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PRELIMINARY

HAZARDS SUMMARY REPORT

for the

VALLECITOS SUPERHEAT REACTOR

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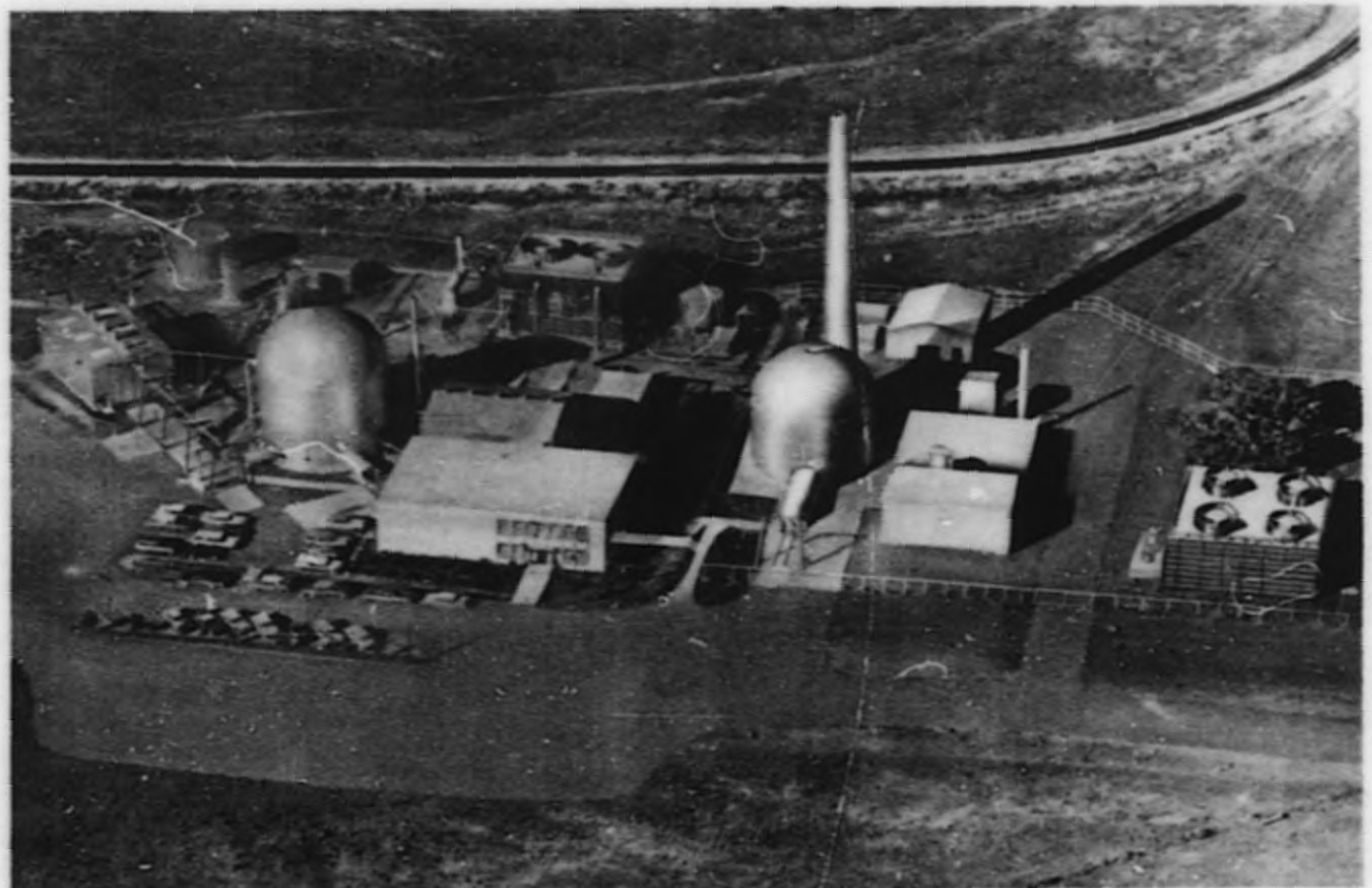
ARTIST CONCEPTION OF VSR



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PART I – SUMMARY AND CONCLUSIONS

A. INTRODUCTION

1. Background

1.1 The General Electric Company is designing a nuclear superheat reactor for operation as part of the experimental facilities at its Vallecitos Atomic Laboratory in Alameda County, California. The reactor is designed to operate initially as a separate superheat reactor at 12.5 megawatts thermal output, utilizing saturated cooling steam from a gas-fired boiler. Location of the Vallecitos Superheat Reactor (VSR) immediately adjacent to the Vallecitos Boiling Water Reactor (VBWR) facility will permit utilization of saturated steam from the VBWR at a later date as well as sharing of other VBWR equipment including the turbine-generator unit.

1.2 The VSR will be built as a developmental tool to be employed principally as a test bed for superheat reactor fuel. As such, it will be employed in such aspects of superheat fuel research as fuel design, fuel burnup, limitations on power capabilities of fuel and cladding materials, and fission gas release from defected fuel. The VSR is expected also to provide valuable information on the physics of steam-cooled reactors, turbine contamination from carryover of radioactivity from the reactor, and the integration of two reactor systems. The design and operational experience with the VSR is expected to make significant contributions to the design of large future superheat reactors.

2. Purpose and Scope of Report

2.1 This Preliminary Hazards Summary Report has been prepared for submission to the United States Atomic Energy Commission in compliance with Part 50 of the regulations governing the licensing of production or utilization facilities, pursuant to the Atomic Energy Act of 1954, as amended, and contains the general information required by 10 CFR 50.34.

2.2 Design of the VSR has progressed through the conceptual and scoping design stages into the detailed design phases. This design work, together with the associated safeguard analyses, is of sufficient scope to provide substantial assurance that an adequately safe plant of the type proposed can be designed and built at the Vallecitos Atomic Laboratory site. The principal purpose of this report is to justify issuance of a Construction Permit for the VSR.

2.3 Subsequent to issuance of a Construction Permit and prior to issuance of a Provisional Operating License, this report will be supplemented by sufficient additional information, as required by 10 CFR 50.35 and 50.36, to assure that the final design will provide reasonable assurance that the health and safety of the public will not be endangered and to permit conversion of the Construction Permit into an Operating License.

2.4 This report recognizes many of the aspects of operating the VSR in conjunction with the VBWR. However, General Electric does not intend to operate the VSR in conjunction with the VBWR, or to make any major interconnections, until after some experience has been gained through operation with the gas-fired boiler as the only source of cooling steam. Consequently, this report places primary emphasis on VSR operation with the gas-fired boiler and at a thermal power level of 12.5 megawatts. Future reports will fully support operation also using VBWR steam and will justify increasing the 12.5 megawatt thermal power limit.

B. SUMMARY

1. Reactor

1.1 The VSR is a light-water-moderated, thermal-spectrum reactor, cooled by a combination of moderator boiling and forced convection cooling with saturated steam. The reactor core consists of 32 fuel bundles containing 5300 pounds of UO_2 enriched in U-235 to 3.6%. The fuel elements are arranged in individual process tubes that direct the cooling steam flow and separate the steam from the water moderator.

1.2 The reactor vessel is designed for 1250 psig and operates at 960 to 1000 psig. With the reactor operating at 12.5 Mwt, the maximum fuel cladding temperature is 1250°F and the cooling steam is superheated to an average temperature of about 810°F at 905 psig.

1.3 Nuclear operation of the reactor is controlled by 12 control rods actuated by drives mounted on the bottom of the reactor vessel. The water moderator recirculates inside the reactor vessel and through the core region by natural convection.

2. Plant

2.1 The VSR is housed in a conventional cylindrical containment vessel designed for 58 psig. The containment vessel and other plant structures are located adjacent to the VBWR facility. The control room is located in an extension of the VBWR control building. Other plant structures include a cooling tower, a dump condenser and miscellaneous equipment building, a gas-fired boiler, and a 160-foot-high stack for controlled release of radioactive airborne wastes.

2.2 Cooling steam for the reactor is supplied from the gas-fired boiler, and superheated steam from the reactor is discharged to the dump condenser after a desuperheating stage. Provisions are made to supply cooling steam from the VBWR at a later date, and to discharge superheated steam to the VBWR turbine-condenser.

2.3 The reactor auxiliaries include a reactor cleanup and shutdown cooling system, a shield cooling system, and a fuel handling system. Reactor emergency systems include an emergency reactor cooling system, an emergency power system, a control rod scram system, and a liquid poison system. A dual-bus reactor safety system is provided to initiate emergency actions quickly and automatically.

3. Operation

3.1 Operation of the plant follows a set of basic operating principles. Administrative procedures are established to insure that the operating policies, standards, and license limitations for the plant are followed. The testing, startup, and operation of the plant will be accomplished by a carefully selected organization of trained and experienced personnel. These personnel will prepare the detailed operating and emergency procedures for the reactor prior to initial startup.

3.2 A pre-operational test program will be carried out to demonstrate that the plant and equipment have been built according to specifications and that the plant is ready for fuel loading and initial startup. An extensive program of tests and measurements is planned for the period of initial core loading and critical testing. Initial nuclear power operation will be at a low power level, followed by cautious step increases in power until the rated reactor conditions are attained.

4. Safeguards

4.1 Inherent safety features of the reactor include the negative core reactivity effects upon heating the UO_2 fuel (Doppler effect), upon increasing the temperature or void content of the moderator in the operating condition, and upon unflooding the fuel process tubes in the hot condition. Interlocks are supplied as an aid to the procedural controls to avoid operation of the reactor at times when these inherent safety features would not be available.

4.2 Safety features designed into the reactor and plant systems include a system of sensors and devices to detect potentially unsafe operating conditions and to initiate automatically the appropriate countermeasures, a set of fast and reliable control rods for scramming the reactor if a potentially unsafe condition occurs, a manually-actuated liquid neutron poison system, and an emergency cooling system to provide continued steam flow through the reactor core in the event the reactor becomes isolated from either its normal source of steam supply or discharge.

4.3 The release of radioactivity to unrestricted areas is maintained within permissible limits by monitoring the radioactivity of wastes and controlling their release. Sources of radiation are so located and shielded as to ensure that personnel exposures will be within permissible limits in the performance of normal operating tasks.

4.4 Although the possibility of an accident involving the dispersion of substantial quantities of fission product radioactivity is extremely remote by virtue of the many safety features and the application of strong procedural control, the protection of the health and safety of the public is further assured by housing the reactor and many of its auxiliaries within a high-integrity, essentially leak-tight containment vessel.

C. CONCLUSIONS

1. General Electric respectfully submits that the design information contained herein, including the safety analyses carried out as part thereof, provides the reasonable assurance required by 10 CFR 50.35 that the VSR can be constructed and operated without undue risk to the health and safety of the public. Although design and safeguard work is not now complete, General Electric believes that the information contained herein is sufficient to justify the issuance of a Construction Permit for the VSR.

NOTE: The term "Vallecitos Superheat Reactor" (VSR) and "Hook-On Superheat Reactor" (HSR), which was an earlier designation for the reactor, are used interchangeably in the descriptive portions of this report.

PART II - FACILITY DESCRIPTION

A. GENERAL FEATURES

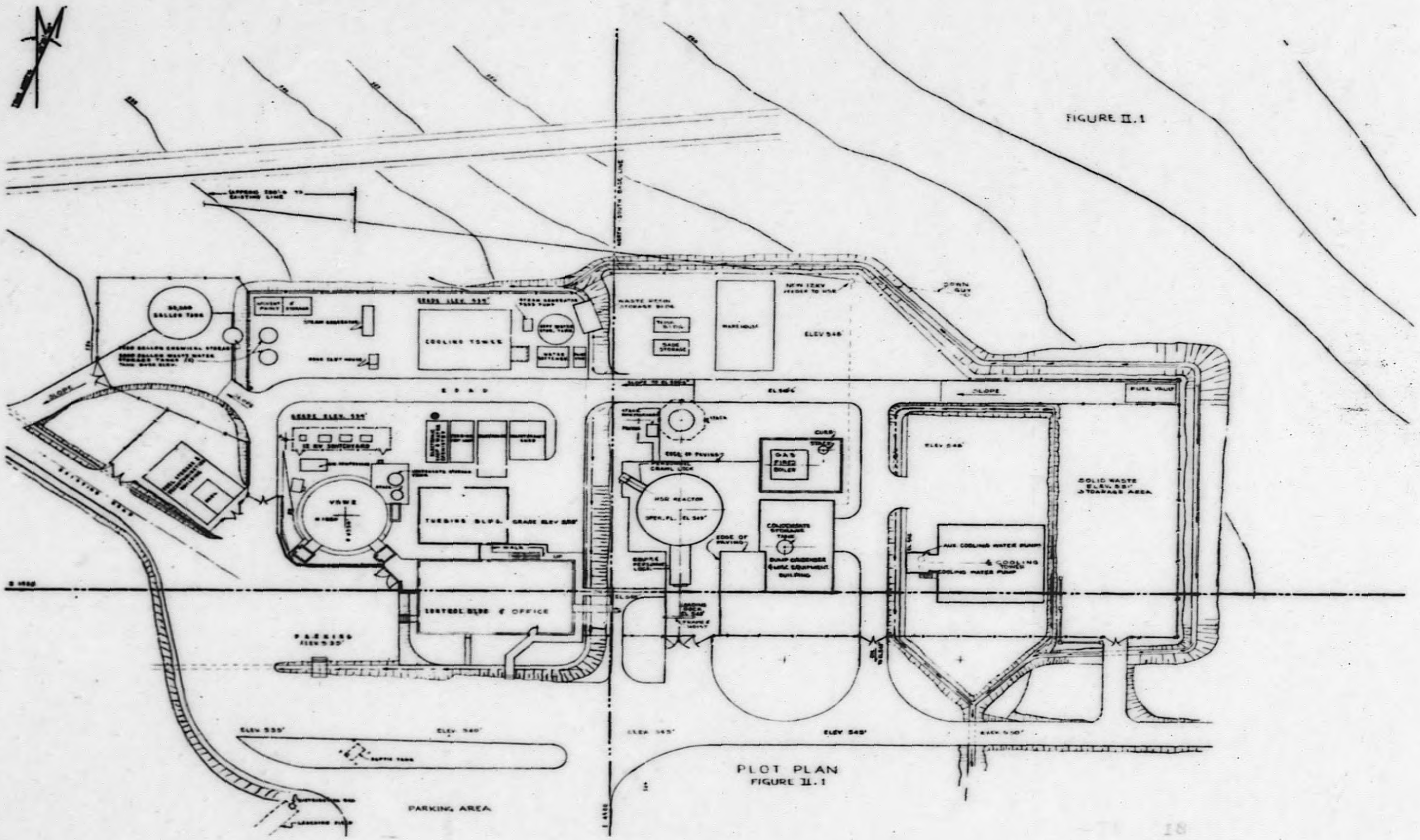
1. Site and Buildings

1.1 The overall plot plan for the VSR is shown in Figure II.1. The present VBWR boiling water reactor installation is shown to the left on the plot plan drawing. The new buildings and equipment are shown to the right. The VSR separate superheat reactor will be housed in a conventional dry containment building of cylindrical shape located due east of the VBWR. The control room for the VSR will be constructed as an extension to the existing VBWR control room. The other major new equipment to be provided with the VSR will be an exhaust stack, a gas-fired boiler, and a condenser and condensate purification equipment located in a miscellaneous equipment building. A new cooling tower and its auxiliaries will be provided also. The gas-fired boiler and condensate equipment will be located east of the containment building. The cooling tower will be located east of the dump condenser and miscellaneous equipment building. The site has been arranged so that there is convenient room for expanding all of the auxiliary buildings and equipment should the need arise in the future.

1.2 The containment building is a steel cylindrical shell, 48 feet in diameter and approximately 128 feet tall, being closed at both ends with hemispherical heads. The reactor auxiliary equipment will be located on three shielded levels below the main operating floor. The reactor core in its pressure vessel will be located to one side of the containment building and the reactor vessel will be surrounded by concrete shielding. The containment building has been located with grade at approximately the operating floor level so that the reactor vessel will be below ground, in order to minimize the shielding surrounding the containment building. Additional details of the containment and auxiliary buildings and equipment will be found in the detailed facility description portion of this document.

2. Reactor

2.1 A simplified drawing of the VSR is shown in Figure II.2. The reactor is contained within a pressure vessel of 7 feet inside diameter, approximately 32 feet in length, and designed for 1250 psig. The reactor will be a light-water-moderated, thermal-spectrum-type reactor, cooled by a combination of moderator boiling and forced convection cooling with saturated steam. The basic reactor core consists of 32 bundles of fuel, each bundle containing 9 fuel elements located within individual process tubes that direct the steam flow and separate the steam from the water moderator. Saturated steam is admitted to the reactor vessel from either the VBWR or the gas-fired boiler, or from a combination of the two. This externally-generated steam mixes with



PLOT PLAN
FIGURE II.1

FIGURE II-2

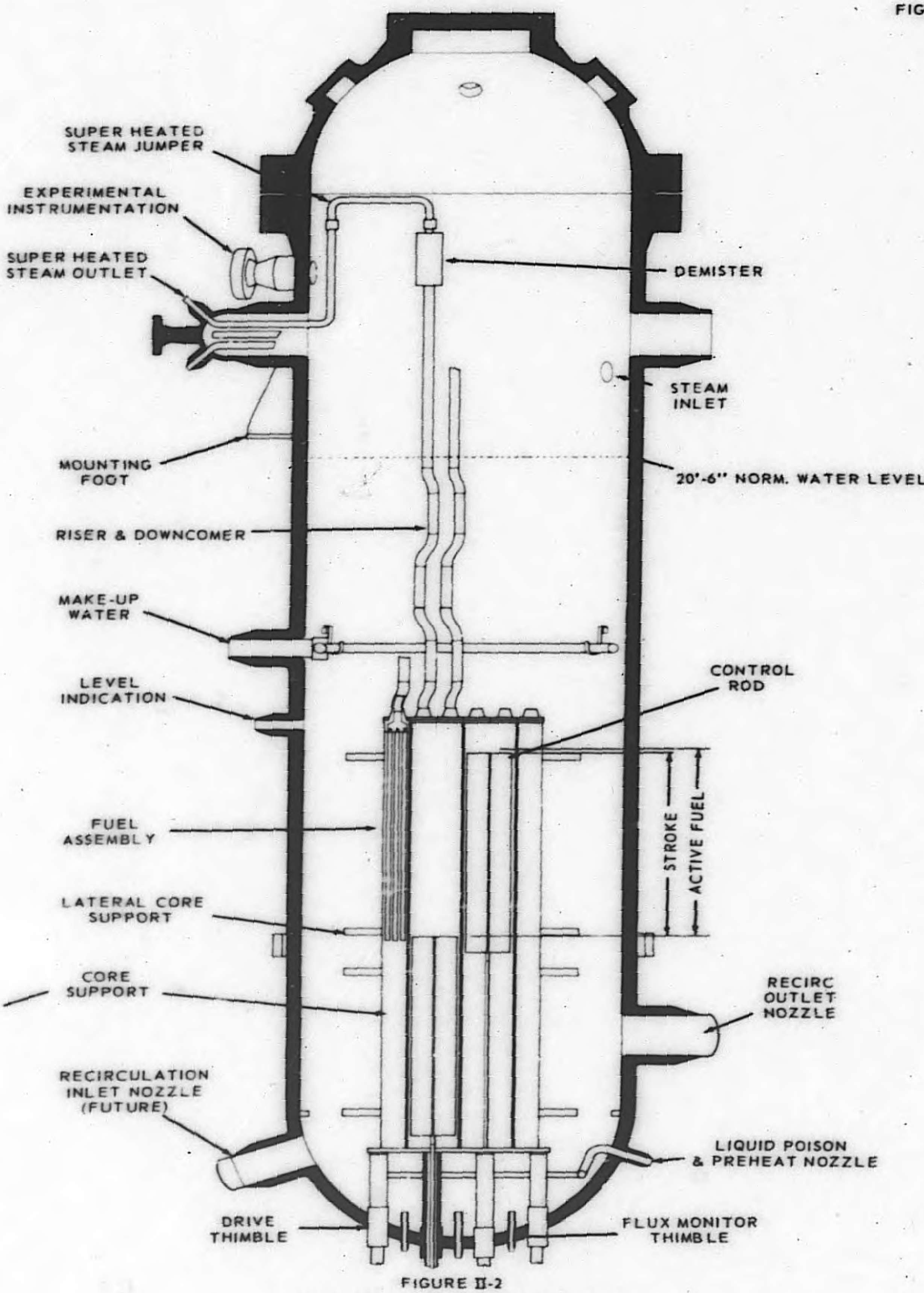


FIGURE II-2

SIMPLIFIED REACTOR SCHEMATIC

steam rising from the boiling moderator and then flows downward in the annulus formed by the process tube and outer fuel element surface. After passing down the outer surface of the fuel elements to the lower end of the process tube, the steam makes a 180-degree turn and then passes upward through the inner pass of the doughnut-shaped fuel elements. The steam is superheated during both passes and is piped from each fuel bundle through individual nozzle penetrations in the reactor vessel. The steam is collected in external steam headers and then leaves the containment building to flow to the steam turbine and the condenser systems.

2.2 Outside of the process tubes, the water moderator flows upward past the fuel-bearing portions of the fuel bundles by natural convection, and then returns downward around the outside of the core shroud that separates the heat-producing region of the core from the downcomer portion of the moderator.

2.3 The nuclear operation of the superheat reactor is controlled by 12 cruciform control rods actuated by bottom-mounted control rod drives of the twin-screw type. A detailed description of the reactor construction will be found in a later portion of this document.

3. Process Conditions

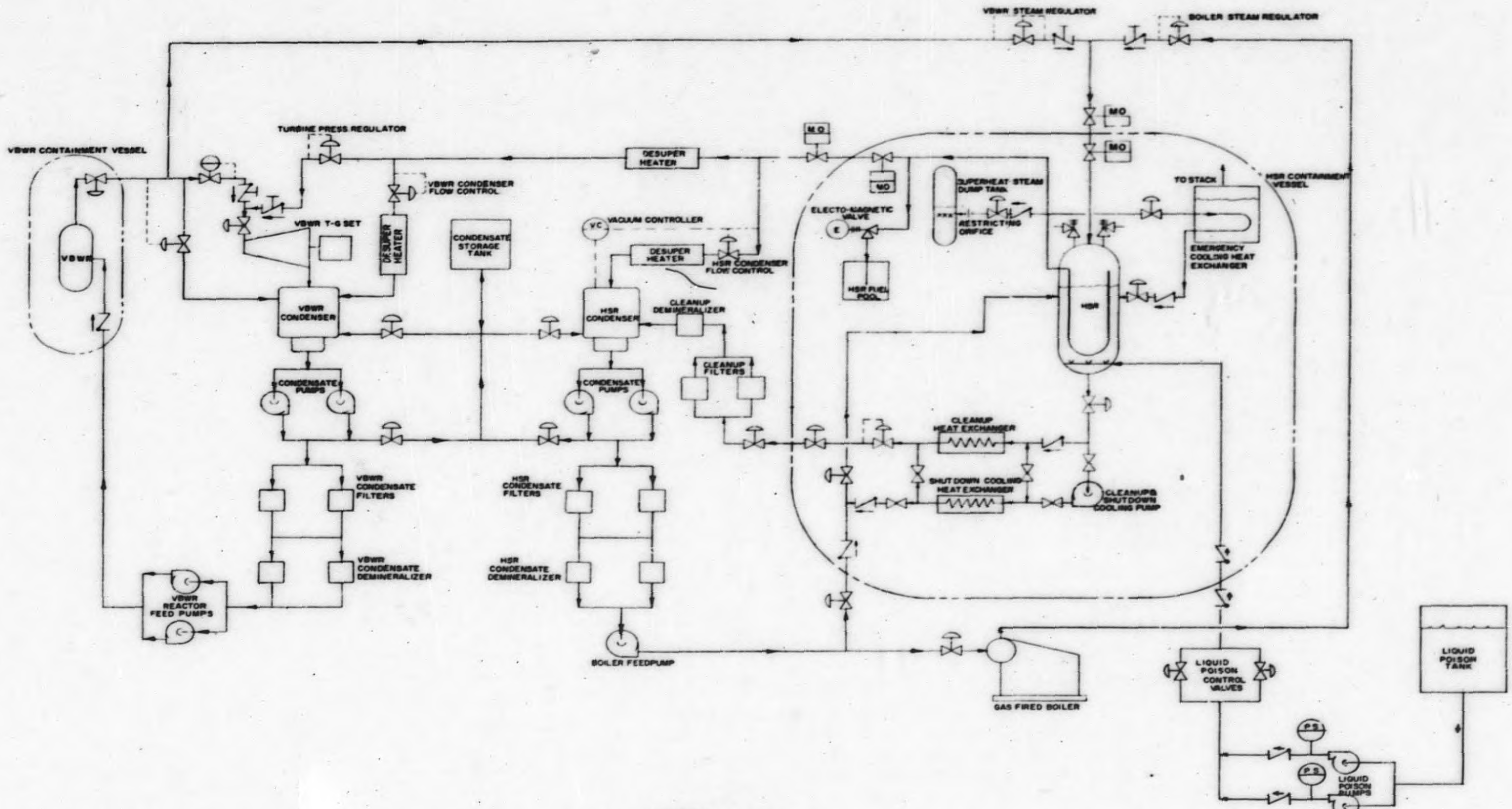
3.1 A simplified flow diagram showing the relationship between the gas-fired boiler, VBWR, VSR, and the auxiliary equipment is shown in Figure II.3. Steam for cooling the superheat fuel elements is obtained from either the gas-fired boiler, the VBWR, or a combination of the two. The VSR will be started up initially with just the gas-fired boiler, without the additional complication of the operation of the VBWR. At a later time, when increased steam flow is required beyond the capability of the gas-fired boiler, the two sources of steam may be paralleled. After the steam has passed through the superheating region of the reactor, it is permitted to flow to either the dump condenser, the VBWR turbine, and/or the VBWR condenser. Some desuperheating of the steam will be required in each instance, to prevent damaging the heat sink and power conversion equipment.

3.2 Auxiliary equipment including the necessary condensate pumps and water purification equipment are provided to enable the VSR to operate independently of the VBWR condenser and condensate system equipment, or in combination with that portion of the VBWR system.

3.3 Significant process variables include the following:

Saturated steam inlet pressure	960 psig
Saturated steam inlet temperature	541°F
Superheat steam outlet pressure	905 psig
Superheat steam mixed mean temperature	810°F
Outlet steam flow	127,500 lbs/hr
Maximum cladding temperature	1250°F
Reactor thermal power	12.5 Mw

FIGURE II 3



REFER TO FIG II 3! FOR VALVE & INSTRUMENTATION LEGEND

FIGURE II 3
SIMPLIFIED FLOW DIAGRAM

4. Nuclear Materials Handled

4.1 Table II.1 presents the kinds and approximate quantities of nuclear materials to be used for initial operations in the VSR.

