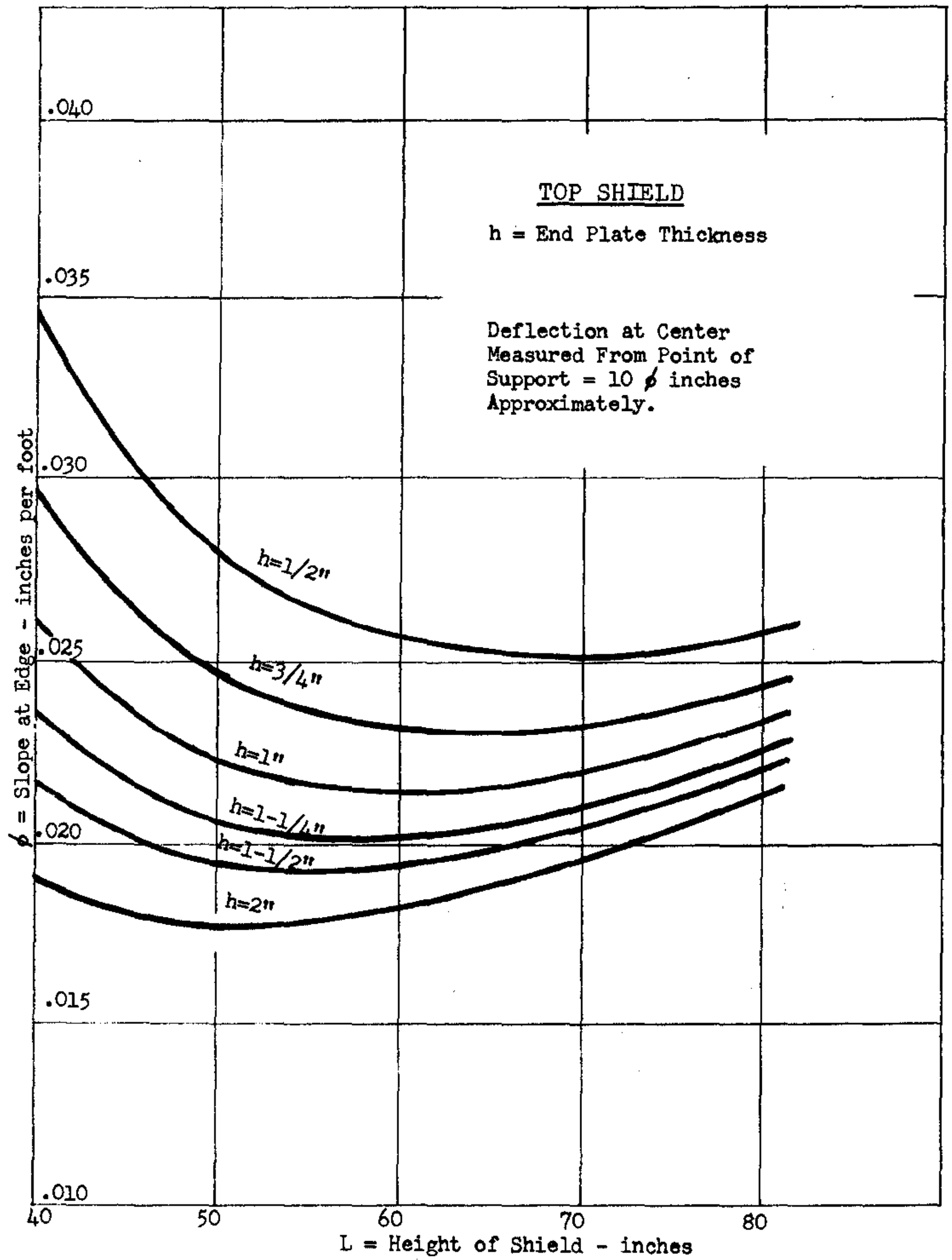


3500 MWT D2O COOLED REACTOR
TOP AXIAL SHIELD DEFORMATION