5. Radioactive Effluents

5.1 Irradiated fuel removed from the reactor will be transported in a shielded cask. The irradiated fuel is normally moved to a water filled spent fuel storage pit located within the reactor enclosure. Here the fuel may be examined or stored for a short period before reinsertion into the reactor, or stored for decay before transporting from the site. Irradiated fuel which is to be examined in detail may be transported in a shielded cask to the Radioactive Materials Laboratory, which is a part of the Vallecitos Atomic Laboratory.

5.2 Spent ventilation air at approximately 15,000 cubic feet per minute and other process streams, which may contain radioactive airborne contaminants, are discharged to the atmosphere from a 160-foot-high stack. Process streams routed to this stack include secondary steam from the emergency heat exchanger, off-gases from the VSR dump condenser hold-up pipe and off-gases from the VBWR condenser hold-up pipe. (VBWR ventilation air, and gland seal off-gases via a hold-up pipe, will continue to go to the existing VBWR stack.) Effluents from the VBWR condenser hold-up pipe and from the VSR condenser hold-up pipe are individually monitored for radioactivity and filtered to remove radioactive particles. The combined effluents from the 160-foot VSR stack are also continuously monitored for radioactivity to assure that the maximum average permissible radiation dose in the environs of 500 mrem per year is not exceeded. The principle radioactive contaminants of the air at the stack are normally expected to be short-lived fission product gases and their daughters which occur from "tramp" uranium and from deliberately ruptured fuel elements. It is anticipated that the normal stack gas activity release rate will be on the order of 100 μc/second. Irradiation contaminants, such as N-13, N-16, N-17, O-19, and A-41 will be negligible in amount.

5.3 Additional amounts of radioactivity (krypton, xenon, iodine, and their daughters) may be released from defective fuel elements. In general, the fuels which will be used in the reactor will not react rapidly with the cooling steam during normal operation. Therefore, in the event of a cladding failure, the release of these fission products is expected to be gradual. The resulting buildup in contamination levels would be detected by the radiation monitoring system (discussed in Section II. H) prior to the release of hazardous amounts of activity. Charcoal filters for iodine removal are provided in both the VBWR and VSR off-gas lines leading to the 160-foot stack and in the VBWR gland seal off-gas line to the existing VBWR stack. Iodine release rates will not exceed environs iodine dose rate limits when xenon and krypton release rates are at the permissible maximum average annual rate. The limits for release of radioactivity from the stack, and the procedures followed when an excessive release is detected are discussed in Section II. J.

TABLE II.1

NUCLEAR MATERIALS HANDLED*

Material	Description	Weight - Lbs.
Uranium Dioxide	95% theoretical density, 3.6% enriched in U-235	5300
Zircalloy II	Channels (32)	720
Stainless Steel	Cladding	809
Stainless Steel	Velocity Boosters	113
Stainless Steel	Process Tubes	567
Stainless Steel	Instrument Thimbles	20
Polonium-Beryllium	Startup Source	----

* Includes non-radioactive materials within active core region.

5.4 Waste waters from the process which contain any radioactive contaminants are normally recycled through filters and ion exchange beds and are reused in the reactor system. Noncontaminated liquids are discharged to one of four retention basins on the site and carefully monitored before release. All discharges are carefully controlled to assure that concentrations do not exceed the limits specified in Part 20 of the Regulations. The average daily volume for these waste waters is approximately 60,000 gallons from the VSR during operation. The radioactivity in this waste stream is normally less than 1×10^{-7} μc/ml. If significant contamination is found in the retention basins, the contaminated waste may be given to an AEC licensed waste disposal company.

5.5 Gradual accumulation of radioactivity in the reactor water is prevented by employing water purification systems consisting of mechanical filters and ion exchange resin demineralizers. Handling of the radioactive spent units from these systems is discussed in Section II. J.

5.6 All other material and equipment not mentioned specifically in the paragraphs above will be monitored for radioactive contamination if it has been used in radiation areas. Such materials will be appropriately packaged and handled to assure adequate control of contamination for on-site storage or for off-site shipment.

5.7 The performance of irradiations may require the handling and transporting of radioactive materials in the form of fissionable test pieces, nonfissionable test pieces, and the components associated with the irradiation. This material will be transported to customers where such customers have licenses to receive it, to the other facilities on the site for postirradiation examination or storage, or to licensed disposal contractors.

B. REACTOR DESCRIPTION

1. Reactor Vessel and Internals

1.1 The reactor assembly is shown in Figure II.4. The reactor vessel, Figure II.5, accommodates the initial core of 32 fuel bundles, and provides positions for a total of 36 bundles. The pressure vessel size will accommodate an expanded core of up to 60 bundles at some future date. The vessel is fabricated from carbon steel with internal stainless-steel cladding, and will have external thermal insulation. The initial core will contain 32 superheat fuel bundles and 12 cruciform stainless-steel control rods. Each fuel bundle will have a downcomer for saturated steam and a riser for superheated steam. Each riser is connected to an individual nozzle in the pressure vessel by means of a pipe jumper.

1.2 The vertical core load is taken by the pressure vessel through the 24 drive thimbles in the lower portion of the vessel. Horizontal support is provided from the vessel shell at three elevations.

1.3 Feedwater is brought into the vessel through a sparger located approximately halfway up the reactor vessel shell. Saturated steam is brought into the vessel above the moderator water level and is directed upward to avoid mixing with the water. Superheated steam is also removed from the vessel above the water level. Nozzles located in the vessel wall above the water level are provided for special test instrumentation.

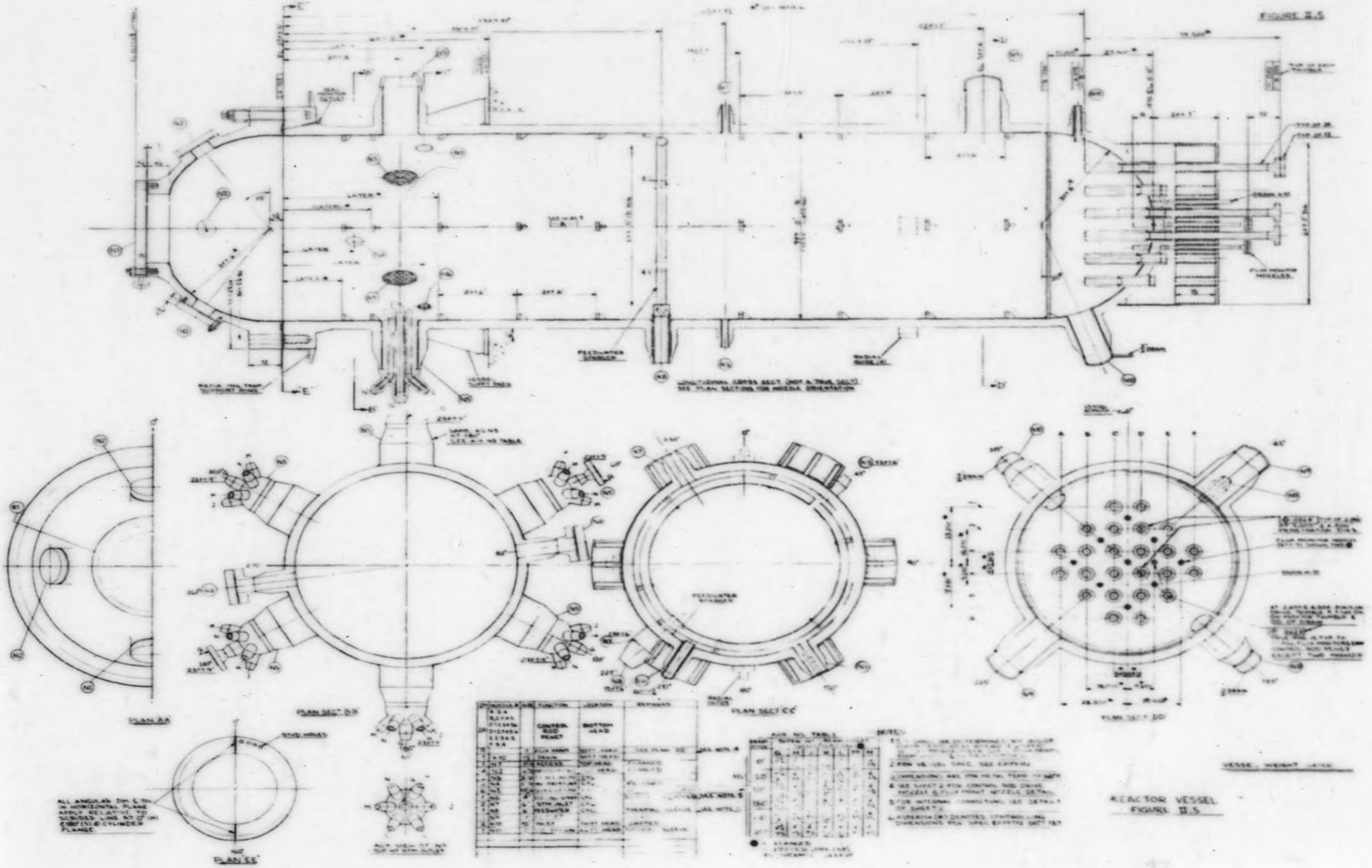
1.4 A sparger located in the bottom head of the vessel provides for the introduction of saturated steam for preheating the moderator. This sparger also provides for the introduction of liquid poison into the moderator. A series of special nozzles located in the bottom head is provided to accommodate irradiation wires used in measuring neutron flux levels within the fuel-bearing region of the reactor core. A drain nozzle is located in the bottom vessel head.

1.5 "Operating water level" measurement is provided for by two sets of diametrically opposite nozzles in the vessel wall. "Flooding water level" measurement is provided for by a nozzle in the head and by one of the lower nozzles used for operating water level measurement. A tank used for increasing the height of shielding water above the core during refueling is attached to the reactor vessel flange and extends upward into the space below the top head shield plug.

1.6 The reactor vessel will be 7 feet inside diameter and 31-1/2 feet inside height. It will be designed in conformance with the ASME Boiler Code, Section I, to contain saturated steam at a pressure of 1250 psig. The vessel will be fabricated of carbon steel plate base material internally clad with stainless steel. The vessel will be similar in most respects to a boiling water reactor vessel, except for the provision of additional outlet nozzles for the removal of superheated steam. The design temperature of the vessel is 650°F.

1.7 The vessel diameter provides for possible future integral boiling and superheating reactor cores. Two 10-inch inlet nozzles and two 14-inch outlet nozzles will permit later moderator forced circulation flows up to three million pounds per hour. Initially, these nozzles will be capped and not used. The vessel height will provide for bottom entry control rods, a 60-inch active core height, a minimum of 6 feet of water over the core for shielding purposes, and vertical space above the water level for internal steam separation from a boiling moderator.

FIGURE II 4
REACTOR ASSEMBLY



REACTOR VESSEL
FIGURE II.5

1.8 Nozzles and vessel penetrations are tabulated on Figure II.5. These include twenty-four control rod thimbles, of which twelve are required for the initial core and twelve are provided for later expansion. Of the nine irradiation wire penetrations, the central five will be used on the initial core and four are provided for later use. Two 6-inch-diameter steam inlet nozzles capable of handling 300,000 pounds per hour of saturated steam are provided. Two 6-inch-diameter instrument nozzles are provided below the head closure flange. These will be used for special test instrumentation by the experimenters. Thirty-two superheated steam outlet nozzles (plus eight spare outlet nozzles) with thermal sleeves will be provided. Together, these 32 nozzles will be capable of handling up to 300,000 pounds per hour of superheated steam flow at up to 1100°F without exceeding a 200 foot-per-second steam velocity. The upper head will provide a 31-inch-diameter center access port and four 8-inch-diameter viewing ports. These will allow direct refueling of the center 12 fuel assemblies without requiring removal of the vessel head.

1.9 External vessel attachments will include a ring on the vessel flange for attachment of the refueling tank. The vessel supports will be located below the superheat steam outlet nozzles. Attached to the lower head of the reactor vessel will be a 6-foot-diameter tray which will be filled with lead, steel shot, or concrete after assembly to provide a lower radiation shield. Attachments for external thermocouples and insulation will be provided also.

1.10 Vessel internal pads and attachments will include provisions for horizontal support of the core and for support of the spargers and internal steam piping. The internal stubs of the control rod drive thimbles will provide vertical support for the core. The feedwater sparger will be designed to provide for uniform flow distribution of 15,000 pounds per hour of feedwater, and will provide for later modification to handle feedwater flows up to 300,000 pounds per hour. A steam inlet baffle will be provided to prevent impingement on, and excessive turbulence of, the water surface. Additional internal pads are provided to permit the installation of experimental equipment or to make core modifications. Irradiation test coupons of material taken from steel used in forming the cylindrical shell of the vessel will be installed in the space surrounding the core and will be used to obtain information on material property changes during the life of the reactor vessel.

1.11 The core support assembly, shown on Figure II.4., consists of a support plate and honeycomb assembly. The support plate is stainless steel and will rest on the top of the internal stubs of the control rod drive thimbles which are located in the bottom vessel head. This plate has guides to locate the honeycomb horizontally relative to the plate. Holes are provided in the plate to clear the extension shafts which connect the control rods to their drives. The amount of clearance between the extension shafts and the support plate will control the amount of water that circulates by the control rods. Additional holes are provided for inserting the irradiation wire tubes.

1.12 The honeycomb assembly is made of stainless steel channels fastened together in a manner that provides square openings to receive the nose pieces of the fuel channels, cruciform openings for guiding the control rods, and square openings for inserting irradiation wire tubes. Latching lugs are provided inside the fuel channel nose piece opening for latching each fuel channel to the honeycomb. Orifice support brackets are located below the level of the fuel channel nose pieces to permit insertion of an orifice for any fuel channel if orificing of moderator water flow to a fuel bundle is desired for a specific test. The honeycomb is supported horizontally near its top and bottom from the pressure vessel wall. A guide and support is provided around the top of the honeycomb to locate and hold the shroud which surrounds the fuel channels.

1.13 The core area structure consists of a shroud, fuel channel assemblies, and irradiation wire guide tube assemblies. The stainless steel shroud is supported and located at the lower end by the honeycomb assembly and at the upper end by a horizontal support from the vessel wall. The shroud locates the fuel horizontally and prevents cross circulation of cooling water through the core.

1.14 The fuel channels are approximately 5.85 inches square by 1/16-inch thickness, and are fabricated from zirconium. At the lower end, a hollow square stainless steel channel nose piece is fastened inside the zirconium channel. The nose piece has a tapered end to facilitate guidance into the honeycomb and has latch springs for positive hold-down to the honeycomb. During normal refueling the channels remain in the core. However, the channel (after fuel removal) can be readily removed with a special tool if it is desired to do so. Openings are provided near the top of the channels to provide for water circulation and steam exit. The channels form the guides for the upper end of the control rods.

1.15 The stainless steel wire irradiation guide tubes (combined with the honeycomb, support plate, and vessel thimbles) provide for the insertion and removal of the wire irradiation tubes. Spacer blocks are fastened to the upper end of the guide tubes. These spacers locate the fuel channels and guide tubes relative to each other. The lower end of the guide tube fits into and rests on the honeycomb. A "washer" on the guide tube provides the surface for resting on the honeycomb, and when the fuel channels are in place, prevents the removal of the guide tube. By removing the four adjacent fuel channels, a guide tube can be removed from the core.

1.16 The reactor internals are designed to accommodate either a single-pass or a double-pass type of fuel bundle. Each fuel bundle contains an array of process tubes to separate the steam from the moderator water and a concentric riser and downcomer sub-assembly that is used to admit steam to the fuel bearing portion of the bundle and to connect the superheated steam exit plenum to the reactor vessel outlet nozzle. A fuel element is located within each process tube and is supported by one of two tube sheets that directs the flow within the fuel bundle and forms a steam distribution header.

1.17 The superheated steam riser tube is welded to, and receives superheated steam from, the inside chamber of the header formed by the two tube sheets. The saturated steam downcomer tube is outside of and concentric with the riser and is welded to the outer cone of the fuel bundle. The annulus formed by the two concentric tubes supplies saturated steam to the outer chamber of the bundle header. Sections of tubing are placed inside of and spaced from the riser to form a dead steam area to provide insulation for the riser tube. These concentric tubes are offset along their length to eliminate a neutron streaming path through the reactor water. At a point below the vessel flange, the riser passes through a demister, and the downcomer is fastened into the demister. The downcomer and the outer wall of the demister form an annulus which contains stainless steel screen or wool. Saturated steam enters the bottom of the annulus, flows through the stainless steel screen (which removes any water that is in the steam) and into the downcomer. The riser, after passing through the demister, ends above the normal moderator water level in a socket for the jumper.

1.18 Stainless steel jumpers connect the risers from the fuel bundles to the piping from the reactor vessel nozzles. The jumpers, like the risers, will contain an inner tube to form a dead steam insulating blanket. Each end of the jumper will have a remote operable joint which allows removal of the jumper for refueling purposes. The joint will have a metal-to-metal or a stainless-steel-clad asbestos gasket joint. The joint will be closed with a nut similar to a pipe union or with a toggle. The final design will be determined by further studies and tests.

1.19 The piping from each jumper will go down the inside vessel wall, around the wall, and out through one of the six large nozzles to one of the smaller superheat steam outlet nozzles which penetrates the vessel. (The arrangement of large vessel nozzles with smaller nozzles which actually penetrate the reactor vessel shell is shown on Figure II.4) This piping will also contain internal tubing for insulating purposes. The use of insulating tubing within the piping, and the flexibility provided by the piping arrangement from the fuel bundle to the outlet nozzle, make it unnecessary to use any special flexible joints in the internal superheat lines to allow for thermal expansion and mechanical misalignment.

1.20 Guides will extend horizontally from the vessel wall to the demisters and up to the jumpers. Holes and slots will be provided in these guides for the superheated steam piping and jumpers to limit their travel. These guides will limit the travel of the pipes and demisters to prevent damage from vibration and accidental dropping of items on the pipes. Additional guides may be added as indicated by future studies.

2. Core Layout

2.1 The core lattice consists of 36 6.60-inch square spaces with control rods located on alternate corners of the lattice. Initially, the core will operate with 32 fuel assemblies, leaving the four corner core positions unfueled. A typical cross-section of the core is shown on Figure II.4, the reactor assembly drawing. In the 32-bundle loading with 12 control rods, each of the non-peripheral fuel bundles has a control rod at two diagonally opposite corners of the bundle. This arrangement results in a control blade being on each of the four sides of the non-peripheral bundles. The bundles which form the core periphery have a control blade on two of the adjacent sides which face other fuel bundles. Both of these blades are part of the same control rod.

2.2 The 6.6-inch lattice permits use of standard drive components proven on other designs. The lattice size chosen also represents a reasonable balance between regionalizing the core into orificed areas while not locating too many elements within each region, or bundle.

2.3 In the initial startup core, all 32 of the lattice positions may contain a "Mark I"-type fuel bundle. However, throughout most of the life of the reactor, it is expected that it will contain experimental fuel bundles of many different types. The lattice permits physical interchangeability of the Mark I and other fuel bundles, with the location of particular bundles dependent principally upon experimental variables and the relative need for frequent removal (the center 12 lattice positions are more convenient for frequent removal). Four of the center 12 positions are piped to the "divert"header, which passes steam directly to the VSR condenser.

2.4 Details of the core layout have been discussed in the previous section on "reactor vessel and internals".

3. Fuel Assemblies

3.1 The basic fuel bundle for this reactor is a process tube type of design in which a metal tube separates the moderator from the cooling steam in contact with the fuel element cladding. In order to eliminate steam pipe penetrations in the portion of the reactor vessel below the active core, the core has been designed to use a U-Tube or bayonet arrangement of tubes to provide a steam flow path for cooling the superheater fuel. If the steam flow in a bundle is to be single pass-parallel, in which steam flows in the same direction on both the outside and inside of an annular fuel element, the bundle will consist of an array of U-Tubes connected to tube sheets at the top of the bundle. If the steam flow in a bundle is to be two pass-series, in which the steam flows first in one direction along the outside of an annular fuel element and returns in the opposite direction along the inside annulus of the fuel element, then the bundle will consist of tubes closed at one end and connected to a tube sheet at the other. In this bundle, the "Mark I" type, a second tube sheet is attached to the fuel elements and assists in separating the incoming and exit steam flow. The proposed arrangement of fuel bundles with respect to the reactor vessel and other internals is shown in the reactor assembly drawing, Figure II. 4. Figure II. 6 is a schematic drawing of a two-pass fuel bundle which shows the process tubes, fuel bundle and fuel element cross-sections, and the steam flow path through the bundle.

3.2 The Mark I fuel bundle consists of nine annular fuel elements arranged in a 3 x 3 array. The fuel elements are supported by tube sheets and are enclosed within stainless steel process tubes that prevent the moderator water from mixing with the cooling steam. Figure II. 6 is representative of the Mark I bundle arrangement, and Figure II. 7 shows some of the details of the Mark I assembly.

3.3 The saturated steam is distributed to the nine fuel elements by means of a steam plenum between the upper and lower tube sheets. The lower tube sheet supports the process tubes and also contains a shoulder that supports and positions the bundle on the zirconium channel surrounding the fuel bundle. The process tube will be formed from type 304 stainless steel of 0.025-inch thickness. The process tubes will be welded closed at the lower end and welded to the tube sheet at the top.

3.4 The upper tube sheet will be welded to a tube extending from the top of the active portion of the fuel element. Both tube sheets, as well as the riser and downcomer tubes to which they are attached, will be formed from type 304 stainless steel. A stainless steel nose piece will support the lower end of the process tubes and guide and protect the process tubes during operation and refueling.

3.5 The arrangement of tube sheets and the annular configuration of the fuel element and process tube form a steam flow path that is first downward over the outside surface of the fuel element and then upward over the inside surface. A stainless steel velocity booster tube, shown on Figure II. 7, is centered in the inside of the fuel element to reduce the flow area and increase the velocity of the cooling steam in this region.

3.6 The Mark I fuel element is formed from two 0.028-inch-thick concentric type 304 stainless steel tubes containing sintered hollow pellets of low-enriched UO_2. The annulus between the concentric tubes, which contains the fuel pellets, will be sealed at both ends by stainless steel plugs.

3.7 The core used for the safeguard analyses in this report is the Mark I core, which consists of 32 bundles of the Mark I fuel type described above. This fuel type is under intensive development at the present time. Individual fuel rods of the Mark I design will have undergone preliminary irradiation in the SADE (Superheater Advanced Demonstration Experiment) loop of the VBWR prior to final design of the fuel bundle. Similar fuel elements having 0.049-inch-thick cladding have been irradiated successfully in the SADE loop for a period of about 100 hours, and at the time of preparation of this report a similar element having 0.028-inch-thick cladding is under irradiation in the loop.