FIGURE 30

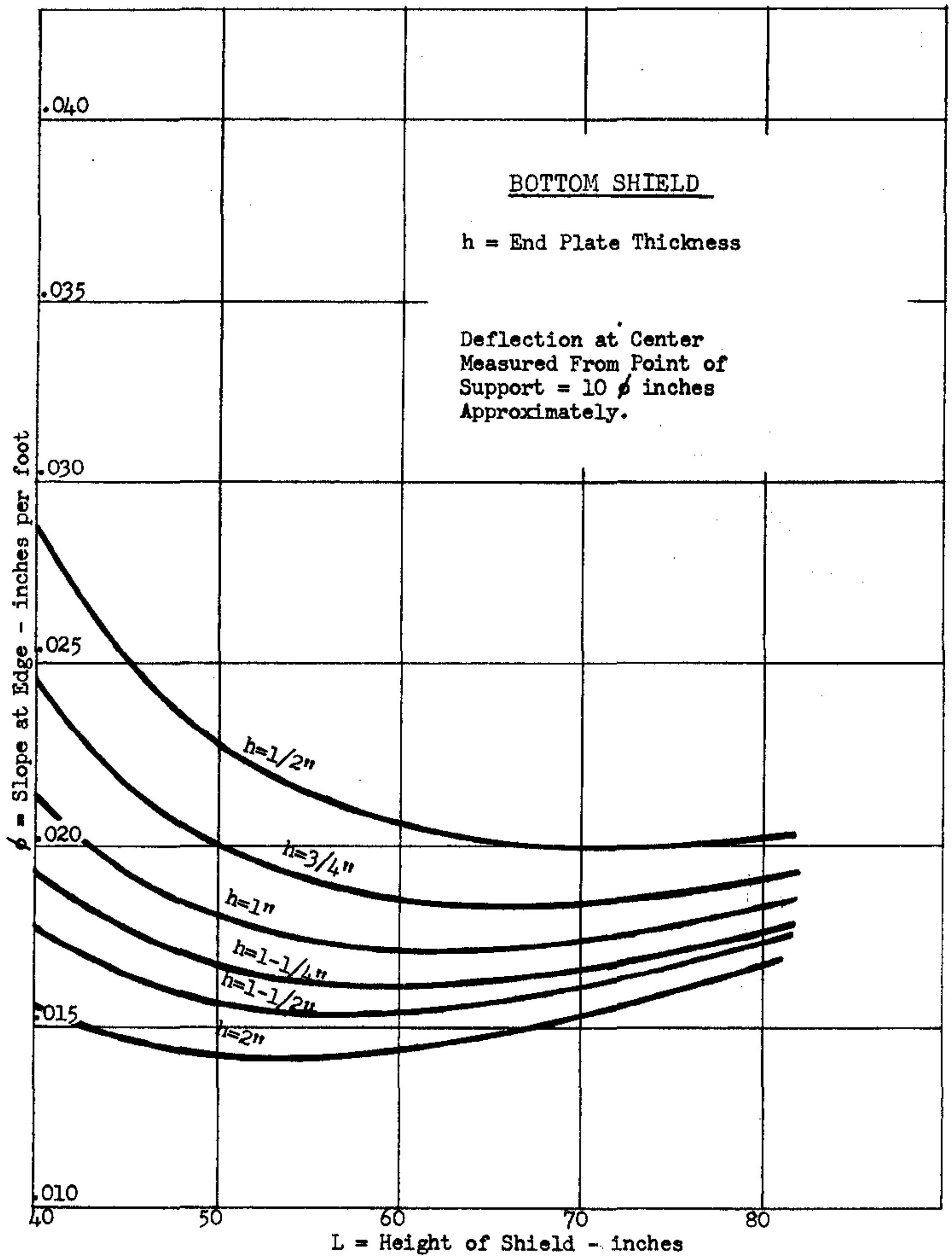


3500 MWT D_2O COOLED REACTOR
BOTTOM AXIAL SHIELD DEFORMATION