FIG. II. 6

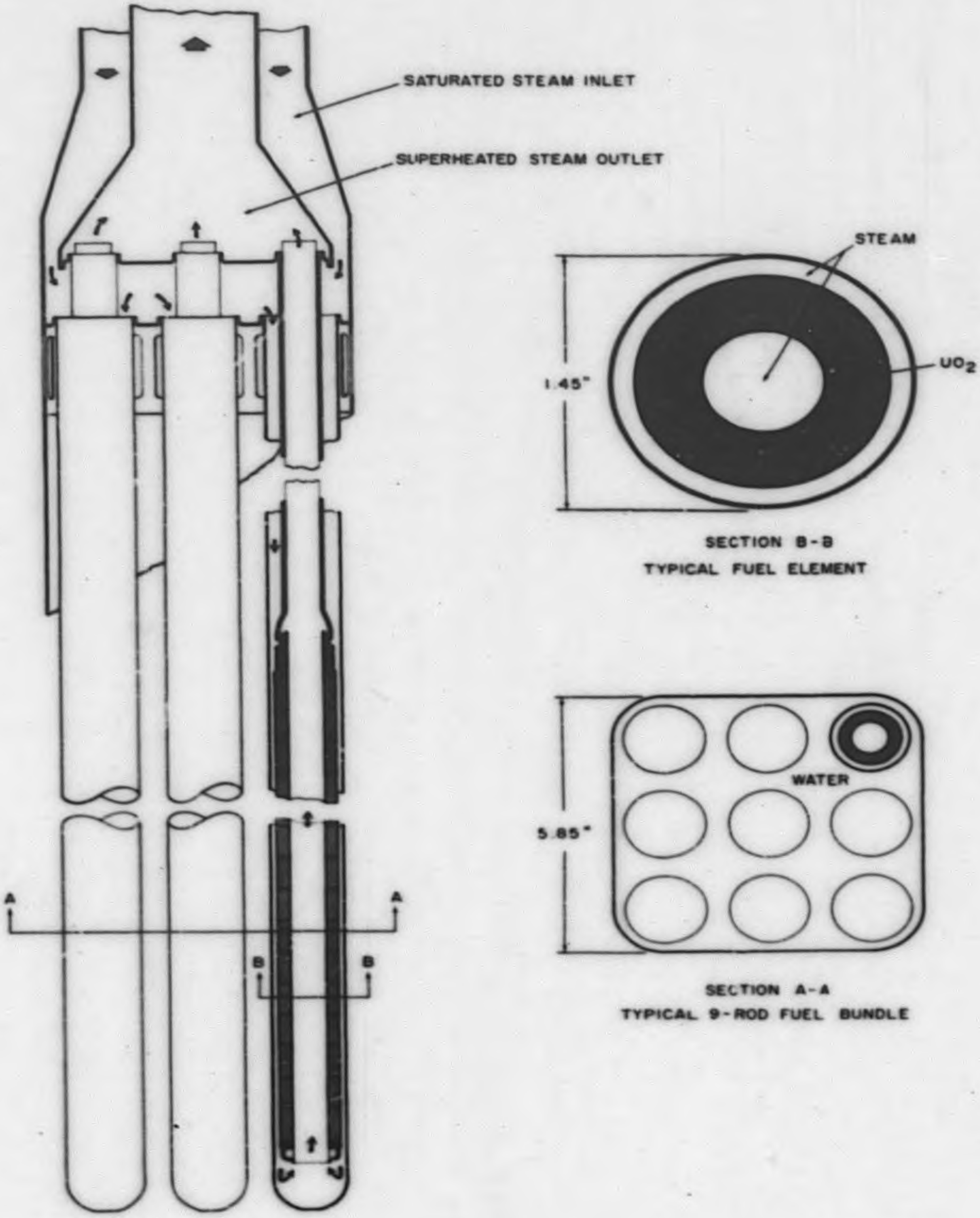
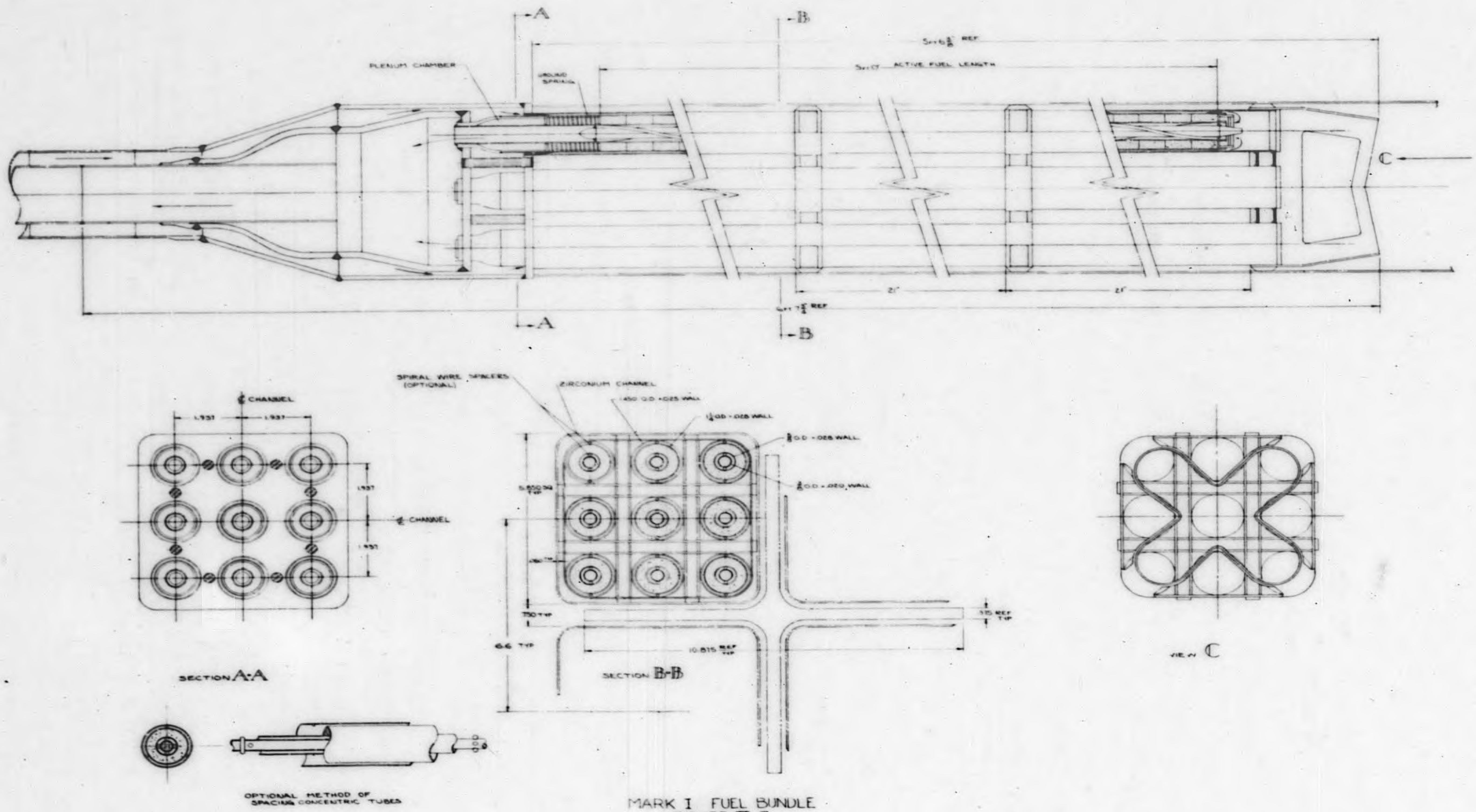


FIGURE II. 6
FUEL BUNDLE SCHEMATIC

FIGURE II.7



MARK I FUEL BUNDLE
FIGURE II.7

3.8 As indicated previously, the Mark I core may be used for initial startup. However, other developmental fuel bundles are under investigation as part of the VSR program, and depending on the results of analyses and preliminary experimental information available by the time of VSR start-up, it may be desirable to include some of these in the initial core. If this should become the case, a specific proposal and evaluation would be presented to the Commission prior to obtaining a license for initial startup of the VSR. For the purposes of evaluating the safety of the VSR design in connection with issuance of a Construction Permit for the reactor, it should be assumed that the initial reactor core will consist of 32 Mark I fuel bundles.

3.9 Section III. J, "Experimental Operation" indicates the type of fuel development to be undertaken in the overall experimental program for the VSR. It is to be noted that the VSR will be used to investigate and develop superheat reactor fuel much in the same way that the VBWR has been used to develop boiling water reactor fuel.

4. Control Rods and Drives

4.1 Each of the 12 control rods is operated by a drive mechanism which is capable of both shim and scram action. Shim control is done by electro-mechanical means employing a 3-phase AC electric gear-motor. Scram action is accomplished by a pneumatic cylinder supplied with air from an air accumulator and compressor, and controlled by solenoid valves which are operated by the reactor safety circuit. The drive is designed for high reliability and simplicity, and with a maintenance schedule compatible with the reactor refueling schedule.

4.2 The control rods are stainless steel of cruciform cross-section, have blades which are about 11 inches wide and 1/4 to 3/8 inches thick, and are 5 feet in length. They will be connected to the control rod drive extension shaft by an interrupted lug type connector. The connector is rotated by turning a shaft that extends through the center of the control rod. The shaft will have a handle above the top of the control rod, but the handle will be below the top of the fuel channels. Sixty-degree rotation will release the connector. The handle will have a latch to hold it in the "connected" position, and will be so designed that it cannot rotate when the adjacent fuel channels are in place. Rollers at the top of the control rods ride on the fuel channels. Rollers at the bottom of the control rod ride on the honeycomb assembly. The stroke length is 4 feet 10 inches. In the fully inserted position, the blade will extend to within 2 inches of the top of the active core; in the fully withdrawn position, the upper tip of the blade will be flush with the bottom of the active core.

4.3 The VSR control rod drive is shown on Figure II. 8. The drives are operated from the control panel in one of the following modes: normal or shim control, all rods run in together at shim speed, or scram. The shim speed of each drive is essentially constant at approximately 1/3 inches per second. Scram operation will give 10% travel in a maximum of 0.6 second and 50% travel in a maximum of 1.0 second, including the delay time.

4.4 Limit switches on each drive are used to indicate the fully inserted and fully withdrawn positions of the control rod at the control panel by means of pilot lights. A synchrotype transmitter and receiver indicate continuously the position of each drive mechanism at the panel. Upon scram, when the control rod shaft disengages from the remainder of the traveling nut-carriage and position indicator, the disengagement is indicated by a pilot light at the panel from a switch on the drive mechanism. These devices will be grouped together and arranged on the control room panel in a pattern simulating the relative location of the rods with respect to each other in the reactor core.

4.5 Whenever the rod is separated from the traveling nut-carriage, a latch on the control rod shaft is free to engage to prevent movement of the rod in the withdrawal direction. This prevents the rod from being expelled from the core by reactor pressure or its own weight in case the scram air supply should fail. The latch cannot engage in such a way as to prevent rod insertion. The nut-carriage is driven "in" automatically when the control rod and nut-carriage are separated after scram.

4.6 The control rod drive mechanism is attached to the control rod by means of a drive shaft and an extension shaft. The drive shaft penetrates the bottom head of the reactor vessel through a shaft seal which is mounted at the bottom of the vessel thimble. The shaft seal permits linear motion of the drive shaft to move the control rod upward into the core or downward out of the core, and also controls the leakage of water around the shaft by means of a series of close-fitting rings. Demineralized water is injected near the middle of the seal at a pressure above the reactor vessel pressure. Hence, the injection water flows both into the reactor and out to a drain to the reactor water storage tank. The shaft seal is removable while water is in the vessel by engaging a plug on the extension shaft with a fixed seat. The seat is located between the shaft seal and vessel thimble, and upon advancing the drive shaft downward past the normal stroke limit, a seal is accomplished. The drive may be removed separately, if desired, leaving the shaft seal in place.

4.7 The pneumatic system, shown on Figure II. 9, provides for rapid insertion of the control rod and for partial balancing of the reactor vessel pressure load on the drive mechanism. The balance pressure at the bottom of the cylinder is regulated during normal operation at 50 or 25 psig (step-varied with reactor pressure), selected in proportion to reactor pressure. Scram action is accomplished by a signal from the reactor safety system which admits air at approximately 250 psig to the bottom of the scram cylinder from an accumulator.

FIGURE II. 8

CONTROL ROD DRIVE ASSEMBLY

FIG. II. 9

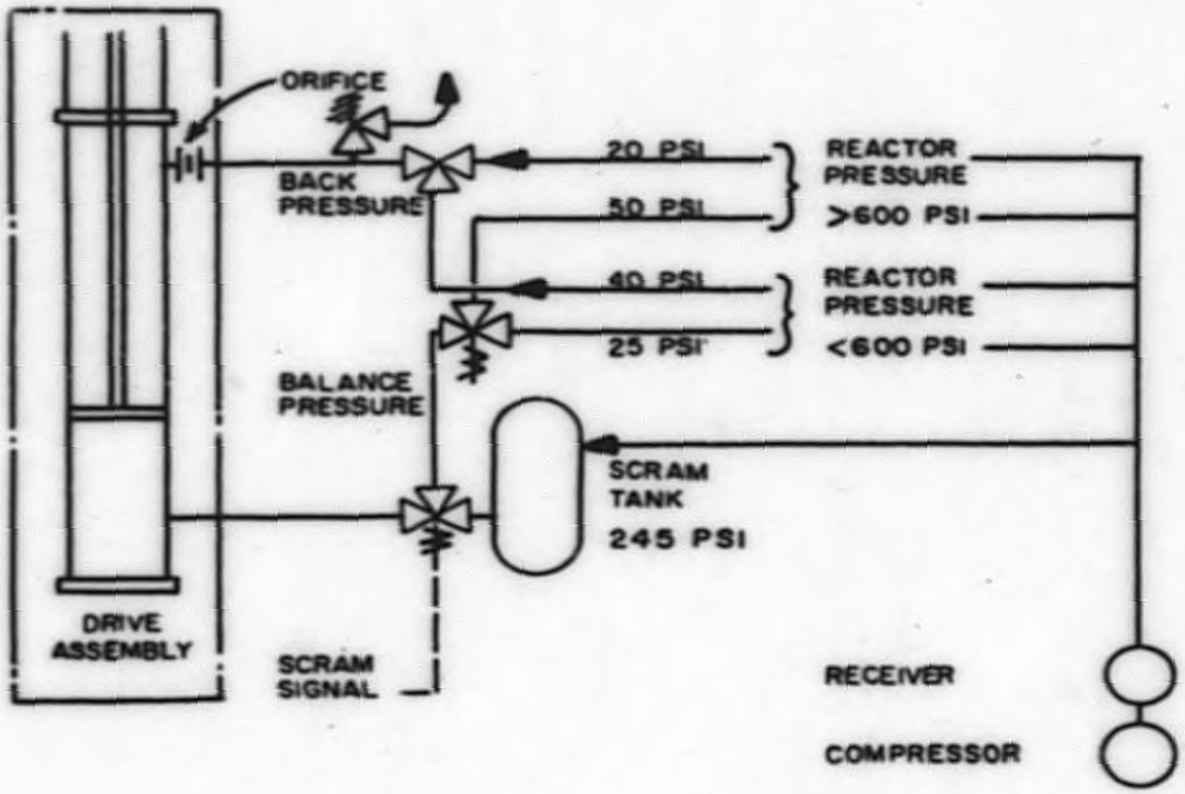


FIGURE II. 9
PNEUMATIC SCRAM SYSTEM

Deceleration of the control rod is controlled by several factors: an initial positive back pressure of 20 or 40 psig (step-varied inversely with reactor pressure) on the top of the scram cylinder; an air cushion and a restricting orifice on the cylinder; and relief valves on the back pressure header. The back pressure assures engagement between the drive shaft and the nut carriage under conditions of low reactor pressure. The pneumatic system is provided with a compressor, filters, various valves for pressure control, and instrumentation for local and remote monitoring of supply pressure, scram tank pressure, back pressure, and balance pressure.

4.8 The shaft seal water system provides demineralized and filtered water for injection near the center of the shaft seal at a pressure above reactor pressure in order to prevent leakage of reactor water from the vessel and to cool the seal and shaft. Part of this water leaks into the reactor and part to a drain. A third connection is made at the high pressure end of the seal assembly in order to check for proper sealing of the thimble valve plug and for flushing out foreign particles from above the seal. Instrumentation is provided to monitor feedwater pressure and filter pressure drop, excessive feedwater flow, and water flow rate to drain.

C. NUCLEAR CHARACTERISTICS

1. General

1.1 The Vallecitos Superheat Reactor is a light-water moderated, steam cooled, separate superheat reactor. As such, it will possess most of the nuclear characteristics of a boiling water reactor, plus the heat transfer characteristics of a gas-cooled reactor. As noted in Table II. 2, about 11% of the active core volume is occupied by superheated steam or void. It is the reactivity effect of unflooding or flooding this volume which introduces several unique considerations in the safe design of this reactor.

1.2 From the nuclear viewpoint, the following are the significant physical characteristics of the Mark I core:

1.2.1 The fuel material is UO_2 of 95% theoretical density enriched to about 3.6% in U-235. The UO_2 is clad with stainless steel. Under normal operating conditions, the average UO_2 temperature is about 1250°F.

1.2.2 The fuel geometry is that of annular fuel rods (Figure II. 10) arranged in a square array of 9 superheat fuel rods with each rod surrounded by a zircaloy channel for the purpose of directing the moderator flow. The control rods operate within the 3/4-inch water gap between zircaloy channels (Figure II. 11).

1.2.3 The moderator is light water. Under normal operating conditions ($\sim$1000 psi, saturation temperature $\sim$ 545°F), there is steam formation in the upper part of the core. Steam voids are confined to the moderator enclosed by the zircaloy channels and range up to 30% of the moderator volume at the exit of the hottest bundle. The overall core average void content at full power is about 6% of the moderator volume within the bundles.

1.2.4 The core consists of 32 fuel bundles arranged in a 6x6 array with corners removed as shown in Figure II. 12. The active fuel height is 60 inches, and the mean core diameter is 42. 1 inches.

1.2.5 The Mark I core is designed for an average burnup life of 5000 Mwd/T. The enrichment of the Mark I fuel has been set to achieve this exposure which corresponds to an initial infinite multiplication constant of 1. 26.

1.2.6 Reactivity control is achieved with twelve cruciform blades of stainless steel containing no additional poison, arranged as shown in Figure II. 12. The control system is capable of controlling a reactivity change of 11% $\Delta k/k$ at 68°F and 15% $\Delta k/k$ at 545°F. The control rods enter the core from the bottom and are continuously positionable over a 58-inch stroke.

1.2.7 The fuel rod and bundle design was chosen to achieve a water-to-uranium oxide volume ratio of 3. 7. This relatively high water-to-fuel ratio helps to minimize the positive reactivity effect caused by accidental flooding of the superheat passages. On the other hand, the high water-to-fuel ratio results in positive void and temperature coefficients at room temperature, and a positive reactivity addition due to unflooding the

TABLE II. 2

MATERIAL VOLUME FRACTIONS IN THE MARK I CORE

Uranium Dioxide	0.1679
Water	0.6231
Stainless Steel	0.0608
Zircaloy	0.0355
Superheated Steam Passages	
Outer Annulus	0.0645
Inner Annulus	0.0300
Central Fuel Rod Void	0.0182
	1.000

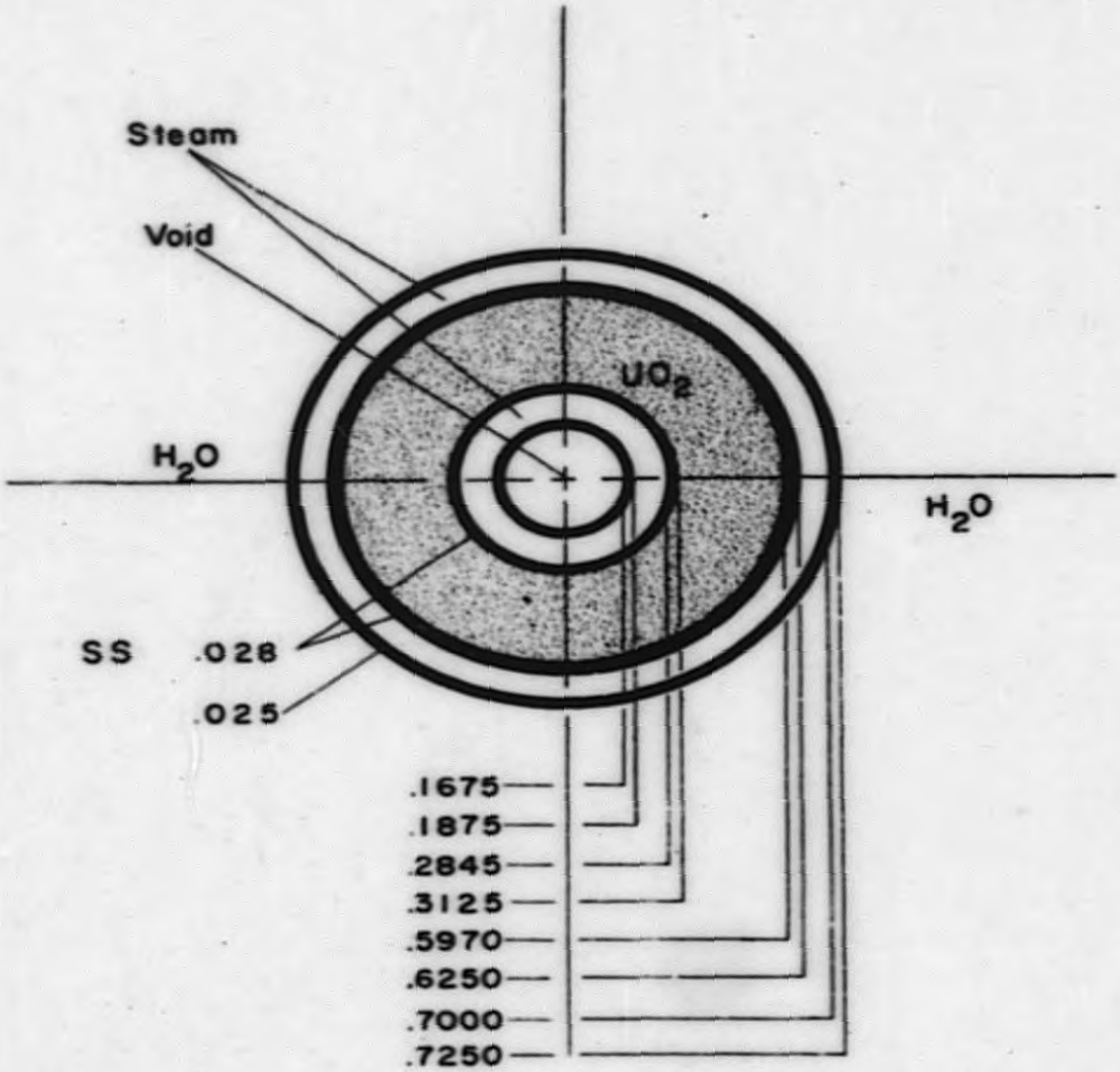


Figure II. 10

Process Tube Assembly

Note: all dimensions expressed in inches

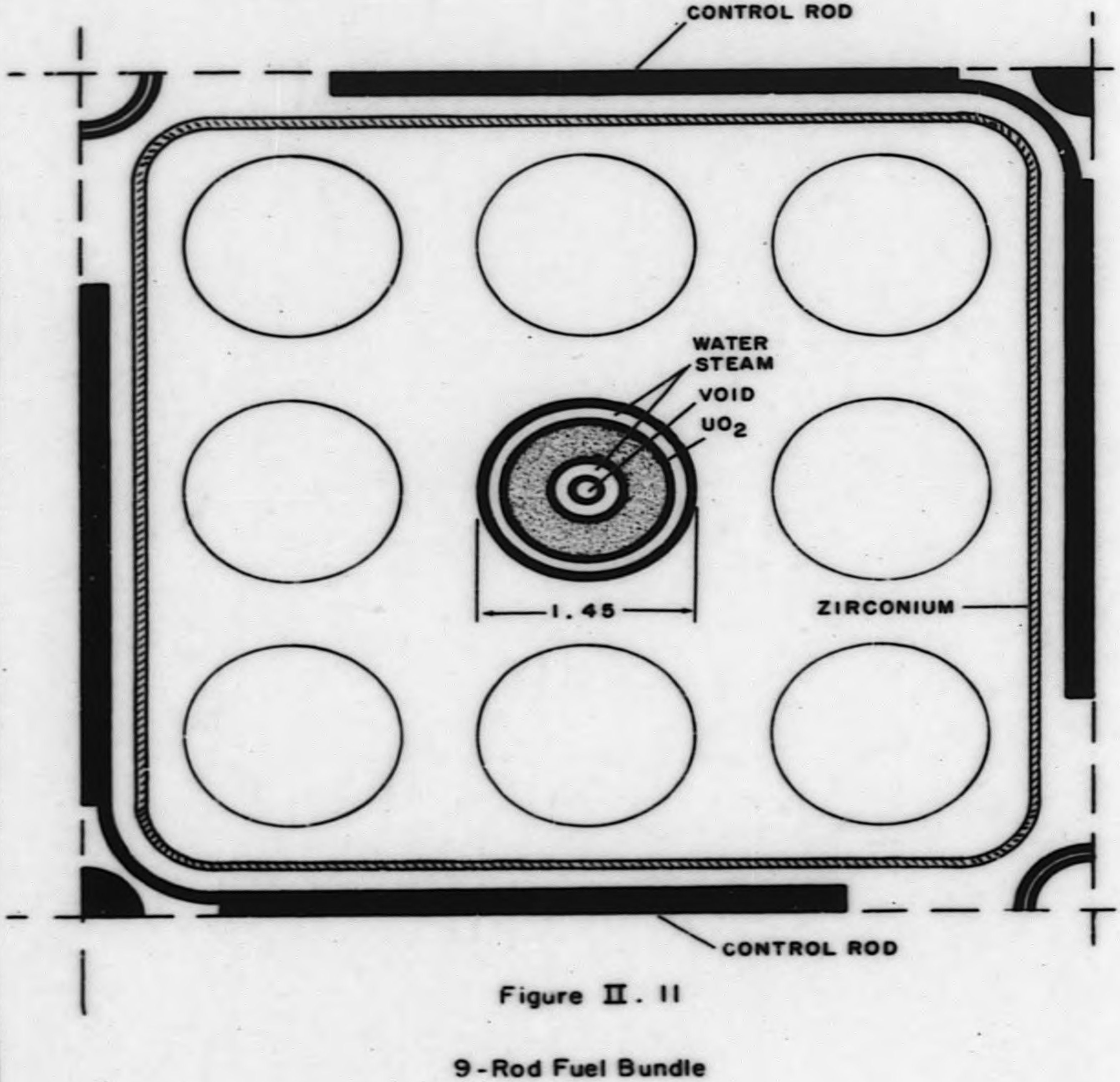


Figure II. 11

9-Rod Fuel Bundle

FIG. II. 12

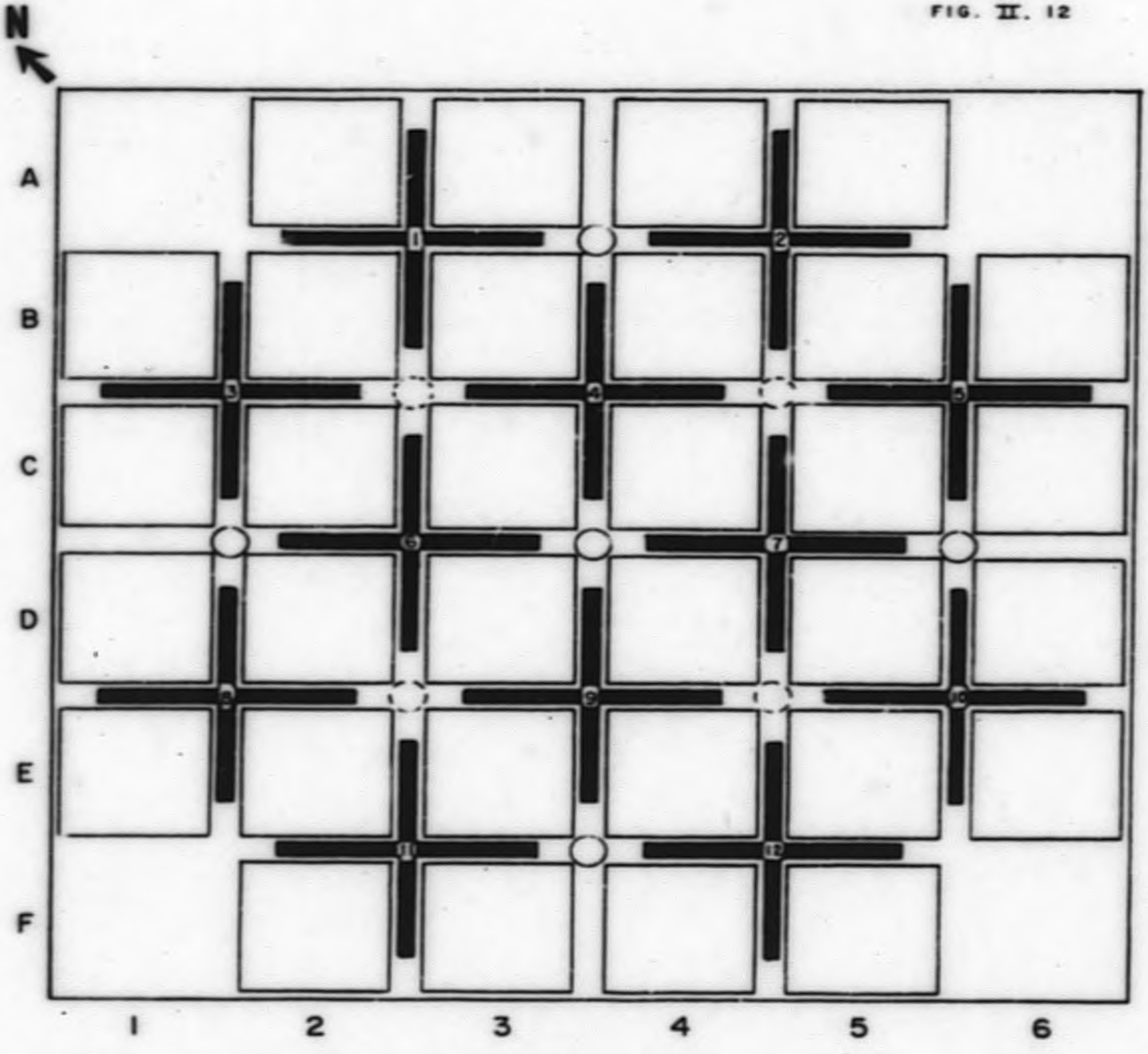


Figure II. 12

Core Arrangement with Rods Inserted

superheat passages at room temperature. The latter results make it advisable as part of the startup procedure to preheat to near the operating temperature before unflooding and rod withdrawal are initiated.