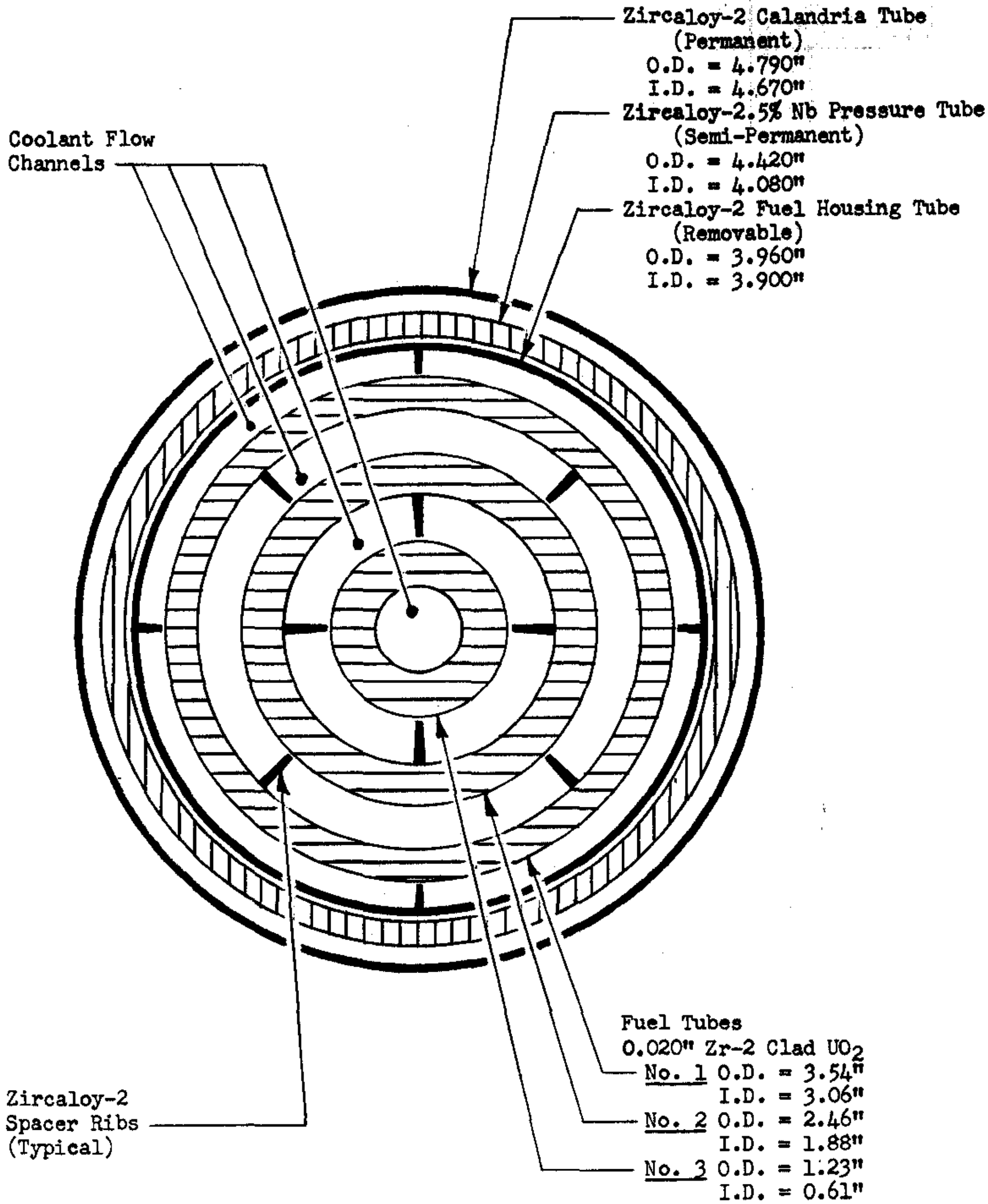
8300 MWT D_2O COOLED REACTOR
TOP AXIAL SHIELD DEFORMATION