1.2.8 Other significant nuclear lattice parameters are presented in Table II. 3 for the control-free condition.

1.3 The physics design criteria which are most significant to reactor safety are presented in Section IV. D. These have to do principally with the reactivity coefficients associated with changes in temperature and voiding in the reactor, as well as with control rod strength, reactivity shutdown margin, and nuclear stability. The detailed physics calculations discussed and summarized in the subsequent paragraphs have been guided by these safety criteria as well as by other design and operating criteria.

2. Reactivity Coefficients

2.1 Table II. 4 summarizes the reactivity coefficients calculated for the Mark I core. Operation with the indicated positive void and temperature coefficients and the cold positive unflooding effect will be avoided by preheating the core to a water temperature of about 545°F before unflooding or rod withdrawal are initiated.

2.2 Figure II. 13. a illustrates k_{eff} (control rods withdrawn) as a function of water temperature for both the flooded and unflooded conditions. Figure II. 13. b illustrates the effect on k_{eff} of fully inserting the control rods. These figures show that the sign and magnitude of the temperature coefficient is strongly dependent upon control rod conditions within the core. A core that is just critical (rods partially inserted) will have a temperature effect curve between these two extremes. Figure II. 13. c is such a plot for the VSR with the four central control rods fully inserted and the outer rods withdrawn. Figures II. 13. a, b, and c include the effects of Doppler broadening of the U-238 resonances and the enhanced leakage effect ("Behrens' correction") for the void steam channels in the unflooded condition.

2.3 Table II. 5 illustrates the full-core temperature coefficients. If the rods are withdrawn uniformly ("ganged"), creating a "pancake" critical region (small core) with high leakage, the temperature coefficient will be more negative than if alternate rods are withdrawn ("dispersed" rod pattern) such that the effective critical volume approaches that of the full core. The effect of critical core size on flooding reactivity effects, discussed in paragraphs 2.12 and 2.13 of this section, is applicable in explaining the effect of core size on moderator temperature coefficients also.

2.4 The void coefficient calculations for this core are based on the assumption that voids are produced only within the fuel channels. If voids were formed in the spaces between channels, the resulting reactivity effects would be positive. However, since the nuclear heating is predominantly through fissions in the fuel rods, the initial voids for any power level increase would always be within the channels. Under normal operating conditions, there is no net steam formation outside the channels.

TABLE II. 3

LATTICE PARAMETERS - CONTROL RODS OUT

	Cold, 68°F		Hot, 545°F		Hot, 545°F*
	Unflooded	Flooded	Unflooded	Flooded	Unflooded
τ	39.97	30.57	62.46	49.42	66.55
L^2	3.22	2.82	6.50	5.63	6.82
M^2	43.19	33.39	68.96	55.05	73.37
p (U-238)	0.899	0.898	0.865	0.865	0.857
k_∞	1.195	1.153	1.257	1.229	1.258
$B^2{}_g$	0.002025	0.001974	0.001872	0.001875	0.001896
k_{eff}	1.099	1.082	1.113	1.114	1.104
ℓ	4.93×10^{-5}	4.99×10^{-5}	4.75×10^{-5}	4.73×10^{-5}	4.75×10^{-5}

* Avg. core voids = 6%; includes Doppler correction for an average fuel temperature of 1250°F.

Legend:

τ	= Fermi age, cm^2
L^2	= Thermal diffusion area, cm^2
M^2	= Migration area, cm^2
p (U-238)	= Resonance escape probability
k_∞	= Multiplication constant, infinite lattice
$B^2{}_g$	= Geometric buckling, cm^{-2}
k_{eff}	= Multiplication constant, finite lattice
ℓ	= Neutron lifetime, sec

TABLE II. 4

REACTIVITY COEFFICIENTS

Temperature Coefficient, $\Delta k/k$ per °F increase:	
Cold, clean core, unflooded, control rods out	$+0.9 \times 10^{-4}$
Hot, clean core, unflooded, control rods out	-0.6×10^{-4}
Void Coefficient, $\Delta k/k$ per 1% void increase in moderator:	
Cold, clean core, unflooded, control rods out	Zero
Hot, clean core, unflooded, control rods out	-1.0×10^{-3}
Doppler Coefficient, $\Delta k/k$ per °F increase in fuel temp.	-0.7×10^{-5}
Cold, clean core	
Maximum Flooding Effects, % $\Delta k/k$	
Cold, unflooding, control rods dispersed	+ 1.3
Hot, flooding, control rods ganged	+ 0.7
Maximum Pressure Coefficient Effect, % $\Delta k/k$	
(Collapse of 6% voids at 545°F with control rods out)	+ 0.5

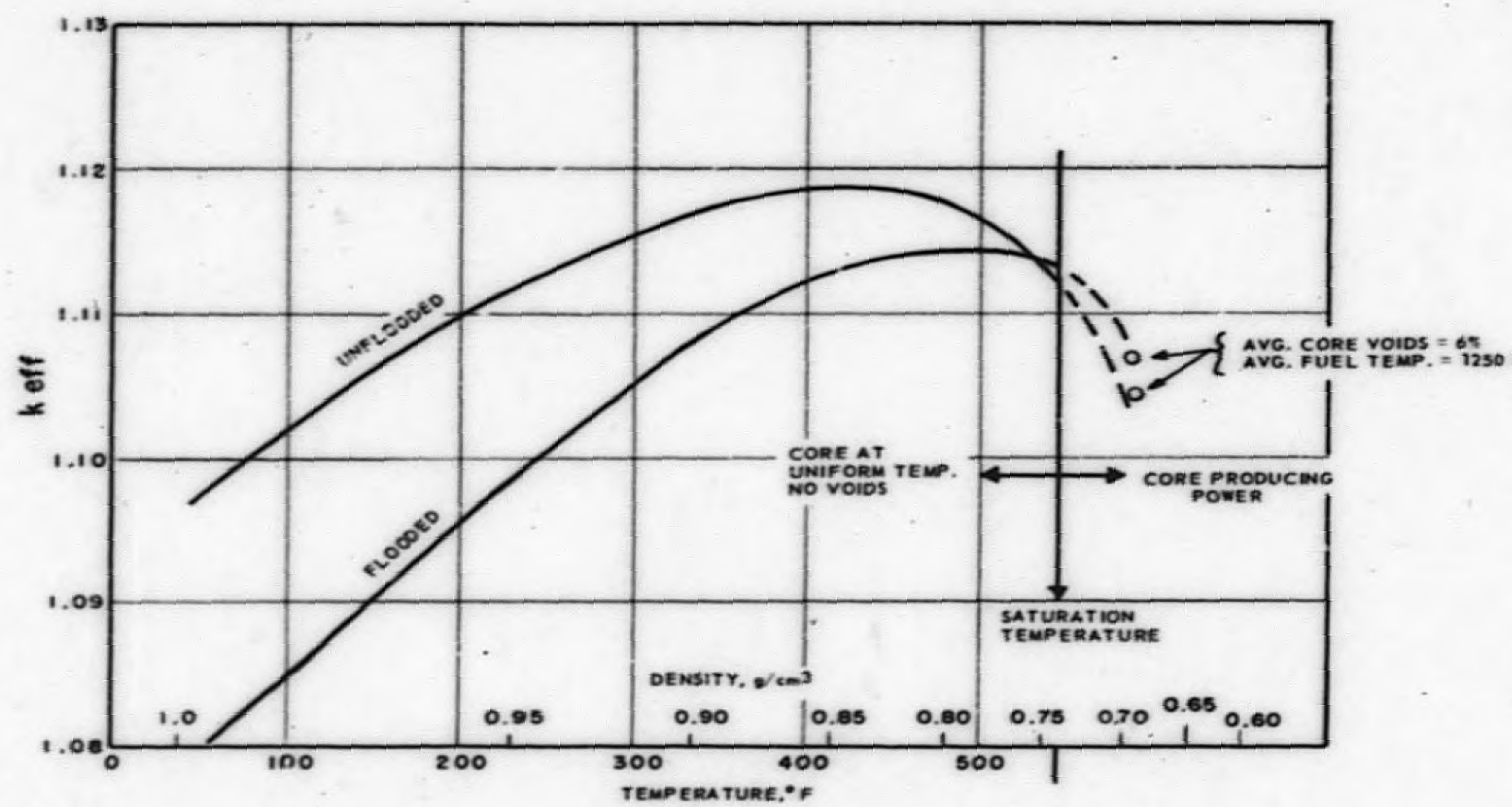


FIGURE II.13.a

k eff versus TEMPERATURE CONTROL RODS OUT

FIG. II. 13 a

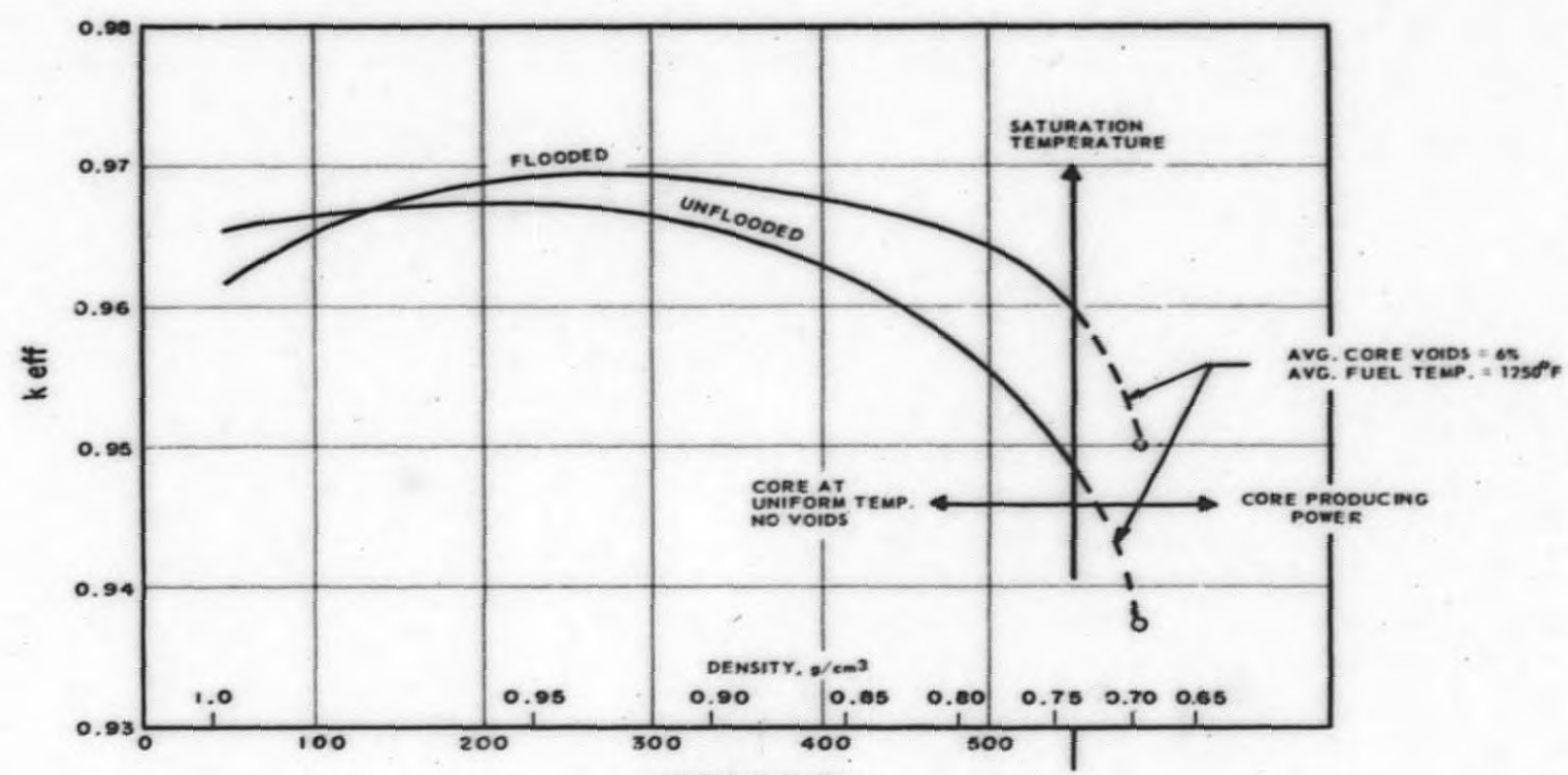


FIGURE II.13.b
k_{eff} versus TEMPERATURE CONTROL RODS IN

FIG. II. 13b

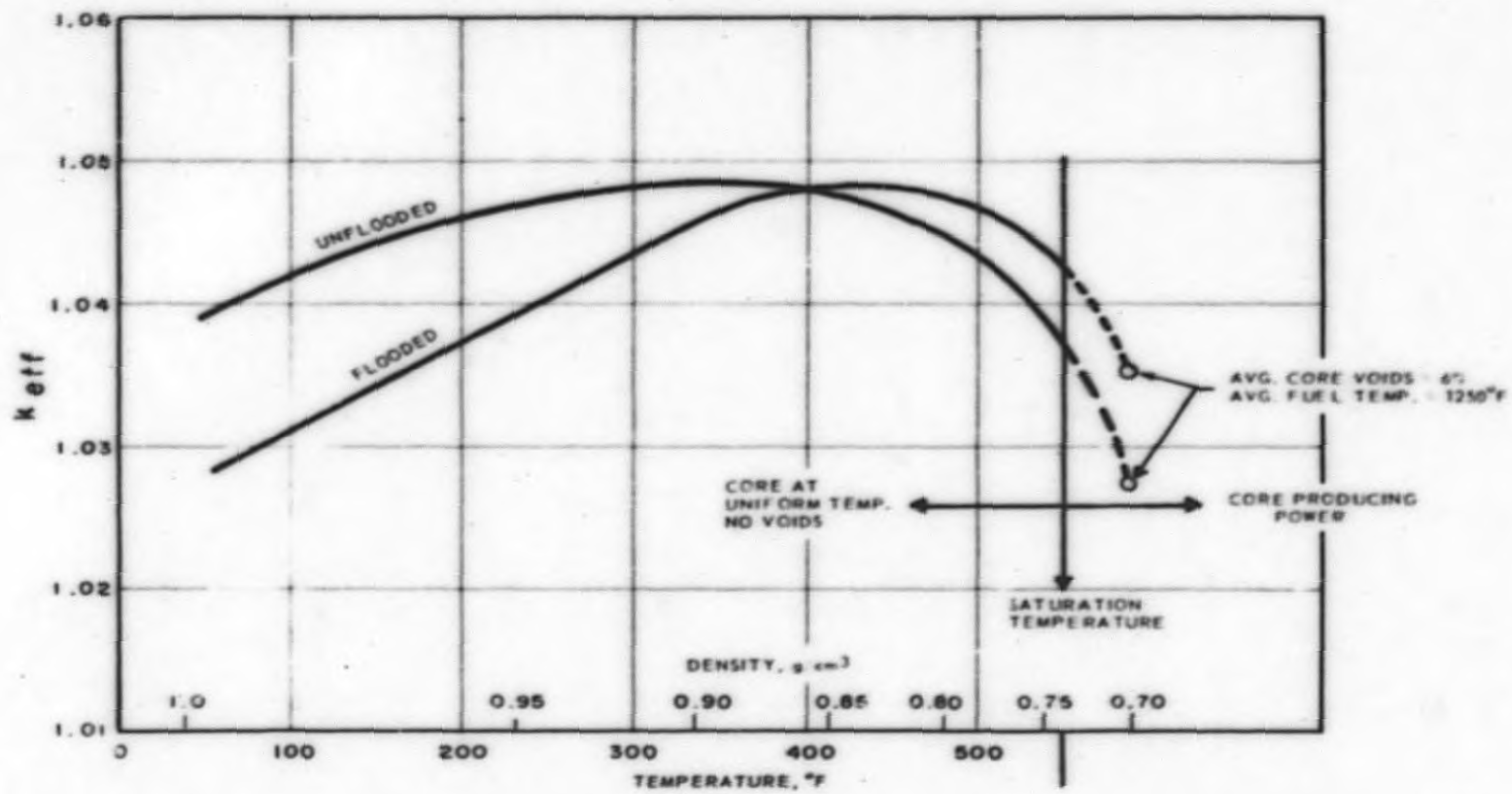


FIGURE II.13.c
k_{eff} versus TEMPERATURE, FOUR CENTRAL RODS IN, EIGHT PERIPHERAL RODS OUT

FIG. II.13c

TABLE II. 5

MODERATOR TEMPERATURE COEFFICIENTS

($\Delta k/k$ per °F increase)

	Cold, 68°F		Hot, 545°F	
	Flooded	Unflooded	Flooded	Unflooded
Clean Core Control Rods In	$+0.4 \times 10^{-4}$	$+0.2 \times 10^{-4}$	-0.6×10^{-4}	-0.9×10^{-4}
Clean Core Control Rods Out	$+1.2 \times 10^{-4}$	$+0.9 \times 10^{-4}$	zero	-0.6×10^{-4}

2.5 The effective void coefficient is the summation of many effects. When void formation occurs, the hydrogen content of the system drops sharply with several consequences:

2.5.1 Fewer neutrons are scattered past the capture resonances in U-238 so that more non-fission absorption occurs.

2.5.2 Fast neutrons have a greater probability of interacting in other rods so that fast fission increases.

2.5.3 Both τ and L^2 increase so that the epithermal and thermal captures in control rods increase, since more neutrons will reach the rod surfaces. There is also an increase in core leakage.

2.5.4 Reduction of the hydrogen absorption increases the ratio of fission absorptions to total absorptions.

2.5.5 The power distribution reshapes itself, falling in the regions where voids are produced and rising in the non-void regions.

2.6 Items 2.5.1 and 2.5.3 produce negative contributions to the void coefficient, but items 2.5.2 and 2.5.4 are positive effects. Item 2.5.5 is not actually a positive effect, but by decreasing the "worth" of the neutrons in the void regions, it detracts from the potential total negative effect.

2.7 As indicated in Table II.4, the void coefficient may be positive with all control rods withdrawn in the cold reactor condition, but becomes negative below the normal operating temperature. The reactor is preheated to the hot condition while still subcritical as described in paragraph 2.1 of this section. It should be noted that the void coefficient is a function not only of temperature but of void content as well. Thus, creation of moderator voids in the cold condition would add reactivity only up to a particular void content beyond which additional void formation would cause negative reactivity effects. The void coefficient calculations are made assuming the superheat steam passages of the core within the process tubes are unflooded. This is appropriate since any power level sufficient to create voids in the moderator would cause prior voiding of the superheat steam passages. The reactivity effects of voiding the steam passages are considered separately from the void coefficient of the moderator, and are discussed in later paragraphs about the "flooding" and "unflooding" characteristics of the reactor.

2.8 The negative Doppler fuel temperature effect results from the dependence of the neutron absorption in the U-238 capture resonances on the relative motion of the U-238 nucleus, which is a function of the temperature of the UO_2. The increased relative velocity of the U-238 nucleus as the temperature increases makes the resonance peaks appear broader so that non-fission neutron absorption increases, producing a negative reactivity effect. This is one of the principal self-shutdown mechanisms for the VSR and has the added important characteristic that it is instantaneous, whereas the moderator temperature and void effects (exclusive of prompt gamma neutron heating) depend on transferring heat through the low conductivity UO_2 and the steam passages, resulting in a delay of several seconds between a power rise and corresponding rise in water temperature and void formation. There is an interesting "ratchet" effect here, in that while a temperature rise can be produced suddenly in the fuel by a step change in power, the temperature cannot fall suddenly because of the insulating property of the steam annulus and the low thermal conductivity of UO_2. This effect will tend to damp out reactivity oscillations.

2.9 The rate of change of reactivity with fuel temperature increases as the fraction of neutrons captured in the resonances increases. The Doppler coefficient thus tends to be higher at operating conditions due to the reduction in water density. This effect is counteracted by the fact that the Doppler coefficient decreases as the fuel temperature increases. The net result is that the Doppler coefficient remains relatively constant from cold to operating conditions, as seen in Table II. 6. However, under conditions of a fast transient, during which the moderator temperature remains relatively constant, the Doppler coefficient drops with fuel temperature as shown in Figure II. 14. The coefficient is inversely proportional to the square root of the absolute temperature of the fuel when the moderator temperature is fixed.

2.10 Another important facet of the Doppler coefficient is that it does not change appreciably with time since the U-238 content of the reactor remains essentially constant throughout core life. The temperature and void coefficients do decrease with time because of their dependence on control rod effects, reaching their most positive values at the end of core life when the control rods are near the fully withdrawn positions.

2.11 It is necessary to flood the superheat passages of the core as a means of reducing radiation to personnel performing refueling operations or other core maintenance. Accordingly, it is necessary as a part of the startup operation to unflood ("blow-out") these passages. The calculated reactivity effect of unflooding the superheat passages at room temperature is shown in Table II. 7. Table II. 8 illustrates the unflooding and flooding effects at a water temperature of 545°F. From the two tables it is seen that the reactivity effect of cold unflooding is positive while the effect of hot unflooding is negative. These trends may also be seen graphically on Figures II. 13. a, b, and c.

2.12 The changes in the flooding effects due to differences in control rod patterns, initial reactivity, superheated steam volume and moderator volume can be investigated using a simplified model. Employing a one-group approximation, the effective multiplication constant can be written:

$$k_{eff} = \frac{C\, k_{\infty}}{1 + B_g^2 \cdot M^2}$$

where the "C" term represents the effect of the control system and is less than or equal to 1.0. Differentiating the above expression, one obtains an expression for determining order of magnitude estimates of $\Delta k_{eff}/k_{eff}$ due to flooding.

TABLE II. 6

DOPPLER COEFFICIENTS

($\Delta k/k$ per °F Increase in Fuel Temp.)

Fuel Temperature	Moderator Temperature	$\Delta k/k$ per °F
68°F	68°F	-0.7×10^{-5}
1250°F	545°F	-0.5×10^{-5}

FIG. II . 14

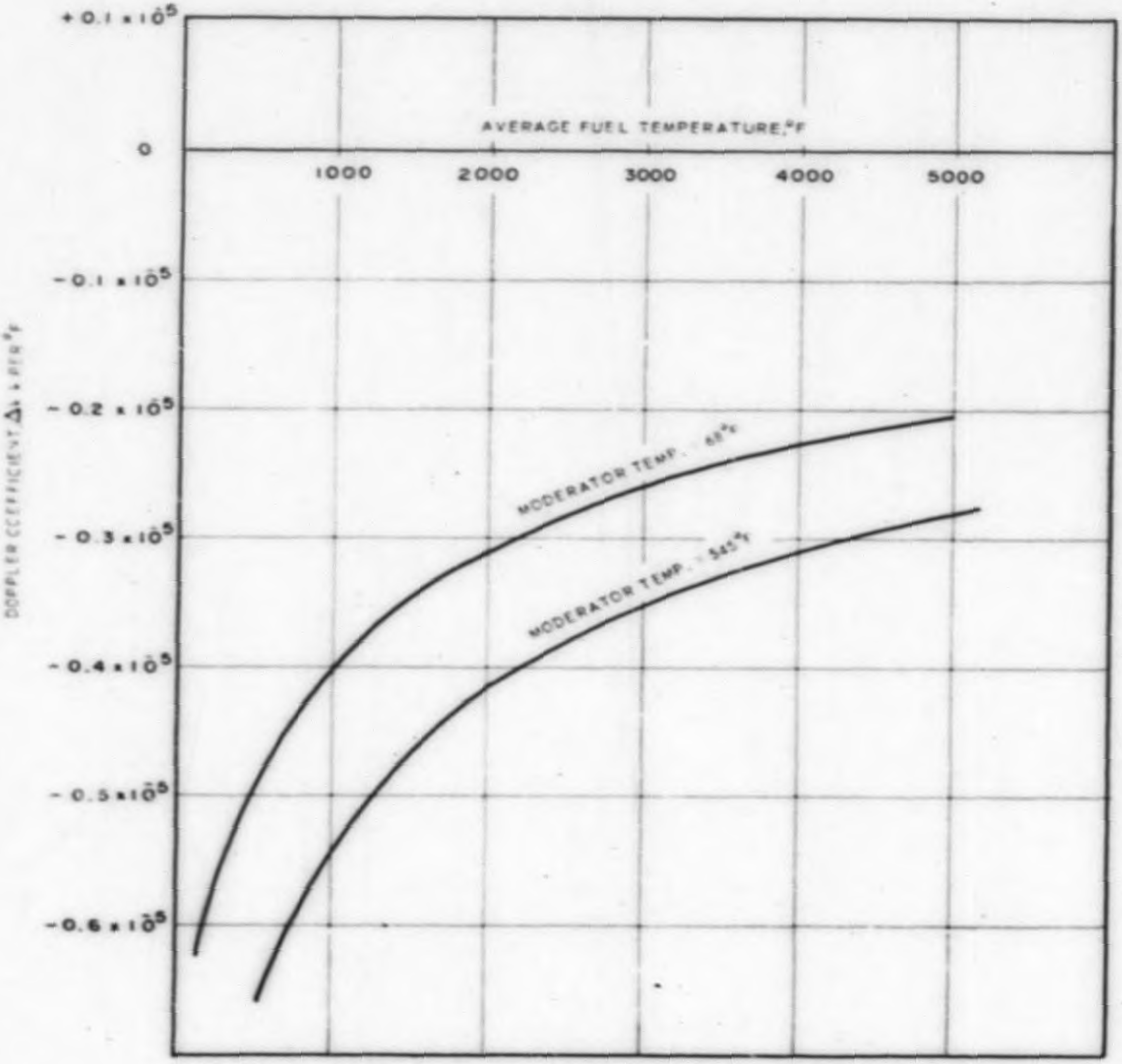


FIGURE II.14

DOPPLER COEFFICIENT versus FUEL TEMPERATURE

TABLE II. 7

COLD, UNFLOODING REACTIVITY EFFECT

	% $\Delta k/k$
Control rods ganged*, clean core	+ 0.2
Control rods dispersed, dirty core	+ 1.3
Control rods in, clean core	+ 0.4

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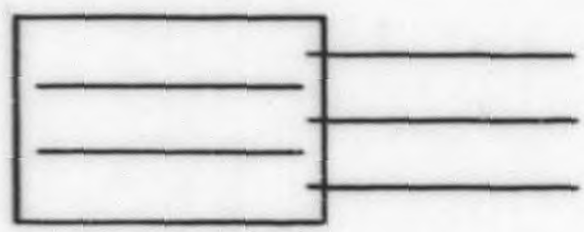
*

$k_{eff} = 1.00$



Rods ganged: $B_g^2 = 0.005\ cm^{-2}$ **Rods dispersed:** $B_g^2 = 0.002\ cm^{-2}$

TABLE II.8

HOT UNFLOODING AND FLOODING REACTIVITY EFFECTS

	$\% \Delta k/k$
Unflooding, control rods in, clean core	- 1.3
Flooding, control rods ganged, clean core	+ 0.7
Flooding, control rods dispersed, dirty core	+ 0.1

$$\frac{\Delta k_{eff}}{k_{eff}} = \frac{\Delta k_{\infty}}{k_{\infty}} = \frac{\Delta (B_g^2 \; M^2)}{1 + B_g^2 \; M^2} + \frac{\Delta C}{C}$$

2.13 When the core is flooded, the hydrogen content of the lattice increases sharply with the following principal consequences:

2.13.1 Thermal utilization (f) decreases. The increase in hydrogen absorption decreases the ratio of fission absorptions to total absorptions. (Negative effect.)

2.13.2 Resonance escape (ρ - U^{238}) increases. The increased ratio of N^H/N^U results in fewer neutrons being absorbed by the U-238 resonances. (Positive effect.)

2.13.3 Migration area (M^2) decreases. A decrease in core leakage increases reactivity. The change in M^2 effectively changes the core reflector worth (B^2). This leakage effect is greater in the rods "ganged" case than for rods "dispersed" due to the difference in B_g^2 (sketch, Table II.7). (Positive effect.)

2.13.4 Further, for critical regions with control, the epithermal and thermal captures decrease since fewer neutrons reach the control rod surfaces. This positive effect is minimized by the use of "gray" rods in the VSR as compared to "black" rods.

2.14 In the cold case, the reactivity effect due to a change in thermal utilization (f) predominates giving positive unflooding; whereas, in the hot case the major effect is due to the change in migration area (M^2) resulting in positive flooding.

2.15 The following steps summarize ways which have been used to minimize the reactivity effects due to unflooding or flooding the superheat passages of the VSR core.

2.15.1 Minimize the volume occupied by the superheat passages. This step minimizes ΔM^2 and Δk_{∞} caused by flooding or unflooding.

2.15.2 Minimize k_{∞}. A reduction in k_{∞} results in a smaller B_g^2 which is a multiplier of the ΔM^2 due to flooding. (The value of B_g^2 remains relatively constant between the flooded and unflooded condition so that $\Delta (B_g^2 \; M^2) \approx (B_g^2 \; \Delta M^2)$.

2.15.3 Adjust the water-to-fuel ratio to optimize the values of the positive hot flooding effects against the cold unflooding effects and the void and temperature coefficients. An increase in water-to-fuel ratio decreases the hot flooding reactivity but causes the cold unflooding effects and the moderator coefficients to be less negative. Also, increasing the water-to-fuel ratio reduces the neutron economy of the plant. Accordingly, any change in the water-to-fuel ratio must be carefully weighed against the disadvantages of the change.

2.16 In boiling water lattices, it is necessary to determine the positive reactivity effect of a sudden collapse of voids due to a significant increase in pressure. The relatively small amount of voids (6% average) in the VSR lattice, together with a relatively small negative void coefficient results in a maximum reactivity change of less than + 0.5% $\Delta k/k$ for the sudden collapse of all voids in the core. This is indicated as the maximum pressure coefficient effect in Table II.4.

3. Reactivity and Control Requirements

3.1 The Mark I fuel is enriched to yield an average core lifetime of 5,000 Mwd/T. In addition, the core must have sufficient reactivity to allow for the formation of xenon and samarium. Table II.9 illustrates the core reactivity balance in the hot condition, and the calculated values of k_{eff} for a number of hot and cold conditions.

3.2 Referring to Table II.9, it is seen that the control system has been designed so that the reactor will remain subcritical for all conditions even if the strongest control rod were to be completely removed from the core. Table II.10 illustrates the calculated values for the maximum control rod worths, together with their associated core and rod conditions.

4. Neutron Flux and Power Distribution

4.1 Figure II.15 illustrates the relative power distribution by fuel rod and bundle for one-fourth of the core, with control rods out. Figures II.16.a and II.16.b illustrate the calculated power distributions within the bundles in the x-y plane. For calculational purposes, the fuel, clad, water and steam were considered to be homogeneously mixed with appropriate self-shielding corrections included for thermal flux heterogeneity.

4.2 The average power density within the hottest rod for the case illustrated by Figure II.15 is seen to be 1.87 times the core average for the case with control rods withdrawn. Applying an axial peaking factor of 1.41 gives an absolute peak power generation of 2.6 times the core average with control rods withdrawn (clean core). This is higher than expected for a realistic case with rods withdrawn (end of life, dirty core) since non-uniform fuel burn-up will lower the 1.87 gross radial factor. Figure II.17 shows a more realistic relative power distribution in the x-y plane for a just-critical reactor in which the central four control rods are inserted to flatten the gross radial power.

4.3 Figure II.18 illustrates the gross radial flux distribution for the three energy groups of neutrons used in the diffusion theory analysis of this core. The following tabulation illustrates the average flux by group within the reactor core:

Fast neutron flux (10 Mev to 5.53 kev)	1.43×10^{13} nv
Resonance neutron flux (5.53 kev to 0.625 ev)	6.81×10^{12} nv
Thermal neutron flux (less than 0.625 ev)	8.59×10^{12} nv

TABLE II.9

CORE REACTIVITY

	% Δk/k
Hot reactivity balance:	
Lifetime effects*	.07
Xenon and samarium poisoning	.04
TOTAL Δk/k REQUIRED	.11

* U-235 depletion, Pu buildup, long-term fission product poisoning.

k_{eff}, hot:	
Clean, unflooded, control rods out	1.11
Clean, flooded, control rods out	1.11
Maximum, control rods in, flooded	0.96
Maximum, one stuck control rod, flooded	0.99
Total controls worth, unflooded	15% Δk/k
Total controls worth, flooded	14% Δk/k
k_{eff}, cold:	
Clean, unflooded, control rods out	1.10
Clean, flooded, control rods in	1.08
Maximum, all control rods inserted, unflooded	0.96
Maximum, one stuck control rod, unflooded	0.99
Total controls worth, unflooded	12% Δk/k
Total controls worth, flooded	11% Δk/k

TABLE II.10

MAXIMUM CONTROL ROD WORTH*

	% $\Delta k/k$
Cold, unflooded, one central control rod (remaining 11 rods fully inserted)	2.6
Hot, unflooded, 6% void, rod No. 9, (Nos. 4, 6, 7, 11, 12 withdrawn) (Nos. 1, 2, 3, 5, 8, 10 inserted)	3.2

* Control rod numbers are illustrated on Figure II.12.

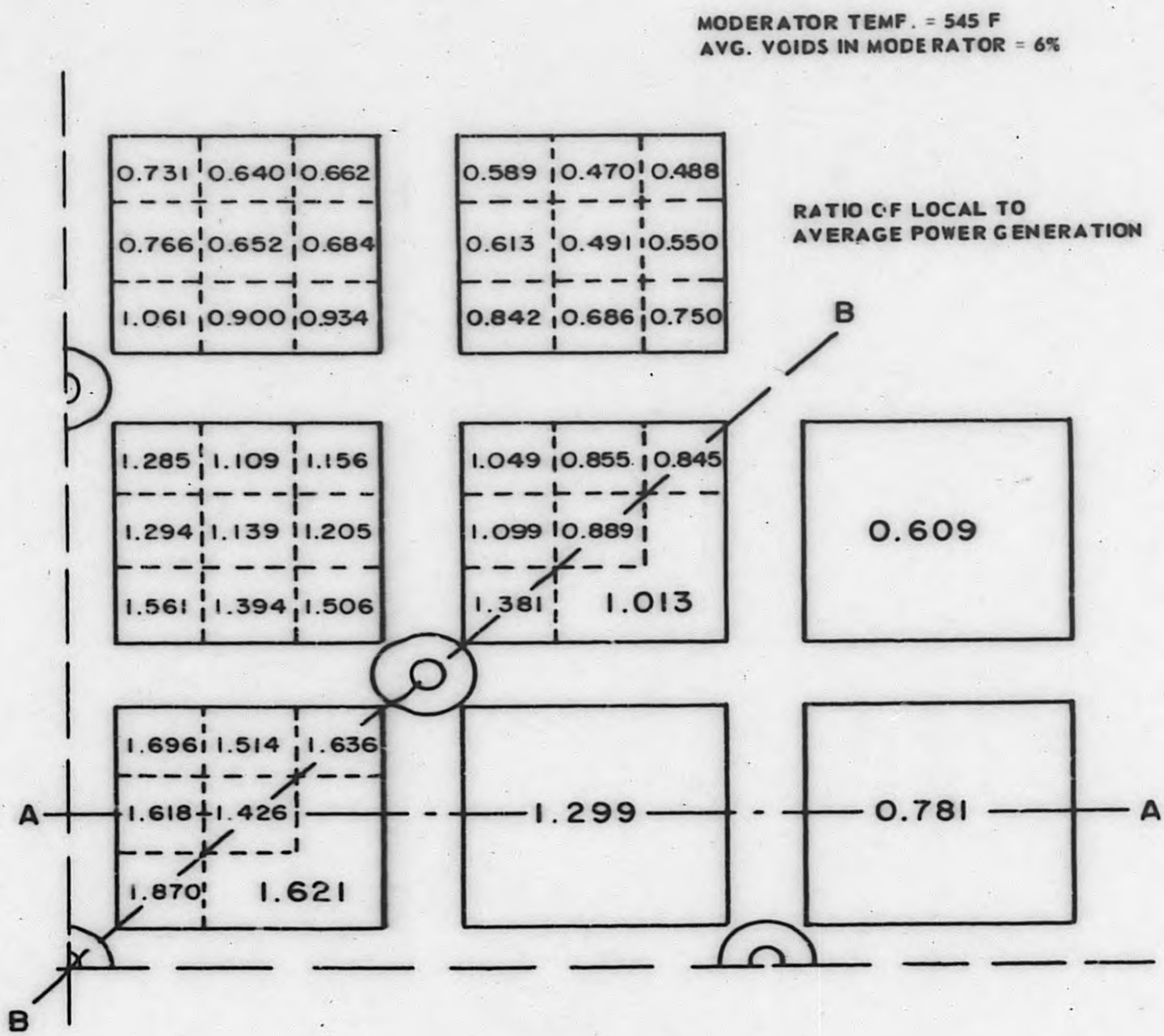
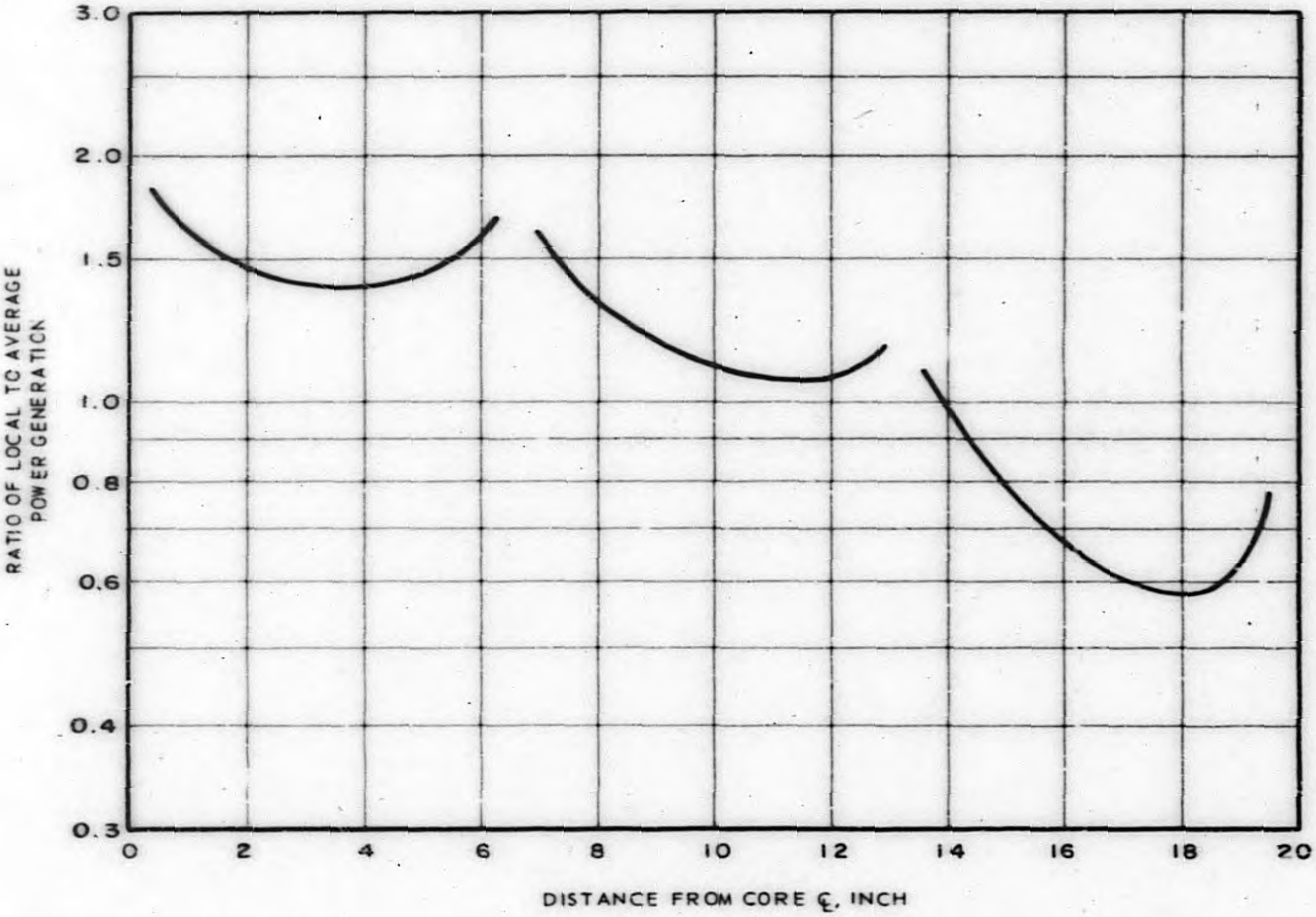
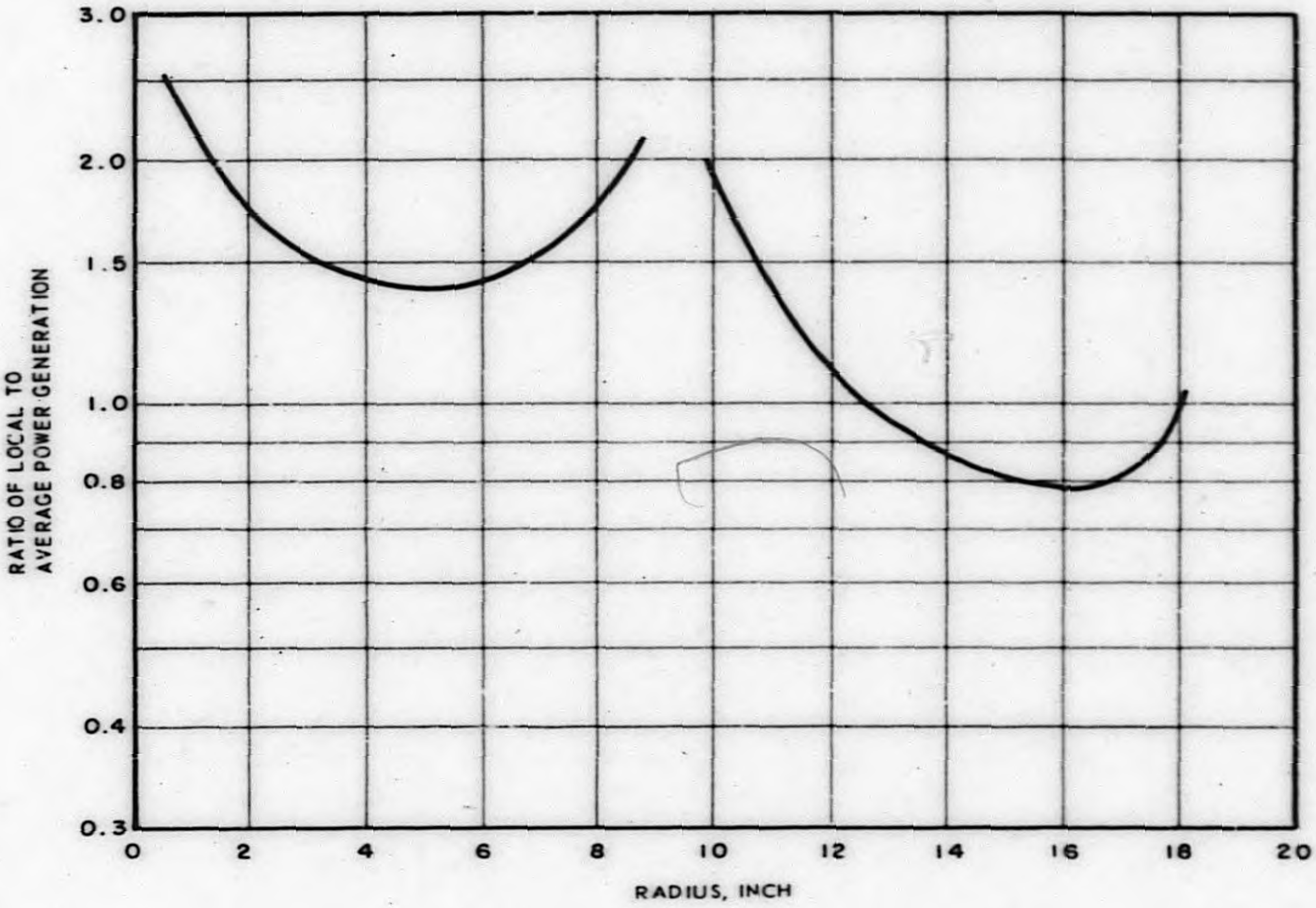


FIGURE II.15
RELATIVE POWER GENERATION, CONTROL RODS OUT



*SEE FIGURE II.15
MODERATOR TEMP. = 545°F
AVG. VOIDS IN MODERATOR = 6%

FIGURE II.16.a
HORIZONTAL POWER DISTRIBUTION TRAVERSE
X-Y PLANE ALONG A-A*



*SEE FIGURE II.15
MODERATOR TEMP. = 545°F
AVG. VOIDS IN MODERATOR = 6%

FIGURE II.16.b
DIAGONAL POWER DISTRIBUTION TRAVERSE
X-Y PLANE ALONG B-B*

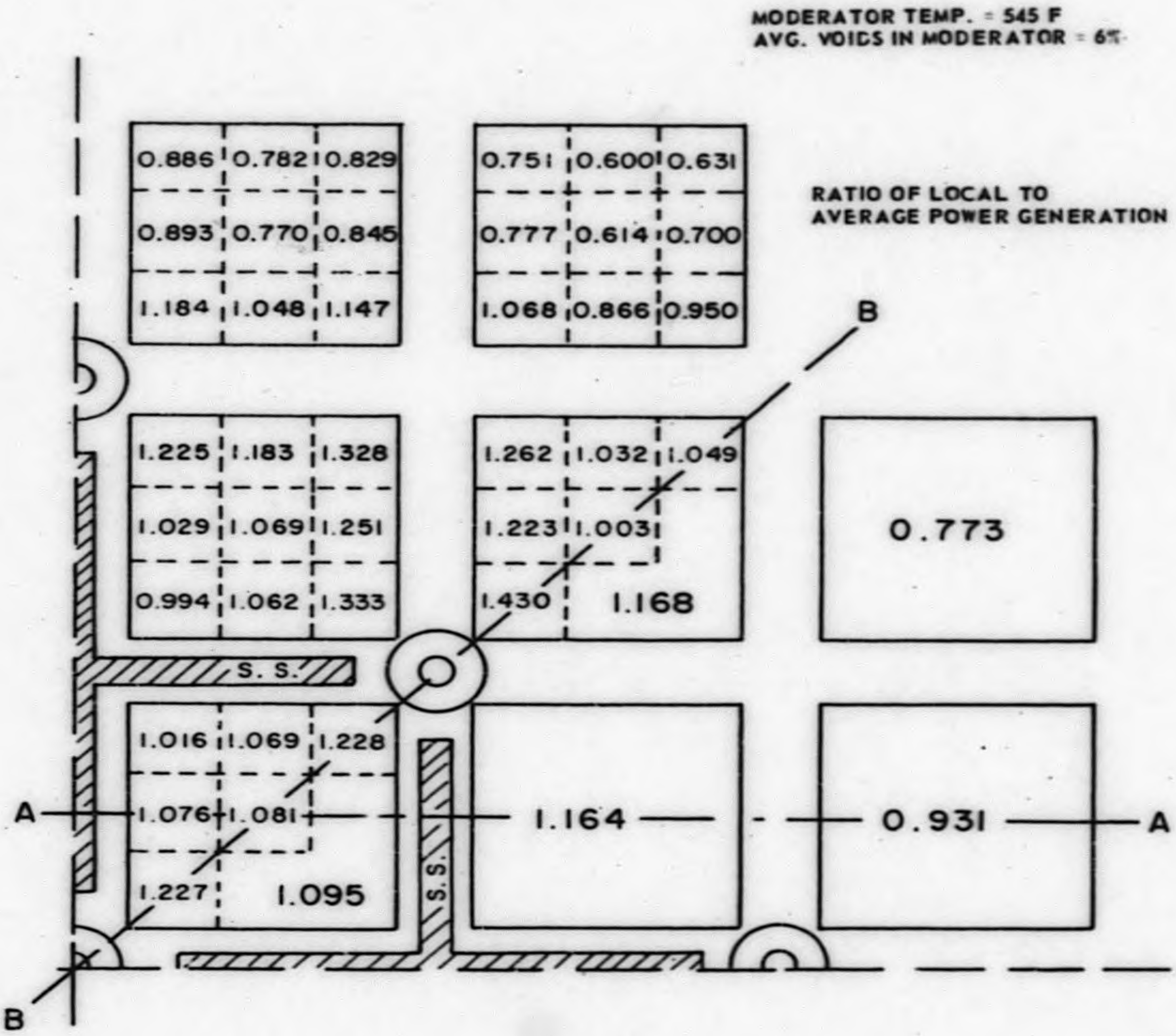


FIGURE II.17
RELATIVE POWER GENERATION, FOUR CENTRAL CONTROL RODS IN

FIG. II. 18

MODERATOR TEMPERATURE = 545 F
AVERAGE VOIDS IN MODERATOR = 6%
THERMAL POWER = 12.5 MW

GROUP	ENERGY	AVG. FLUX, N/CM^{2}SEC.
1	10 mev $\geq$ E $\geq$ 5.53 x 10^3ev	1.43 x 10^{13}
2	5.53 x 10^3 ev > E $\geq$ 0.625 ev	6.81 x 10^{12}
3	0.625 ev > E $\geq$ THERMAL	8.59 x 10^{12}

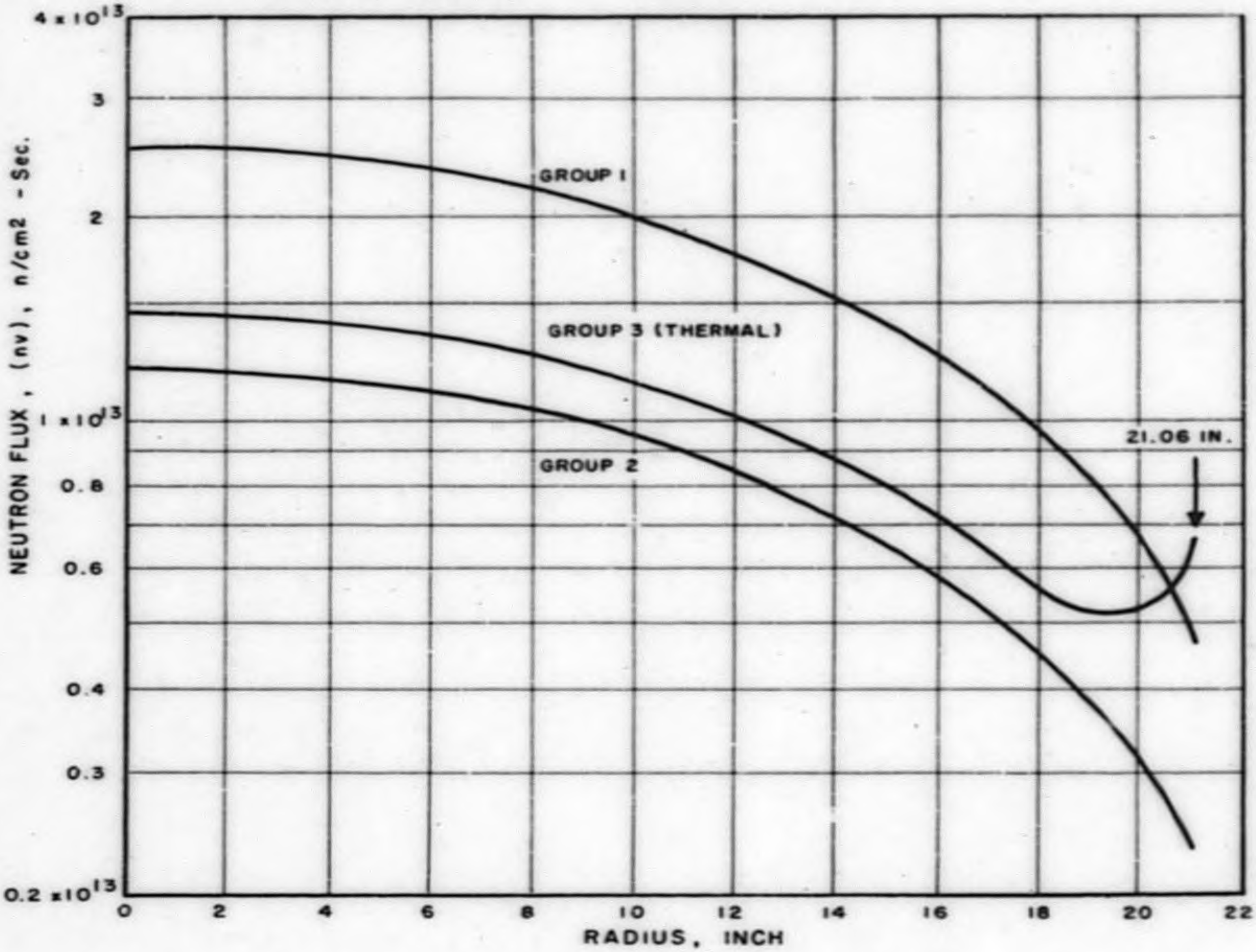


FIGURE II.18
RADIAL FLUX DISTRIBUTION IN EQUIVALENT
CYLINDRICAL CORE

5. Effects of Irradiation on Core Characteristics

5.1 As fuel irradiation progresses, a number of the core characteristics undergo significant change. The reactivity of the fuel changes as the initial U-235 is burned and plutonium and fission products are produced. Because of the initial enrichment level involved and the low conversion ratio, the reactivity in the VSR always decreases with exposure. The control rods are withdrawn to compensate for the gradual decrease in reactivity. The power distribution varies as a result of the control rod movement. The void distribution varies with the change in power distribution.