FIGURE 32



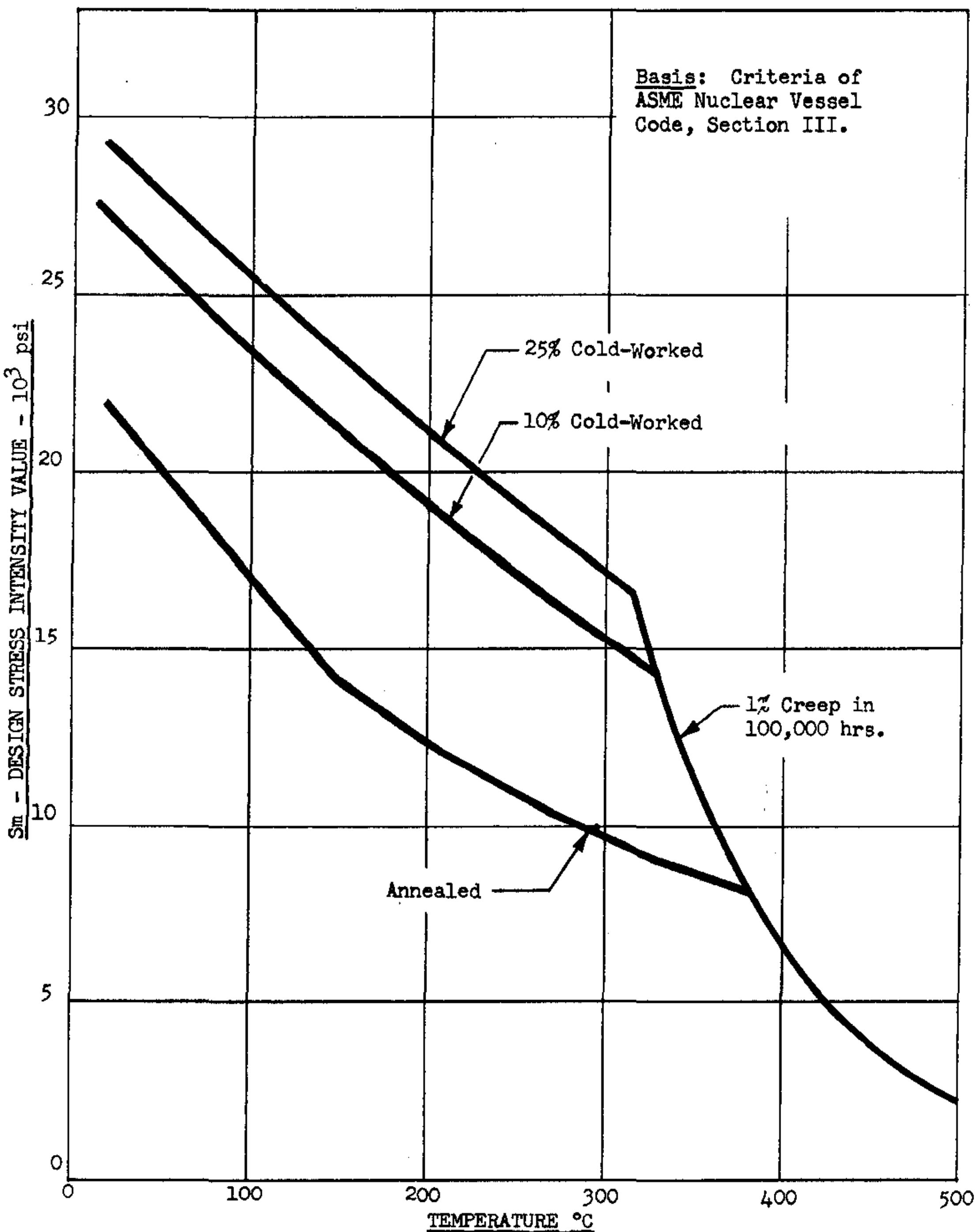
8300 MWT D_2O COOLED REACTOR
BOTTOM AXIAL SHIELD DEFORMATION

FIGURE 33



8300 MWT D_2O COOLED REACTOR
FUEL CROSS SECTION

FIGURE 34



ZIRCALOY-2 DESIGN STRESS