5.2 Within the fuel bundles, the rods near the periphery of a bundle are irradiated to higher exposure than the center rods because of the higher neutron flux in this area. Therefore, the local power peaking in the fuel bundles will tend to be reduced with burnup due to the more rapid depletion of U-235 in the locations near the water gaps. The overall power distribution across the core also will tend to become flatter with exposure because of the more rapid burnup of fuel in the central, higher neutron flux regions. This flattening will be partially over-ridden by withdrawal of control rods in the high flux regions to compensate for burnup.

5.3 The reactivity coefficients of moderator temperature and voids, and cold unflooding, become less negative with exposure primarily due to the removal of control rods from the core. The increase in control rod worth with an increase in void or temperature is an important effect which contributes to the reactivity coefficients. Because of the relatively high enrichment in U-235 the buildup of plutonium in the fuel will have a negligible effect on these coefficients.

5.4 In the initial operating condition, the effective delayed neutron fraction including the contribution of U-238 fissions is 0.007. Because of the smaller delayed neutron fraction which is characteristic of plutonium, the delayed neutron fraction will decrease to 0.0065 with irradiation.

6. Nuclear Stability

6.1 The reactor will obtain the major part of its saturated steam supply from an external source. Accordingly, a relatively small volume percent of the moderator will be occupied by saturated steam. Figure II.19 illustrates an approximate axial distribution of the average radial steam volume percent. The average steam volume percent inside the channels is about 6%, which is quite low compared to that in boiling water reactors. The reactivity represented by the steam bubbles in the core is approximately 0.5%. This amount of reactivity is well below the void reactivity of many operating boiling water reactors which have exhibited good stability, and is therefore expected to be well below that which may be safely tolerated without incurring any tendency towards "chugging" instability in the VSR. Thus, the boiling moderator in the VSR should not produce any undesirable nuclear effects.

6.2 Xenon-induced oscillations in the reactor power distribution can exist only in reactors that operate at high thermal flux levels and have cores with a linear dimension whose square is much greater than the neutron

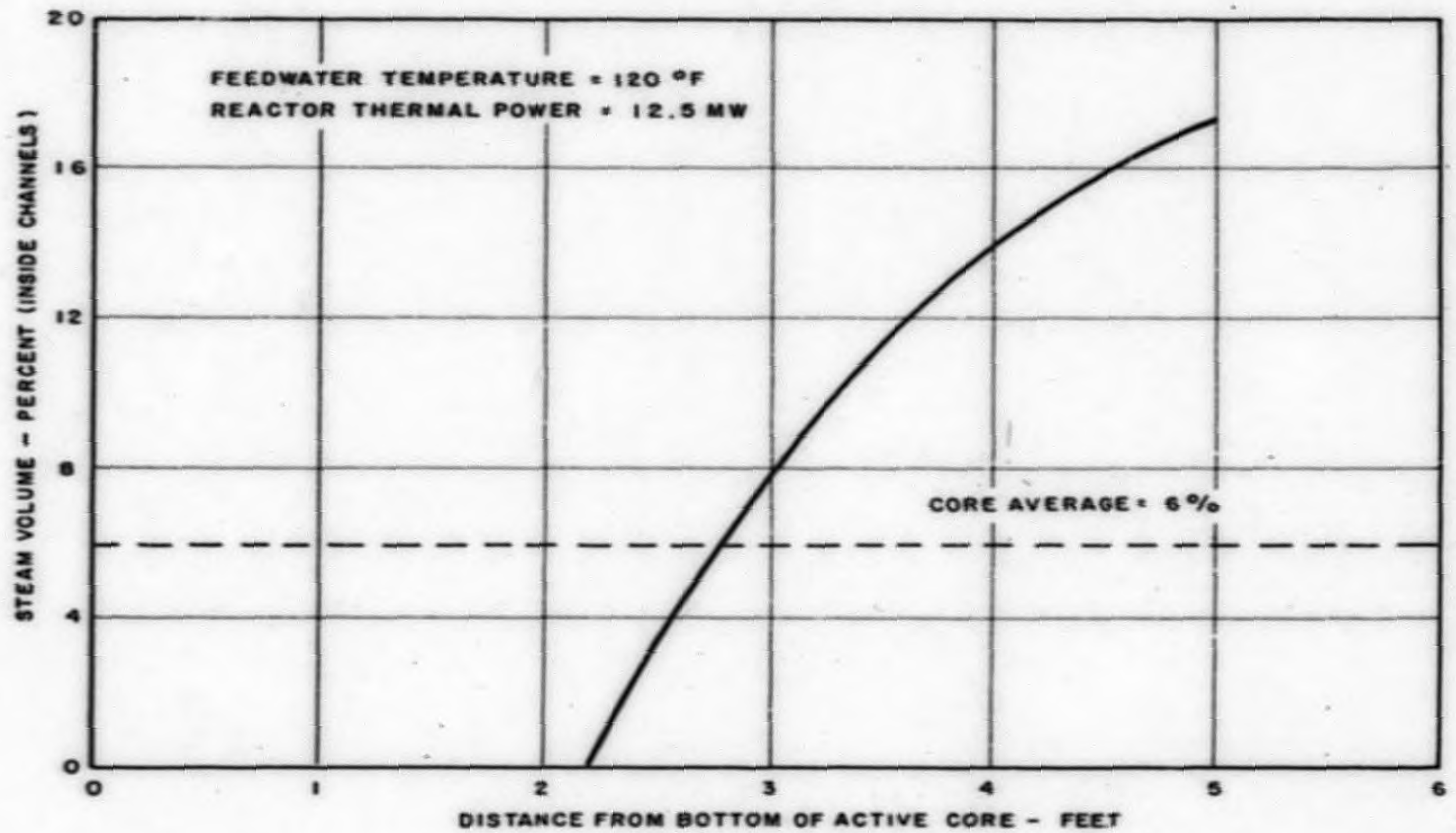


Figure II. 19

Saturated Steam Volume In Core

FIG. II. 19

migration area (by at least a factor of 1000[1]). Such oscillations have been observed in the PWR at Shippingport and in the heavy water moderated reactors at Savannah River. However, the core of the VSR is small. The square of the linear dimension of the VSR core is only approximately 250 times the equivalent migration area. Therefore, no xenon instability difficulty is expected. This theory is borne out by other small reactors, like EBWR and VBWR, whose stability in this respect has been established by a great deal of operational experience.

6.3 Apart from the core size, analytical studies on boiling water reactors show a tendency for voids to exert a damping effect on any xenon oscillations. Such oscillations, if they were to occur (for example, due to improper movement of control rods by the operator), would have a long period and could be stopped by local adjustment of control rods.

6.4 The principal effects upon the VSR of transients or changing conditions in the VBWR would be those associated with inlet steam flow and pressure. Since the inlet steam flow will be pressure-regulated between the VBWR and VSR, no possibility of inducing instabilities in the VSR from changes in the VBWR is foreseen. The more significant effect on VSR of changes in VBWR will be non-nuclear, e.g., reduction of inlet steam flow when VBWR power is reduced.

7. Reactor Shielding

7.1 To allow for future power increases, the shielding analysis has been based on a reactor thermal power of 30 Mw. At this power level, four feet of heavy concrete (density of 225 lb/ft^3) or six feet of ordinary concrete (density of 150 lb/ft^3) is required above the reactor vessel top headfor neutron and gamma shielding. This top shield plug will be offset sufficiently to prevent direct streaming of radiation around its periphery. A side shield thickness of 7-1/2 feet of ordinary concrete will be required to limit the dose rate to 30 mrem/hour.

7.2 Some shielding will be required below the reactor vessel bottom head. A 15-inch thickness of ferrophosphorous concrete (300 lb/ft^3) at this location would be adequate to limit radiation levels at full reactor power to about 100 mrem/hour in the sub-reactor room. This would permit limited access during operation, if required, and would keep shutdown radiation in this location to negligible levels. Because the control rod drives penetrate this lower shielding, clearances between the drive housings and shielding will be minimized to prevent radiation streaming. If, despite this precaution, neutron streaming should be excessive, it would be possible to fill the small remaining clearances with polyethylene strips to correct the situation.

7.3 For the 42-inch equivalent diameter core operating at 30 Mwt, the fast neutron flux ($>$ 0.2 Mev) at the inside radius of the reactor vessel wall at core midplane is about 1.6×10^9 nv. Over the 20-year design life of the vessel, this exposure could produce an integrated fast neutron flux of about 1×10^{19} nvt. This integrated flux is below the

(1) D. Randall and D. St. John "Xenon Spatial Oscillations" Nucleonics Vol. 16, No. 3, p. 82 (1958).

design limit (Section IV. D) of 5×10^{19} nvt, which is itself based on present knowledge of the vessel embrittlement effects of the fast neutron flux.

7.4 The maximum gamma heating in the vessel wall, which occurs also at the core midplane, produces less than a 10°F temperature differential across the wall. The attendant thermal stress is only about 1300 psi, so that no special thermal shield is required in this reactor.

7.5 The water shielding provided above and below the core limits head activations to low values. Activation of the top head will produce a dose rate of about 15 mr/hour, four hours after shutdown, when the head is viewed from the top. Negligible activation of the bottom head will occur, so that the largest source in that area during shutdown will be from corrosion product buildup at the bottom of the vessel.

8. Critical Experiments

8.1 Although there are no plans to mock-up the Vallecitos Superheat Reactor core in a special critical experiment prior to the release of final specifications, there are two critical testing programs from which pertinent experimental data will be forthcoming. These are (1) the AEC-sponsored Superheat Critical Program and (2) the Initial Loading and Critical Testing Program planned for the VSR during the startup operations.

8.2 The first program (AEC Superheat Critical) has been active since mid-October, 1960, and has as its primary objective the measurement of physics parameters and the correlation of these data with analyses for purposes of improving analytical techniques and obtaining a basic understanding of superheat lattice parameters. An annular fuel and process tube geometry similar to that used in the VSR is being used in this experimental program, and a wide range of water-to-fuel ratios and critical sizes is being studied. This critical experiment program will bracket or approach the VSR design point in terms of water-to-fuel ratio and assembly size. Temperature coefficients, void coefficients, flooding versus unflooding reactivity effects, controls worth, migration area, and resonance escape probability are some of the measurements being made over a range of lattice spacings. Significant amounts of data in these categories have already been taken. These have been correlated with analyses, and improvements have been made in analysis procedures on this basis. In general, the agreement between analysis and experiment is quite good, and based on the correlation of AEC Superheat Critical measurements to date it is felt that the physics analysis of the VSR design is on a sound and reliable basis.

8.3 The final confirmation of the VSR nuclear design (relating to thermal performance, reactivity coefficients, and control worths) will take place during the initial startup phase of the reactor. During this critical test program, the core will be loaded in a stepwise manner in the reactor vessel, with multiplication runs between steps. A "just critical" small core will be assembled prior to full core loading. This will provide a check on the physics predictions and a base point from which the full core excess reactivity may be calculated. The program of physics measurements at very lower power levels will include the items listed in the subparagraphs which follow.

8.3.1 The calibration of control rods and determination of excess reactivity.

8.3.2 The determination of fuel bundle worths.

8.3.3 Measurement of fuel and moderator temperature coefficients. The fuel and moderator temperatures will be varied independently in an attempt to separate their reactivity effects.

8.3.4 Measurement of void coefficients.

8.3.5 Neutron flux and power distribution measurements.

8.3.6 The reactivity effects of flooding and unflooding for several control rod patterns and moderator temperatures. The flooding and unflooding of the process tubes will be accomplished in each case with the reactor sub-critical. The reactor will then be made critical after pre-heating to the desired temperature. By these procedures, transient reactivity effects will be avoided while the reactor is critical.

8.4 The above measurements will provide important data for determining desirable modes of operation (control rod patterns) and for interpretation of full power test data (correlation of steam outlet temperature distributions with fuel bundle power distributions). Further, the critical testing program will validate the values of reactivity coefficients and rod worths used in the analyses of postulated accidents, thus lending credence to these analyses and to the evaluations of reactor safety based on them.

D. REACTOR HEAT TRANSFER AND FLUID FLOW

1. General Characteristics

1.1 This section describes the calculated thermal characteristics of the Mark I core. As indicated in Section II. B. 3, which describes the Mark I fuel bundle, only limited experimental evidence is currently available on this fuel design. However, tests in the SADE loop of the VBWR will provide substantial verification of the design prior to operation of the VSR. The present calculations are based on best estimates and extrapolation of what data do exist. It is anticipated that the Mark I core will prove to be conservative in design and provide a reliable basic core into which more developmental types of fuel bundles can be inserted for irradiation.

1.2 Figure II. 20 shows the VSR grid with the 32 fuel bundles divided into five zones with respect to heat generation and steam flow rate. The numbers within grid locations depict the zone assigned to that fuel bundle.

1.3 Table II. 11 presents the basic reactor parameters for operation of the Mark I core at a peak operating cladding temperature of 1250°F. Table II. 11 includes the parameters for individual fuel bundles in each of the five thermal zones defined on Figure II. 20. The selection of a 1250°F peak clad temperature results in a reactor thermal power level of about 12.5 Mw. It should be recognized, however, that the 1250°F value is selected somewhat arbitrarily and does not represent any firm thermal limit for the fuel. Since the criteria for establishing the maximum allowable surface temperature are not well defined, initial operation of the VSR will be at a lower peak temperature (and power), with step-wise increases up to the 1250°F, 12.5 Mwt condition. Since one of the purposes of the reactor is to investigate such things as realistic cladding temperature limits, it is anticipated also that further step-wise increases will be made to cladding temperatures above 1250°F and thermal powers above 12.5 Mwt on a planned experimental basis. The SADE loop experiments are expected to provide better data on cladding temperature limits for initial operation of the Mark I core.

1.4 The steam flow in each fuel bundle is orificed for the flow which will result in the desired surface temperature. In practice, this would be accomplished by varying the steam flow with outlet motor-operated valves and orifices. Thus, the outlet temperature in each type of bundle (by zone) is different. Experiments are now contemplated to confirm the ability to predict the surface temperature on the basis of outlet steam temperature and the heat generation. The present belief is that the surface temperature can be calculated accurately provided the power distribution is known. During initial operation, it is planned to measure the power distribution and then correlate this to the fuel element outlet temperatures measured in fuel bundles which are fully instrumented for this purpose.

FAR SIDE FROM
ENTRANCE TO CONTAINMENT

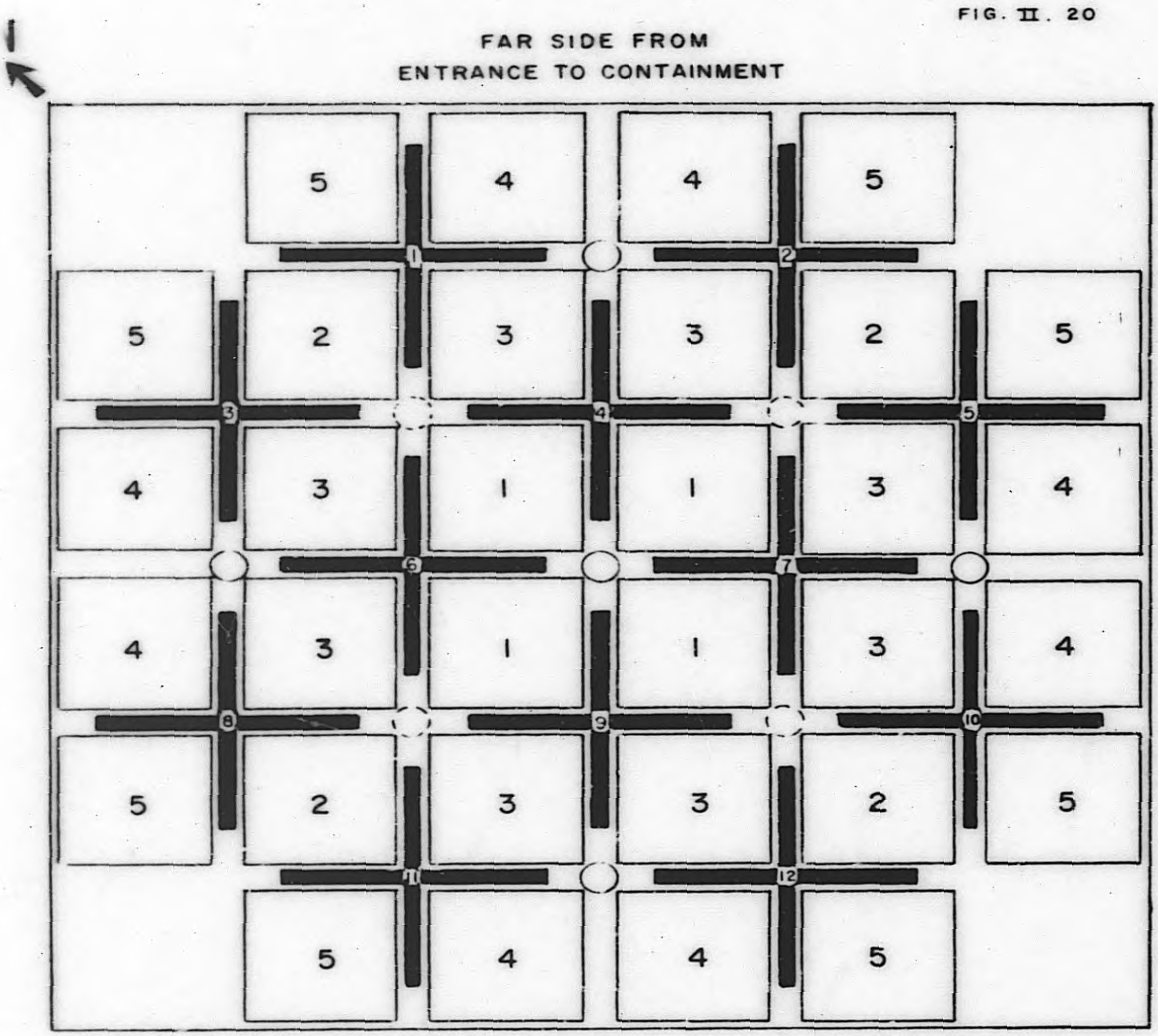


Figure II. 20

Thermal Zoning of Grid Locations

TABLE II. 11

REACTOR DATA SHEET

Overall Reactor Data

Reactor thermal power	Mwt	12.5
Power added to steam (net superheat)	Mw	7.7
Power added to moderator (heat transferred through process tube)	Mw	4.2
Power added to moderator (nuclear)	Mw	0.6
Steam flow from boiler	lb/hr	114,000
Steam flow from moderator	lb/hr	13,500
Steam flow through superheat passages	lb/hr	127,500
Steam inlet pressure (moderator pressure)	psig	960
Steam outlet pressure (at vessel wall)	psig	905
Core ΔP	psi	42
Inlet steam temperature	°F	541
Outlet steam temperature	°F (mixed mean)	810
Make-up water inlet temperature	°F	130
Moderator average exit quality	% by weight	1
Moderator average core quality	% by volume	6

Reactor Core

Number of fuel assemblies (bundles)		32
Number of elements per assembly		9
Total number of fuel elements		288
Number of control elements		12
Control rod pitch	inches	9.334
Circumscribed core diameter	inches	58.2
Equivalent core diameter	inches	42.1
Active core length	inches	60
Core length to equivalent diameter ratio		1.42
Total weight of UO_2 in core	lbs	5,300
Average UO_2 density (% of theoretical)	%	95
Fuel	Sintered UO_2	
Fuel geometry	Annular pellets - process tube	
Fuel cladding material	304 stainless steel	
Process tube O. D.	inches	1.45
Process tube wall thickness	inches	.025
Outer clad O. D.	inches	1.250
Outer clad thickness	inches	.028
Inner clad O. D.	inches	.625
Inner clad thickness	inches	.028
Fuel pellet O. D.	inches	1.190
Fuel pellet I. D.	inches	.629
Velocity booster O. D.	inches	.375
Velocity booster wall thickness	inches	.020
Outer flow annulus gap	inches	.075
Inner flow annulus gap	inches	.097

Table II. 11 (Continued)

Core Heat Transfer***

Heat transferred from fuel	Btu/hr	40,500,000
Total heat transfer area	ft^2	685
Average core heat flux	Btu/ft^2 hr	59,250
Maximum core heat flux*	Btu/ft^2 hr	225,000
Maximum to average core heat flux		3.8
Average heat generation in fuel	Btu/ft^3 hr	5,050,000
Maximum heat generation in fuel*	Btu/ft^3 hr	17,600,000
Maximum to average heat generation		3.5
Fuel volume	ft^3	8.01
Total core volume	ft^3 - (liters)	48.3-(1375)
Power density	kw/liter	9.07
	kw/kg	5.25
Allowable clad temperature (design)	°F	1250

Individual Bundle Parameters***

Fuel Zone	No. of Bundles	Steam Flow Rate lb/hr	Core Press. Drop psi	Bundle Outlet Temp. °F	Outlet Temp. of Hot Element °F	Nominal Max. Clad. Temp. °F	Max. Clad. Temp. ** °F
1	4	5980	41.6	855	920	1131	1235
2	4	4155	20.0	795	925	1126	1239
3	8	5360	31.5	815	925	1133	1234
4	8	2940	10.1	795	945	1110	1250
5	8	2570	7.3	755	950	1110	1225

* Includes statistical errors in heat generation.

** Includes statistical errors -- see discussion on cladding temperatures in this section.

*** The data presented on core performance corresponds to the power distribution shown on Figure II. 15. More detailed control rod configurations will be analyzed as the design progresses.

2. Superheater Region

2.1 A computer code is used to analyze individual fuel elements in the core. This code, supplemented by hand calculations, considers the following factors:

2.1.1 The heat generation varies axially along the element. The code assumes that heat generation is constant across the fuel radius; the results are later corrected to account for the neutron flux and power generation depression which will occur within the UO_2 fuel pellet radially.

2.1.2 The steam film coefficients are calculated from the local bulk fluid temperatures at each position along the element. That is, the variation of steam properties with temperature is included.

2.1.3 Thermal radiation between the outside clad and the process tubes is included with an assumed emissivity at the surfaces of 0.5, based on experimental data.

2.1.4 There is no axial heat flow within the fuel or clad.

2.1.5 The thermal impedance of the clad is included.

2.1.6 Thermal impedances due to the fuel clad interface of $h = 500$ for the inner clad and $h = 1000$ for the outer clad are included.

2.1.7 Crud and scale thermal impedance were included by hand calculation.

2.1.8 A correction was also included to account for the skewed radial heat generation across an element, which causes extremes of hot and cold fluid streaming in the narrow annulus. This effect assumes that the gas on one side of the annulus does not mix with that on the opposite side during that pass.

2.2 For a specified fuel geometry, flow rate, material properties, and element heat generation, the computer code produces peak clad temperatures, peak fuel temperatures, heat lost to moderator, pressure drop through core, and important temperature distributions. The analysis assumes that the pressure drop across all 9 elements in a given fuel bundle is the same. Since the heat generation varies considerably among the 9 elements in a single bundle (see Figure II.15), different individual flows result in each element within a bundle. This effect can reduce the flow in the hottest element of a bundle by as much as about 12% below the average flow in that bundle. From these flows and individual element heat generations, the nominal operating peak clad temperatures are obtained.

2.3 The steam coefficients used are calculated from the equations:

$$N_{Nu} = 0.023\, N_{Re}^{0.8}\, N_{Pr}^{0.4} \qquad \text{when } N_{Re} > 2000$$

$$N_{Nu} = 0.26\, N_{Re}^{1/3}\, N_{Pr}^{1/3} \qquad \text{when } N_{Re} < 2000$$

where N_{Nu} is the Nusselt number
N_{Re} is the Reynolds number
N_{Pr} is the Prandtl number

and the properties are evaluated at the bulk steam temperature. These equations are known to correlate experimental data for annular flow to within about 15%.

2.4 Table II.11 includes both a "nominal maximum" and a "maximum" hot element cladding temperature for each of the five fuel bundle zones. The "nominal maximum" value includes all the factors listed in paragraph 2.1; i.e., it is the most realistic calculation of a peak cladding temperature that the present knowledge and analytical techniques can produce. The nominal maximum temperature is, therefore, the peak temperatures which would be expected to occur in most cases. The "maximum" clad temperature includes, in addition, factors which account for possible errors in heat generation, steam flow and heat transfer film drop. These factors are combined statistically since they represent random variations. Since each of the possible errors could be negative as well as positive, the peak clad temperature has as high a probability of being below the nominal maximum by the amount calculated as being above it and producing the "maximum" clad temperature indicated in Table II.11. For the Mark I core, the 1250°F design limit on clad temperature has been taken to apply to the case where the statistical error factors are included.

2.5 The factors alluded to in the paragraph above, which together account for the difference between the "nominal maximum" and "maximum" clad temperatures, include consideration of the following errors:

2.5.1 Errors in the calculation of heat generation in each element that could be contributed by errors in the physics calculations and variations in the fuel density, enrichment and dimensions are included.

2.5.2 Errors in steam flow are considered from orificing errors, instrument or thermocouple errors, flow maldistribution with a fuel bundle, and flow maldistribution among bundles due to bundle entrance effects and location in the core.

2.5.3 Also included are possible errors in the film drop calculations due to errors in the heat transfer coefficient (paragraph 2.3 above), local mechanical and dimensional variations of the flow annulus affecting film drop, and local fuel pellet density and volume peaks outside nominal tolerances.

2.6 The cladding temperatures shown in Table II.11 and discussed in the paragraphs above do not include an "overpower allowance" for such things as operational transients, maneuvering and "hunting" about the desired power, or transients in the "accident" category. Since the cladding, at 1250°F, is not near any known hard limit, the principal effect of exceeding this value for short periods is expected to be a possible decrease in long-term life of the cladding. The following subparagraphs give an indication of the effect of overpower on cladding temperatures:

2.6.1 For every °F rise in steam temperature in a hot element, the peak clad temperature would be expected to rise about 2°F based on steady-state fuel element characteristics. The alarm levels for high outlet bundle temperature will be set for a rise of about 30°F. Hence, only relatively little maneuvering above the desired steam temperature will be permitted and the cladding rise will not be large at the alarm point.

2.6.2 A 10% overpower would increase the clad temperature in a hot element by about 85°F and the UO_2 center temperature by about 200°F. The 10% overpower would cause an alarm on the neutron flux instrumentation and would cause the mixed mean core outlet steam temperature to rise about 40°F.

2.6.3 The neutron flux scram level will be at about 125% of nominal rated power. If the reactor were to be operated at 25% overpower on a steady-state basis, the hot element clad temperature rise above the rated value would be about 200°F. The mixed mean outlet steam temperature would rise about 100°F.

2.7 For the cladding, the more severe types of transients will be those which might occur from such things as scram without a reduction in steam flow, initiation of emergency cooling, and flooding of the process tubes. In such cases, the cladding suffers a severe thermal shock due to the large temperature differences which can be involved. The effects on fuel element end closure, clad shrinkage over the UO_2, and longitudinal differential expansion between the inner and outer clad are still under study. It is not expected that a fuel rupture will occur, but repeated cycling of this kind could lead to fissures and subsequent release of gaseous fission products.

2.8 The UO_2 fuel temperature is not a limiting factor in the steady state design of the Mark I core. Calculations indicate that the maximum UO_2 steady state temperature will be below 3000°F, as compared with the melting point for UO_2 of about 5000°F. At similar heat fluxes, SADE loop tests on fuel similar to the Mark I type confirmed that the UO_2 was below its recrystallization point. During rapid transients, the fuel temperature will be expected to exceed 3000°F. Examples of the amount of UO_2 temperature rise are indicated in paragraph 2.6 above, and in Section IV of this report. For most situations, it is expected that the clad would probably fail before the UO_2 would reach 5000°F. Therefore, the clad temperature is limiting in this design and the fuel center temperature is not of primary importance.

3. Moderator Region

3.1 The uninsulated process tubes on the Mark I fuel elements receive heat by two mechanisms: convection from the superheated steam in its first pass, and radiation from the outside clad. The process tube dissipates this heat directly into the moderator through a boiling water film. The process tube remains at a temperature very close to the saturated water temperature because the net thermal impedance of the boiling water film and process tube are much less than the thermal impedance of the steam film on the inside of the process tube. The heat loss from the process tube is essentially independent of the moderator void formation, and was determined from the fuel element analysis code both in quantity and axial distribution.

3.2 The heat loss characteristics of individual fuel elements show that heat losses to the moderator for fuel bundles in each of the five core zones are as follows:

Core Zone	Heat Loss Per Bundle
1	33%
2	37%
3	35%
4	40%
5	44%

This heat loss totals about 4.2 Mw in the 12.5 Mw core. Another 5% of the heat produced by the core is generated directly in the moderator and structure, bringing the total heat unavailable for superheating to about 4.8 Mwt or 40% of the heat produced. It is anticipated that in the future the reactor will be used to test fuel having insulated process tubes. This would substantially increase the fraction of heat produced available for superheating.

3.3 The steam-water mixture which forms in the core moderator has a lower density than its corresponding downcomer water leg; hence, the buoyant lift causes a recirculation of the entire loop. A net water recirculation of about 1,425,000 lb/hr would be produced. The voids which form in the moderator depend on the recirculation flow rate, inlet subcooling, and two phase flow relative velocities. With a subcooling of approximately 5.4 Btu/lb, boiling begins on the average at about 45% up the core length. Calculations indicate that the average core exit quality by weight is about 1%. This quality is not significantly affected by the fact that the recirculation rate may be slightly less than calculated. At the core exit, the steam fraction begins to separate and rises to the water surface while the saturated water flows toward the outside perimeter of vessel where it turns into the downcomer annulus and completes the recirculation loop.

3.4 Feedwater is introduced near the top of core level through a feedwater sparger, and enters at a temperature of approximately 130°F. The feedwater rate is 13,500 lb/hr. The subcooled feedwater mixes with the saturated water and produces a net subcooling of 5.4 Btu/lb.

3.5 The steam produced in the moderator has a surface leaving velocity of less than 0.1 foot per second. Based on experimental tests, free surface disengagement can be obtained at up to about 1 foot per second with steam at 6% quality at about 3 feet above the water surface. Demisters have been included at the steam entrance to the fuel bundles at a minimum intake elevation of 4-1/2 feet above the normal water surface. This 4-1/2 foot disengagement distance, coupled with the low surface velocity, may make the use of demisters unnecessary for VSR. Whether demisters are used in the final design will be determined as detailed design progresses and more is learned about the incoming steam effects, surface separation, and any detrimental effects on the fuel. If demisters are used, some bundles may be purposely run without demisters to study these effects.

4. Loss of Flow Effects

4.1 When the reactor is operating at high power, continuity of cooling steam flow is absolutely necessary. If steam flow through the core should be lost, the cladding temperatures would rise very fast and would soon reach temperatures at which clad integrity would be questionable. Even a very fast scram of the reactor is not sufficient protection against loss of flow accidents because a large amount of heat is stored in the UO_2 at a high temperature and must be removed following scram to avoid excessive clad temperatures. The decay heat in the fuel is substantial for some time and must be removed also. Adequate protection can be provided for flow loss situations only with an emergency cooling system which will provide continued high flow for the required period following scram and then lesser flows for a longer period until the decay heat can be dissipated to the moderator directly, without steam flow and without overheating the fuel.

4.2 The heat transfer and fluid flow requirements of the emergency cooling system are discussed in more detail in Section II. F of this report.

4.3 Appropriate steam outlet temperature and steam flow alarms are provided to permit prompt detection and correction of flow reduction situations within quite limited cladding temperature rises. Flow reduction situations beyond reasonable operator control will call for automatic fast shutdown of the reactor and automatic initiation of emergency cooling if appropriate to the situation.

5. Hydraulic Stability

5.1 The nuclear stability of the reactor was considered in Section II. C. 6 of this report. Another potential source of reactor instability to be considered is the pattern of natural circulation of moderator water through the core, on the outside of the process tubes. Large oscillations in gross flow have been observed in low-pressure natural-circulation flowing systems with steam generation. Such hydraulic instabilities have required exit steam volume fractions of 0.6 to 0.9, with the bulk of the data indicating volume fractions above 0.7. It has been observed that the amplitude of oscillation is reduced by an increase in system flow resistance.

5.2 While the problem of hydraulic instability has not yielded completely to analysis, the VSR moderator system appears to be so far removed from instability thresholds that stable operation can be confidently expected. The maximum exit steam volume fraction from the hottest moderator channel in the VSR at rated power will be about 0.2. The high system pressure and high flow resistance tend, in addition, to raise the acceptable range of exit channel volume fractions.

6. Design for High Temperature Effects

6.1 The VSR reactor structure is basically that of existing and proven boiling water reactors. It differs only as required to introduce and remove the steam superheated within the core. The basic approach in designing for the high temperatures has been to (1) isolate the superheated steam from as much of the reactor structure as possible, (2) allow freedom of motion to accommodate thermal expansions, and (3) avoid use of underwater seals, sliding seals, or bellows within the vessel.

6.2 Within the transition cap section of the fuel bundles, superheated steam is isolated from the bundle support structure by the saturated steam flowing down the outer plenum. In the fuel section, the process tubes are exposed to the lower temperature steam flowing down the first pass and are slightly above water temperature. The most complex structure separating the saturated and superheated steam is the inner tube sheet and transition section. Because of their relatively small size, these can be designed to withstand the required pressure differences using thin sections and thus minimizing the thermal stresses.

6.3 The fuel element, suspended from the inner tube sheet, is allowed complete freedom to expand axially within the process tube. The effects of circumferential temperature distributions, possible fuel rod bowing, etc, are still under investigation. Materials for the fuel bundle will be restricted to proven high temperature alloys such as the stainless steels.

6.4 The fuel bundles rest on the channels. Thus, relative vertical expansion between the steam risers and the pressure vessel occurs. Sufficient flexibility has been built into the jumper structure and steam outlet tubes to accommodate this expansion without resorting to bellows or sliding seals. In addition, the riser tubes will be insulated from the superheated steam. The riser pipes will therefore approach the temperature of the saturated steam flowing in the downcomer annulus, minimizing the relative expansion experienced. The risers, downcomers, and steam jumper materials will also be constructed from common high temperature alloys such as stainless steel.

6.5 The superheat steam jumpers are described in Section II. B. 1. The seal design at the couplings is the critical item from the standpoint of high temperature operation. The seal will be designed to maintain minimum leakage under the design conditions and the coupling assembly will be tested prior to finalizing the design to be used in the VSR.

6.6 The VSR pressure vessel differs from those of boiling water reactors primarily in the region of the steam exit nozzles. All nozzles through which superheated steam passes have thermal sleeves. These are being designed to protect both the pressure vessel wall and the steam exit pipe joint against thermal stresses from steady-state or transient conditions.

6.7 The superheat steam piping within the reactor vessel is designed to handle steam at 1100°F. The headering and main superheat steam lines outside the vessel are designed for mixed mean steam temperatures up to 950°F. A maximum cladding temperature of 1250°F is used as the design condition for the Mark I fuel type. Cladding temperatures up to about 1500°F are believed tolerable for short periods. The 1500°F value is used in the current design as a guide in the accident analyses, and it is expected that VSR fuel testing will include operation to this temperature on a planned experimental basis.

E. REACTOR AUXILIARY SYSTEMS

1. General

1.1 If taken literally, the heading "Reactor Auxiliary Systems" could refer to essentially all the equipment in the plant, since all the equipment is truly there because of the reactor and auxiliary to it. However, in order to divide the various systems into categories principally along functional safety lines, a more limited definition has been applied to "Reactor Auxiliaries". Herein the term is taken as applying to those systems or functions which are immediate to the reactor but which do not fall into any of the special categories such as emergency systems, waste disposal, or control and instrumentation which are better treated separately. The division between "reactor auxiliaries" and "power plant auxiliaries" also is arbitrary and is assumed to occur just outside the containment shell. Plant structures and miscellaneous service systems which do not fall logically into the other categories also are treated in separate sections.

2. Saturated Steam Supply System

2.1 The VSR has two separate sources of 1000 psi saturated steam: the VBWR and the gas-fired steam boiler. Only the gas-fired boiler is to be used for initial power operation.

2.2 Saturated steam generated in the VBWR pressure vessel will be piped to the VSR containment vessel. Saturated steam generated in the gas-fired boiler will join the steam from VBWR just outside the VSR containment vessel. The combined saturated steam enters the VSR reactor vessel through two nozzles above the moderator level where it joins the saturated steam from the moderator. The saturated steam then flows into and through the superheat process tubes where it cools the fuel and becomes superheated before passing out of the vessel through individual fuel bundle nozzles to the superheat steam collection and headering system. The basic flow path is shown on Figure II. 3.

2.3 The VBWR system is described in the document SG-VAL 2, Third Edition. The gas-fired boiler is described in Section II. G. The steam piping to the VSR reactor vessel will be designed for appropriate conditions of temperature and pressure and will contain control and isolation valves.

3. Reactor Feedwater System

3.1 The reactor feedwater system is shown on Figure II. 3. Both the VBWR and VSR facilities are arranged in similar but separate condensate-feedwater cycles. At VBWR, the condenser level control valve opens to permit condensate flow from the VSR condensate storage tank to the VBWR hotwell on low hotwell level, and a similar control valve opens to permit the VBWR condensate pumps to discharge condensate to the storage tank on high hotwell level. At the VSR condenser hotwell, level control valves operate in a similar manner to permit condensate flow to and from the hotwell on low and high VSR hotwell levels respectively.

3.2 The condensate from the VSR dump condenser will be pumped via the condensate pumps through a full flow filter, full flow demineralizer directly to the VSR reactor feed pump. Any excess or shortage of flow in this line will be either discharged to or made up from the VSR condensate storage tank. The reactor feed pump will discharge to the gas-fired boiler, through a three-element feedwater control valve; to the reactor pressure vessel for moderator makeup through a single-element reactor level control valve; and to the control rod drive water seals through the control rod seal pressure regulator.

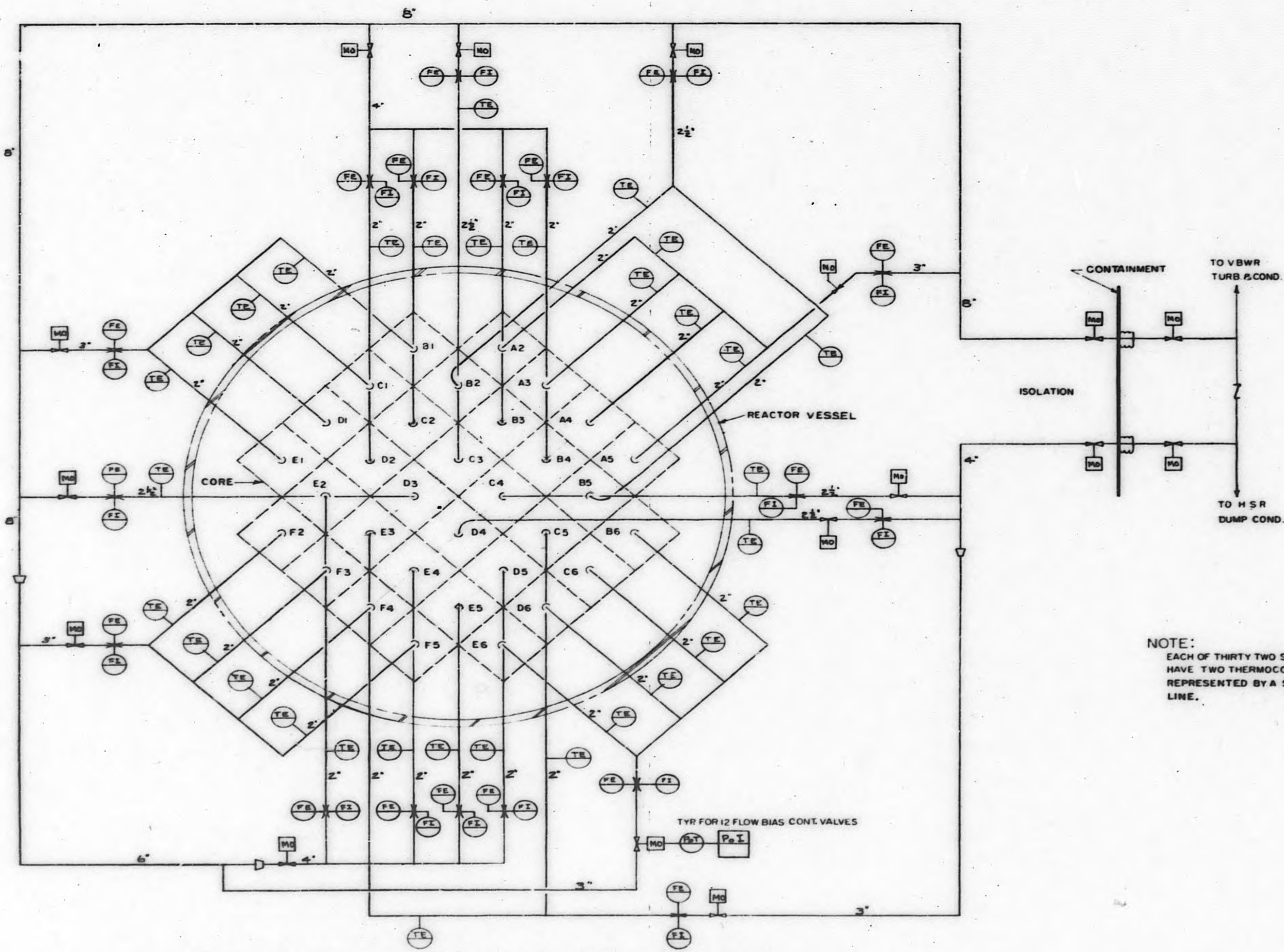
4. Superheated Steam Collection and Distribution

4.1 Superheated steam discharged from the 32 fuel bundles passes from the reactor vessel through 32 individual steam outlet nozzles and is collected and discharged from the containment vessel through the external superheat headering system. The purpose of this system is to provide proper distribution of steam coolant flow through all fuel bundles in the core; to instrument the superheated steam outlet conditions in order to monitor fuel operating conditions; and to direct possible contaminated steam from highly developmental or purposely defected fuel elements directly to the VSR dump condenser. A flow diagram of this system is shown on Figure II. 21.

4.2 Overall steam flow control is described in Section II. H. 7. The superheated steam collection and distribution system provides, in addition, the proper distribution of the total steam flow between the 32 fuel bundles. Slight trimming of the steam flow to a particular fuel bundle or group of fuel bundles is accomplished by movement of the appropriate steam flow bias control valve. Twelve of these control valves are provided, with each valve controlling the outlet steam flow from 1, 2, or 4 bundles as shown schematically on Figure II. 21. Variations in flow requirements of bundles within a group of four fuel bundles controlled by a single valve is accounted for by semi-permanent orifices in the individual lines inside the reactor vessel. These will be placed during refueling based on predicted flow requirements. Depending upon changes in the core loading and experimental requirements, the orifices may be replaced periodically.

4.3 Coolant steam flow through each fuel bundle within the core must be properly apportioned corresponding to the power generated in the fuel bundle. The power distribution within the core will be determined by the gross radial flux distribution across the core, the bundle fuel enrichment, and the bundle coolant velocity and pressure drop. In addition to changes in the radial power distribution due to burnup and control rod distribution, it is anticipated that experimental assemblies with widely differing characteristics will be inserted in the core for experimental purposes. It is desired to match steam flow to power as closely as possible in order to achieve the desired bulk steam outlet flow; to minimize the superheat reactor demand for saturated steam coolant; to achieve reasonably uniform temperatures in the external steam header; and to prevent excessive fuel element cladding temperatures. The orificing of individual lines, external subheadering, and the use of 12 steam flow bias control valves provide considerable flexibility of control of coolant flow to each fuel bundle.

4.4 Each of the 32 superheat steam lines has two thermocouples (total of 64) which are run to the main control room. One of the two is indicated and recorded, and provisions are made for adjustable high and low temperature alarms. The second thermocouple is provided as a standby and its leads can be readily interchanged with the leads of the first thermocouple for verification of readings or replacement in the event of failure. In addition, 18 flow rate indications of individual or groups of steam flow rates are provided on the local control panel in the containment building. A low flow indication on any one of the 18 flow monitors will be alarmed in the main control room. Identification of the point of low flow would then be possible through observation of the local monitors.



NOTE:
EACH OF THIRTY TWO SUPERHEAT STEAM OUTLET LINES HAVE TWO THERMOCOUPLES INSTALLED. THIS IS REPRESENTED BY A SINGLE "TE" SYMBOL ON EACH LINE.

SUPERHEAT STEAM COLLECTION & DISTRIBUTION DIAGRAM

FIGURE II. 21

FIG. II. 21

4.5 The control stations for the 12 steam flow bias control valves are located on the local control panel in the containment building. Three limit switches are provided on each valve to cause automatic control rod run-in and alarm if any one valve is closed beyond a minimum required opening. The switches will be in a two-out-of-three coincident circuit. This will prevent operation of the reactor if, for example, all 12 flow bias valves are not opened sufficiently at reactor startup, or if a valve should be inadvertently closed during power operation. As additional protection against the latter incident, the bias valves will be designed to have a very slow closing movement.

4.6 Highly developmental or purposely defected fuel elements will be placed in fuel bundle locations C-4, C-5, D-4 or E-3 (refer to Figure II.4 for grid numbering system). Steam flow from these four bundles is collected in the divert superheat steam header. The effluent from this header joins the main superheat steam line (outside the containment vessel) in such a manner that it can flow only to the VSR dump condenser and not flow to either the VBWR turbine or condenser. The steam flow from the other 28 fuel bundles is collected in the main superheat steam header. The effluent from this header splits into two lines after leaving the VSR containment vessel. One line (with the check valve) will go to the VSR dump condenser. The second line will be desuperheated and go to the VBWR turbine building. There it will split into two flows; one will be further desuperheated and pressure-reduced for discharge to the VBWR condenser and the other pressure-reduced to 450 psig for delivery to the VBWR turbine.

5. <u>Reactor Water Cleanup and Shutdown Cooling System</u>

5.1 The reactor water clean-up system and the reactor shutdown cooling system have been combined into a single system which, by using the equipment common to both, satisfies the design criteria set forth for each system. This system is shown schematically on Figure II.3.

5.2 During reactor operation, reactor water will bypass the clean-up system pump and will flow at reactor temperature and pressure to a non-regenerative heat exchanger. The heat exchanger will cool the effluent flow of about 3000 lbs/hr to about 120°F. Following the heat exchanger, the water pressure is reduced to 50 psig by an orifice and/or pressure reducing valve. The water then flows through one of two filters and a clean-up demineralizer to the VSR dump condenser hot well. Only one demineralizer is provided since it can be out of service for the short time required to change resins.

5.3 When the reactor is shutdown or when reactor pressure is inadequate to force the water through the clean-up system, the clean-up pump is used and the pressure reducing valve is opened wide or bypassed. Under this condition, the clean-up stream returns to the reactor via the condensate and feed system. When the reactor is shutdown, the shutdown system will be in operation also. The clean-up stream will then become a portion of the shutdown cooling stream as described below.

5.4 The shutdown system is capable of removing 1% of a reactor thermal power of 23 Mw. When the shutdown system is in operation, reactor water passes through the clean-up pump and through the shutdown heat exchanger. The pressure reducing valve is bypassed and the cooled water returns to the reactor via the feedwater line without going outside the containment vessel. If it is desired to also operate the clean-up filter and demineralizer at this time, the flow is split following the heat exchanger to run 3000 lbs/hr through the filter and demineralizer while the remainder of the flow returns to the reactor via the feedwater line. The portion routed through the clean-up demineralizer goes to the condenser hot well and returns to the reactor via the condensate and feed systems.

5.5 When it is desired to remove water from the primary system following refueling and prior to pressurizing the system, it can be removed via the shutdown heat exchanger at a rate of up to 30 gpm and is discharged into the VSR dump condenser hotwell. From there it is routed to condensate storage via the condensate pumps. If desired, a portion may be first routed through the clean-up filter and demineralizer. When the reactor is under pressure, water may be removed from the primary system via the clean-up system at a maximum rate of 10 gpm and routed to the dump condenser hotwell.

5.6 The clean-up demineralizer is a non-regenerative unit. When the mixed bed resins are spent, they are sluiced with condensate to the spent resin tank in the waste system. New resins are mixed in a resin make-up tank and sluiced into the clean-up demineralizer. The demineralizer is then returned to service.

5.7 When filter pressure drop becomes too great, the filter is removed from service and the clean standby filter put in service. The filter cartridges are then removed semi-remotely and replaced with new cartridges. This filter is then placed on standby.

5.8 To facilitate the removal of spent resins, the clean-up demineralizer unit and the filter units are located in the dump condenser and miscellaneous equipment building. A spent resin storage and shipping facility is located adjacent to this area which serves the condensate demineralizer system also.

6. Reactor Shield Cooling System

6.1 The concrete shield around the reactor vessel absorbs the major portion of radiation and neutron flux emanating from the vessel. This heat would cause overheating and damage to an uncooled shield, so a cooling system is installed which will limit shield temperatures to 200°F and temperature gradients to 20°F per foot.

6.2 The auxiliary cooling water pump circulates the cooling tower circulating water through cooling coils embedded in the shielding concrete to provide the desired cooling.

7. Fuel Handling and Reactor Servicing System

7.1 The principal function of this system is to provide access to the reactor internals and to permit loading, unloading and rearranging of fuel assemblies, and handling of control rods and other reactor internals, as well as fuel handling in the storage pool. This is accomplished by a wet handling system. The basic philosophy of the system is to employ water as the shielding medium and do all handling with the simplest tools practicable. Maneuvering, at least during the final movements of positioning of tools, is done manually. Sensing of desired movements is visual through the water shielding medium. All fuel and control rod handling facilities are housed within the reactor containment vessel.

7.2 Equipment will be provided to remotely disassemble fuel riser jumper piping in the reactor vessel, grapple fuel assemblies and draw them into a bottom entry refueling cask while using water cover for shielding. The loaded cask is then transferred to a fuel storage pool by the building polar crane where it is unloaded under water. The fuel is moved in the storage pool with a small crane to fuel storage positions formed by a system of racks supported from the pool floor.

7.3 Prior to shipment of fuel for reprocessing or laboratory examination, the riser piping is cut off by an underwater saw in the fuel storage pool. Limited space and facilities are also provided for underwater examination of fuel bundles in the storage pool.

7.4 The fuel storage pool provided in the containment building for storage and examination of irradiated fuel is shown on the containment building drawings in Section II. L. This pool is equipped with racks designed to accommodate the following equipment:

7.4.1 Approximately 85 fuel bundles on fixed spacing for criticality control. The rack consists of structural members and pipe sections arranged for access to individual positions by the servicing bridge and with provisions for locking the assemblies in place.

7.4.2 Ten control rods. The racks consist of ten vertical channels closely spaced on the pool bottom.

7.4.3 Pool equipment, consisting of portable underwater lights, underwater vacuum cleaner (swimming pool type), and a support bracket for the refueling cask set at an elevation that permits underwater transfer of fuel.

7.4.4 Miscellaneous equipment. These racks consist of simple brackets and racks attached to the walls of the storage pool to permit storage of such items as long handled tools, fuel riser connectors, ruptured fuel storage containers, and general radioactive equipment being held temporarily prior to disposal.

7.5 Material transfer equipment for the fuel storage pool consists primarily of a servicing bridge and attached jib crane. The bridge is operated along two rails one of which is on the edge of the pool and the other in the pool because of the pool geometry. The servicing bridge and crane permits pickup of a fuel bundle (approximately 300 pounds) as it is discharged from the refueling cask, transfer underwater to a fuel storage position, and the reverse of this procedure. It also permits movements of fuel bundles in the pool to the observation platform and to the underwater cutoff saw where the riser piping is removed.

7.6 Other pool equipment used in fuel handling, examination and preparation for shipment includes:

7.6.1 Underwater fuel observation platform designed to support both ends of a full length fuel bundle assembly. Platforms are fitted with hooks and brackets for ease of removal from the pool and for adjustment in depth and horizontal placement in the pool. Matching brackets are provided on the pool wall to accommodate these placements. When not in use for fuel observation, these platforms serve as underwater work tables for work on radioactive equipment.

7.6.2 Fuel receiving stand, consisting of a section of pipe supported on end and braced with light structural members. This stand rests on the bottom of the pool sump during fuel transfers in and out of the fuel storage pool, and acts as a temporary upright storage of a fuel element during unlatching of the refueling cask grapple and grappling of the element with the grapple on the pool servicing bridge.

7.6.3 The underwater saw is a power driven hacksaw designed for underwater cutting of fuel element riser piping when held in the vertical position. It will also be used to cut up this piping and other miscellaneous radioactive equipment in lengths short enough to permit disposal. The saw is equipped with remote controls and a cleanup system to avoid contamination of the pool waters. The entire assembly is frame mounted and can easily be installed or removed from the fuel storage pool for servicing or provide room for other activities.

7.7 Initially, natural convection in the fuel pool will be used to remove the decay heat from irradiated fuel. A pool filter pump system is provided into which a heat exchanger can be installed at some future date if so desired.

7.8 Material handling equipment is provided for refueling, movements of materials and equipment in the fuel storage pool and containment building, receipt and shipment of materials and transfer through the containment airlock. Material transfer equipment used in refueling the reactor consists primarily of a refueling cask, a building polar crane, a cask transfer dolly, and a mobile rack for new fuel. The refueling cask is designed to accept a full length fuel bundle and associated riser pipe in a watertight container which is shielded to limit radiation dose rates to safe levels. The overall cask length is approximately 23 feet and is made in two parts flanged together, so that the bottom portion may be used as a sealed cask for transfer of cut fuel bundles (riser piping removed) for laboratory examination or site storage. The cask is equipped with a power driven winch and a cable fitted with a fuel element grapple to enable it to pull fuel bundle assemblies into the cask and discharge them into the fuel storage pool. The bottom opening of the cask is equipped with a shielded water-tight closure which is opened, closed and sealed by means of controls extended above for access from the servicing platform and main floor during refueling operations.

7.9 Loading of the fuel cask is accomplished in a shallow (approximately 7 feet deep) refueling tank constructed of stainless steel and attached to and supported by the reactor vessel flange. It is equipped with an overflow and drain and will be designed to move with the thermal expansion of the reactor vessel. The tank permits the bottom of the refueling cask to be submerged sufficiently in water when refueling through the ports of the reactor vessel head to afford adequate radiation shielding.

7.10 A reactor servicing bridge supports the refueling cask, supplementary shielding, and operating personnel above the reactor vessel. Adjustable brackets on the refueling cask determine the cask elevation in the refueling tank when suspended from the bridge. The bridge is manually operated on a circular track in the opening above the refueling tank. It has indexed positions for the refueling cask and its angular position is indexed on the circular track in order to spot the cask centerline over any given fuel bundle in the reactor core. Supplementary shielding plates and tread plates are furnished for placement on the bridge as needed. The bridge is easily removable with the polar crane and is normally stored in the reactor vessel head storage pit during reactor operation.

7.11 A turning trunion is furnished with the reactor vessel head and a trunion support frame is built into the concrete walls of the head storage pit. These provisions allow shielded storage and turning of the head for gasket replacement and servicing.

7.12 Miscellaneous grapples and tools will be provided for use in the removal, replacement, and servicing of the reactor vessel head, reactor fuel, control rods and other reactor internals.

7.13 The building polar crane is designed to handle all movements of the concrete shielding plugs located above the reactor, the reactor vessel head, and the reactor servicing bridge refueling cask over the main floor area of the containment building. Both the main hook and the auxiliary hook are equipped with sufficient length of cable to allow the transfer of items to all other floors of the building through equipment hatches provided in the floors.

7.14 A portable fuel rack provides for safe transfer of new fuel from the fuel storage vault through the airlock onto the main floor of the containment building and for temporary safe storage there. Fuel assemblies are locked in place during transit.

7.15 A flanged wheel dolly provides means of transporting the refueling cask through the airlock on rails imbedded in the airlock floor. This operation may be accomplished during reactor operations since the air lock is sized to handle the full cask length while always maintaining one of the airlock doors closed. Outside the airlock, the cask is loaded onto a flat bed truck, using installed hoisting equipment, for transport to the VAL fuel storage facility or to the VAL radioactive materials laboratory.

7.16 Spent fuel from VSR will be stored in the VBWR fuel storage facility prior to shipment for reprocessing. Racks designed to maintain safe spacing for criticality control of the VSR elements will be utilized at this time. License limitations and controls now in effect for this facility will be strictly observed, as will the use of safety equipment and observance of established safe operating procedures.

F. REACTOR EMERGENCY SYSTEMS

1. Emergency Cooling System

1.1 The emergency cooling system is designed to provide protection against sudden loss of steam cooling flow through the VSR fuel. The system is designed on the basis of providing sufficient flow following reactor scram to prevent the fuel cladding temperatures from rising more than 200°F above the design operating temperature. Figure II. 22 shows the minimum required steam flow versus time to limit the clad temperature rise to the specified value of 200°F. Figure II. 23 shows schematically the emergency cooling system which will provide the required flow automatically.

1.2 During normal plant operation, the steam will flow through the thirty-two fuel bundles to the main and divert superheat steam headers and on out of the containment vessel to the VSR dump condenser, VBWR turbine, and/or the VBWR condenser. The valve between the superheat steam headers and the heat exchange coil in the emergency cooling heat exchanger will be open during normal plant operation. This will permit venting of the "on duty" coil during plant startup and eliminate the problem of air fouling the heat transfer through the coil when emergency cooling is required. Steam entering the coil during plant startup will condense in the "on duty" coil and fill it with condensate. The two valves in parallel between the coils and the reactor vessel will be closed during normal plant operation. Sufficient water will be stored in the secondary side of the emergency cooling heat exchanger to maintain operation of the system for at least one hour following initiation of emergency cooling without any make-up water or other attention. Automatic make-up of emergency heat exchanger water will be provided to maintain a minimum water level on the shell side of the exchanger.

1.3 The two valves in parallel between the superheat steam headers and the emergency cooling dump tank will be closed during normal plant operation. The dump tank volume and quantity of contained water are designed to provide the required high emergency steam flows for about the first minute following scram. The method of operation is described in the subsequent paragraph.

1.4 Upon initiation of the emergency cooling system by the safety system, both pairs of "normally closed" valves, shown on Figure II. 23, will be opened. Two "de-energize-to-open" valves are used in parallel at each location to protect against the possibility of a valve sticking closed. Steam will flow to the superheat steam dump tank, with the maximum flow rate governed by the diameter of the orifice at the entrance to the dump tank sparger. This superheated steam flow will discharge into the dump tank water, condensing in the water and heating it to saturation temperature. The continued flow of steam into the dump tank will continue to heat the water, raising the saturated steam pressure in the tank. Superheated steam flow will continue through the dump tank orifice at critical velocity until the downstream pressure reaches about half the pressure at the entrance of the orifice. At this time, the flow rate will decrease rapidly, with steam flow continuing into the tank until the pressure in the tank and the reactor vessel are essentially equal. This flow path will be sufficient to provide the required cooling steam flow for most of the first minute following scram.

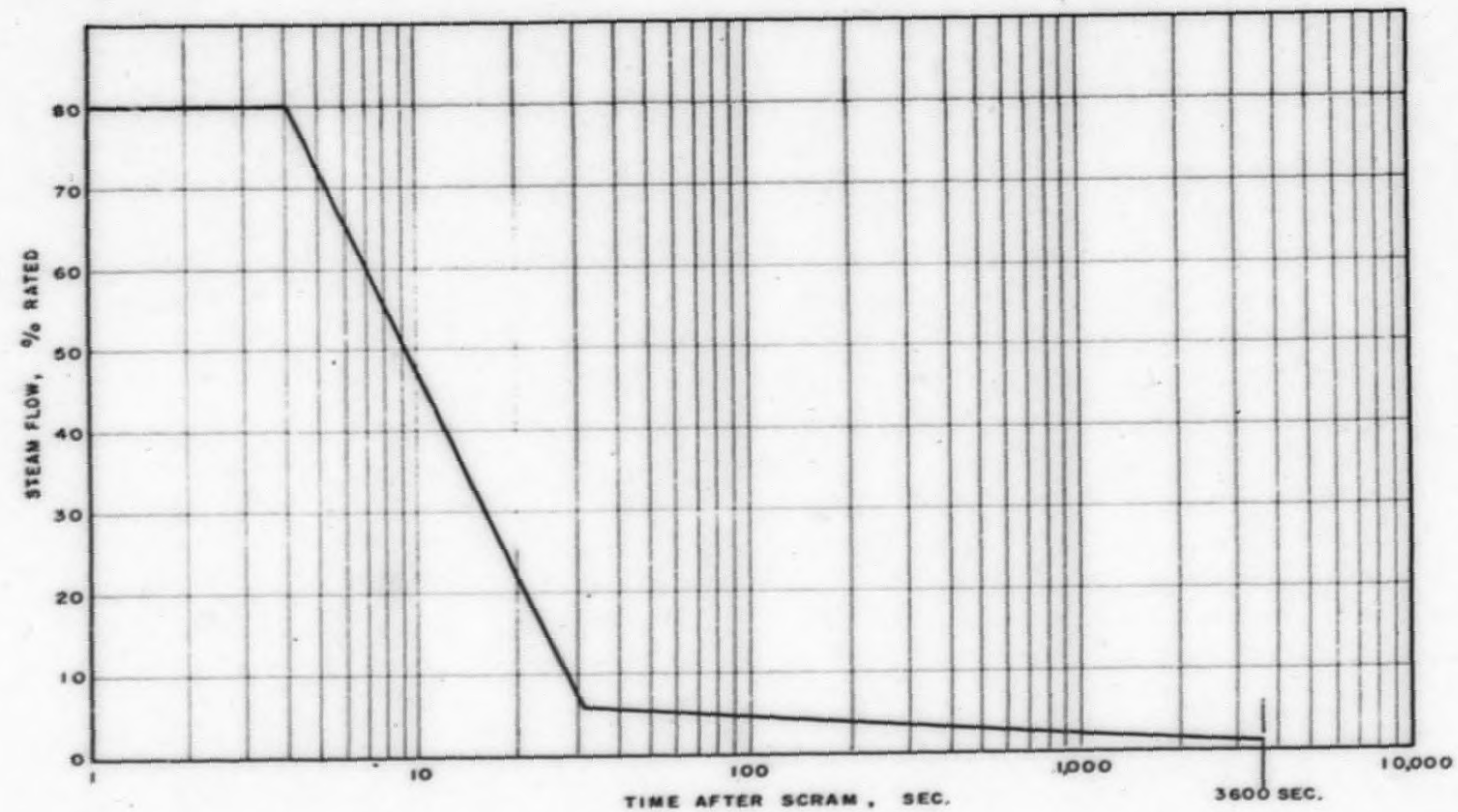


FIGURE II. 22

EMERGENCY COOLING

MINIMUM FLOW REQUIREMENTS

FIG. II. 22

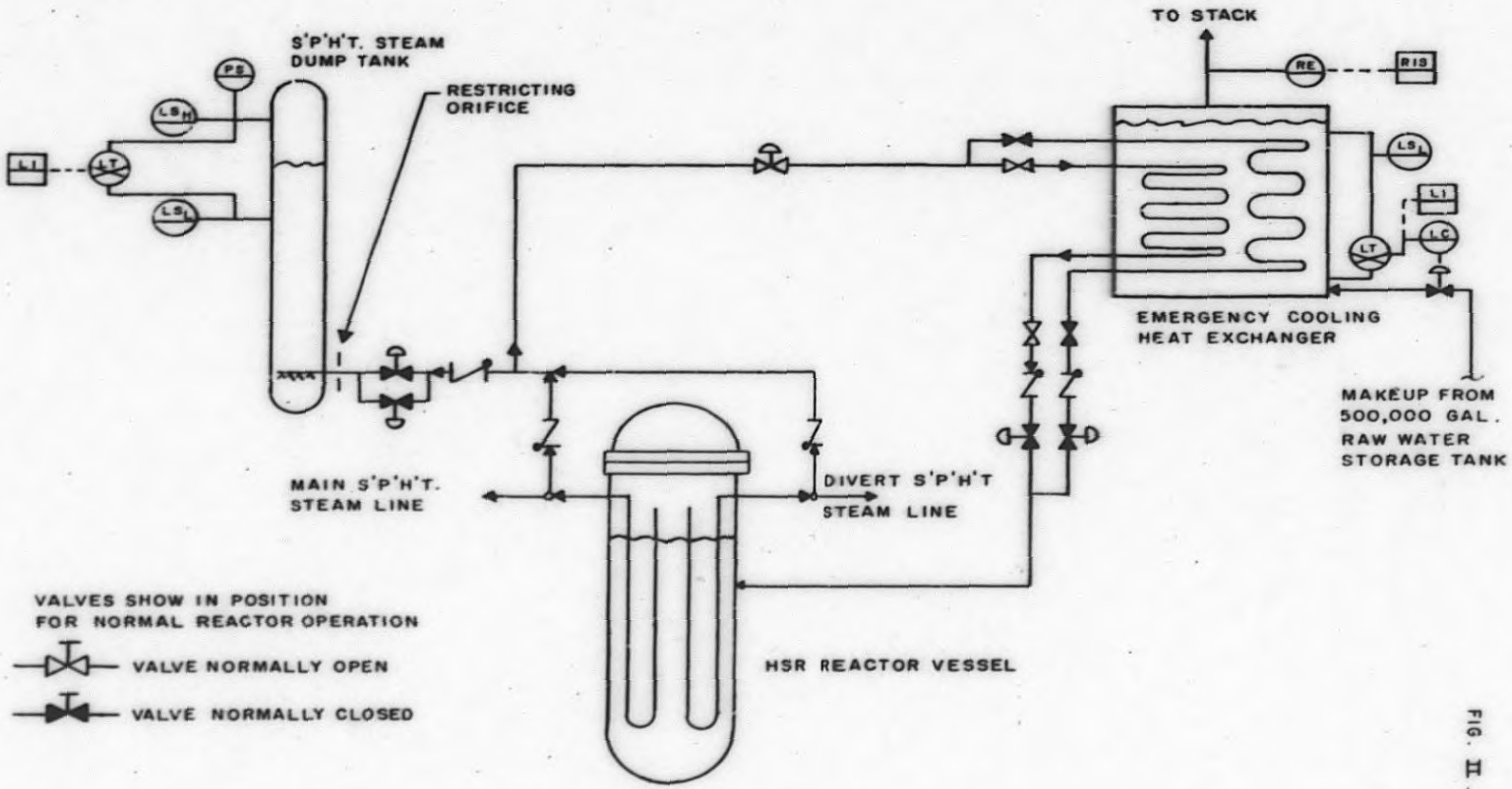


FIGURE II. 23

EMERGENCY COOLING SYSTEM SCHEMATIC

FIG. II. 23

1.5 As noted above, the parallel "normally closed" valves between the "on-duty" coil of the emergency cooling heat exchanger and the reactor vessel open also upon initiation of the emergency cooling system. While high steam flows exist to the dump tank, the steam pressure over the condensate in the condenser coil will be approximately 150 to 180 psi lower than the pressure in the reactor vessel due to the pressure drops in the core and steam piping. As the flow through the fuel bundles and superheat steam lines drop to lower flow rates, this pressure differential will drop also, approximately as the square of the flow rate ratio. The emergency cooling heat exchanger will be located sufficiently higher than the reactor water level (about 30 to 40 feet) to provide sufficient static head in the drain line from the coil to cause the condensate to drain back to the reactor vessel when the flow through the core drops to approximately 20% of full flow. Each coil will be sized to desuperheat and condense steam at a continuous rate sufficient to meet or exceed the minimum flow requirements shown on Figure II. 22 when used in conjunction with the superheat steam dump tank. Return of the saturated condensate to the reactor vessel minimizes the reduction of reactor vessel pressure and water level.

1.6 Two full-duty coils are provided in the emergency cooling heat exchanger. Only one coil will be in operation at a time. The second serves as a full capacity standby in the event the "on duty" coil develops a leak to the secondary side. A leak would be detected by the radiation element on the secondary side exhaust to the stack. If a coil leak should occur, the standby coil would be placed on duty and the leaking coil isolated until a convenient time for repair.

1.7 Instrumentation is provided to insure that the emergency cooling system is always ready for operation if it should be required. Too much water, too little water or high pressure in the superheat steam dump tank will reduce its dump steam capacity. Three level switches at both the high level and the low level points and three pressure switches are provided. A two-out-of-three coincident circuit from the switches will initiate control rod run-in. A separate level transmitter with level indicator and high level and low level alarms in the control room are provided also. Insufficient secondary water in the emergency cooling heat exchanger will reduce the steam condensing capacity of the coil. Three low level switches are provided in a two-out-of-three logic to cause control rod run-in if this should occur. A level transmitter with level indicator and low level alarm in the control room are provided. Automatic water make-up is provided to maintain a minimum secondary-side water level for long-term operation of the emergency heat exchanger.

1.8 The emergency cooling system design is based on the following criteria and assumptions.

1.8.1 Full steam flow capacity through the fuel bundles will be 275,000 lbs per hour. This is substantially in excess of rated flow for the initial 12.5 Mwt reactor operation, providing an additional margin of safety during early operation of the VSR.

1.8.2 Future fuel bundle design may incorporate insulated process tubes, requiring steam flow cooling for one hour as shown on Figure II. 22. After one hour the decay heat can be transferred to the moderator without the cladding exceeding the 200°F temperature rise or need of steam flow cooling. The power density of the fuel may be approximately twice that of the initial startup core.

1.8.3 At the time emergency cooling is initiated, it should be assumed that the reactor is scrammed, the inlet steam may have been stopped or is not available, the source of cooling steam is from flashing of the moderator, the reactor water level is at the low water scram level, the reactor pressure is down to 900 psia, and closure of the superheat steam isolation valves is initiated.

1.8.4 The system operation under the conditions listed in paragraph 1.8.3 above must not reduce the reactor pressure sufficiently to cause severe thermal stress in the reactor vessel.

1.8.5 The system must be capable of operation for at least one hour without replenishing water to the secondary side of the emergency cooling heat exchanger.

1.9 Further analysis of the required emergency steam flow, thermal shock imposed on the reactor vessel under various assumed conditions, and final sizing of equipment remain to be completed. Proper design and operation of the emergency cooling system is essential to reactor safety. It is planned to test the system thoroughly with steam from the gas-fired boiler prior to initial nuclear operation of the VSR.

2. Reactor Safety Valves

2.1 The VSR is protected against overpressure by three safety valves.

2.2 One, an electromatic relief type, is located on the superheat steam line to the superheat steam dump tank. This valve has a design pressure and temperature rating of 1250 psig and 1050°F. The valve discharges below the water level in the fuel storage pool. Its capacity is in excess of 55,000 lbs per hour. It will be set to discharge when the reactor vessel steam pressure reaches 1200 psig.

2.3 The other two safety valves will be mounted on the reactor vessel and discharge to the containment vessel. They have a design pressure and temperature rating of 1250 psig and 575°F. The combined capacity of the two valves will be in excess of 200,000 lbs per hour. The valves will be set to discharge when the reactor vessel pressure reaches about 1220 psig for the first and about 1240 psig for the second.

2.4 The safety valve capacities will be noted to be very conservative with respect to the normal steaming rate of the VSR. The valve capacities are adequate for VSR power levels up to at least 23 Mwt.

3. Liquid Poison System

3.1 The liquid poison system is shown schematically on Figure II. 3. The poison solution utilized is sodium pentaborate dissolved in water. A volume of this solution more than adequate for complete reactor shutdown (cold, clean, and with all control rods out) is stored at ambient temperature in a tank under atmospheric pressure in the dump condenser and miscellaneous equipment building. Two high pressure pumps are provided to deliver poison solution from this tank to the reactor when the reactor is at any pressure up to that for which the safety valves are set. Necessary piping, remotely operable valving and instrumentation are provided. The pumps are electric-motor driven and receive power from the emergency electrical system if normal power is unavailable.

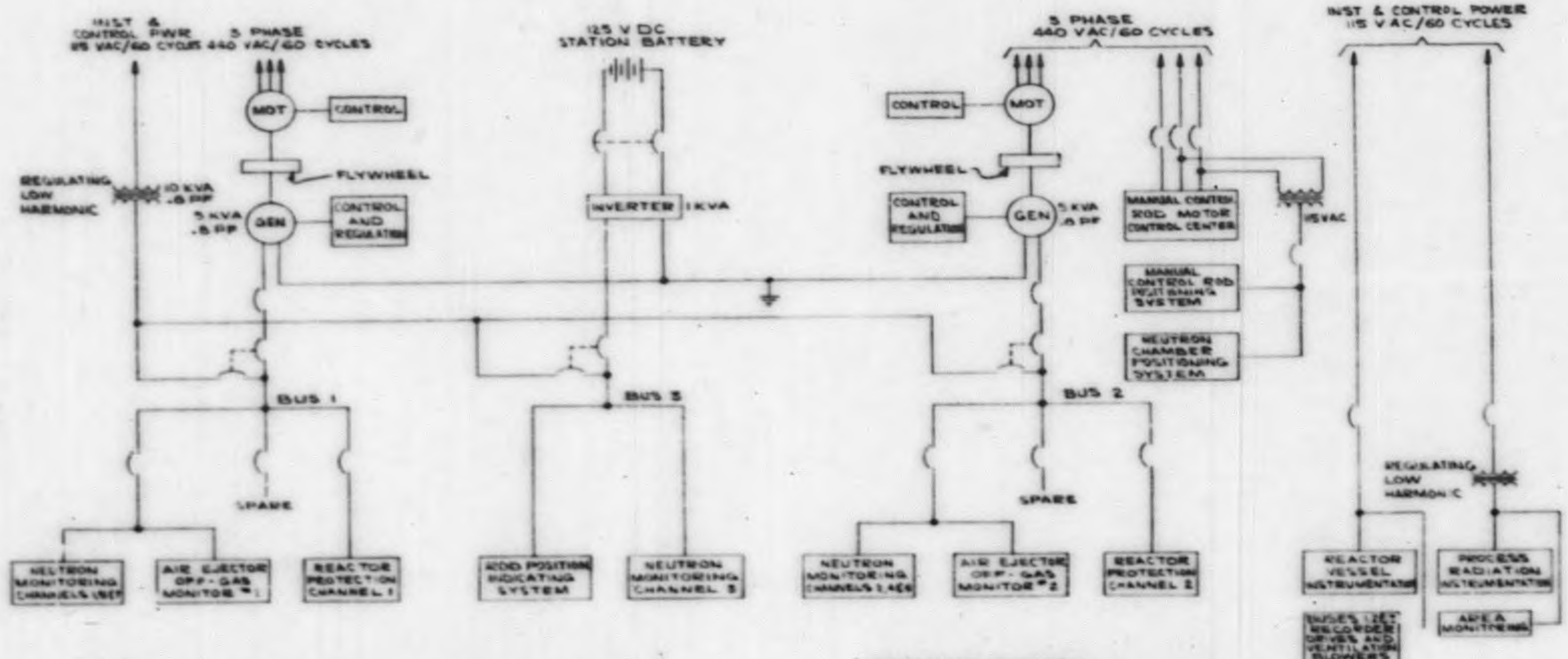
3.2 Injection of the liquid poison solution held in the storage tank is an action which must be manually initiated by the reactor operator. A master control switch is provided to open the two in-line control valves; also, pump controls are provided for pump operation. The pump and valve controls together provide two controls in series for adequate protection against inadvertent operation of the poison system. Valve positions are indicated by pilot lights. Pressure transmitters on pump discharge transmit to indicators in the control room. One pump serves as a backup to the other pump and comes on automatically if the discharge pressure of the first pump does not build up sufficiently shortly after actuation.

3.3 Each pump will have a capacity of about 10 gpm. About 115 gallons of poison solution will be stored and ready for use. Within 10 minutes from the time of actuation, either pump can supply the amount of poison required (about 80 gallons) for complete reactor shutdown. Calculations indicate that a concentration of about 450 ppm of natural boron evenly distributed in the reactor moderator will provide the required reactivity control for complete shutdown.

4. Emergency Power System

4.1 Emergency electrical power is available from a 75 kw automatic quick-starting diesel generator to provide power to certain vital loads necessary to monitor and control the reactor system following a failure of the normal electrical power. In addition, one channel of the power level neutron monitoring system and the control rod position indicating system is always normally operated from the station battery. This makes it possible always to monitor the reactor neutron power level and control rod positions.

4.2 The normal power supplies for the reactor safety system instrumentation and control rod system are shown on Figure II. 24. Emergency power from the diesel generator is available for these loads (busses 1, 2, 3, 4) as well as the other loads listed below.



NOTES:

I – BUSES 1,2 & 3 AND INST & CONTROL POWER CIRCUITS SHOWN ARE RESERVED SOLEY FOR THE INDICATED SYSTEMS.

II – BUSES 1,2 & 3 SHALL BE PROVIDED WITH ONE SYSTEM GROUND. THIS GROUND SHALL BE COMMON WITH THE GROUNDING POINTS OF ALL PANELS AND THE STATION GROUND.

REFERENCE DRAWINGS:

(LATER) NEUTRON MONITORING SYSTEM
AREA MONITORING SYSTEM
NEUTRON CHAMBER POSITIONING SYSTEM
CONTROL ROD POSITION INDICATING SYSTEM
PROCESS RADIATION INSTRUMENTATION
MANUAL REACTOR CONTROL SYSTEM
(LATER) REACTOR PROTECTION SYSTEM

FIGURE II 24
REACTOR INSTRUMENTATION
POWER SUPPLY

FIG II 24

Load	AC Voltage	Phase	Size of Load
Instrument & Control Bus 1, 2, 3 (Safety System)	115	1	10 kva
Control Rod System Bus 4	440	3	10 HP
Liquid Poison Pump	440	3	10 HP
Containment Building Air Lock Doors	115	1	1 HP
Station Battery Charger	115	1	10 kva
Condensate Transfer Pump	440	3	5 HP
Emergency Lighting	115	1	2 kw
Instrument Air Compressor	440	3	20 HP
Communication & Alarms	115	1	1 kw

5. Other Emergency Systems

5.1 The reactor safety system is another important reactor emergency system. It consists of appropriate detectors and instrumentation to sense off-standard or unsafe operating conditions as well as devices to provide called-for control functions such as scram and containment isolation. The safety system instrumentation is described in Section II. H, the scram devices in Section II. B, and the isolation provisions in Section II. L.

5.2 Another important emergency system is the containment system. This system is described in Section II. L, and includes not only the containment vessel but the isolation provisions and emergency enclosure cooling system as well.

5.3 Certain aspects of the off-gas system may be regarded as part of the reactor emergency equipment also. These aspects are included in Section II. J.

G. POWER PLANT AUXILIARIES

1. Gas-Fired Boiler

1.1 The gas-fired boiler is of conventional fully automatic, water tube, natural circulation, steel cased, outdoor design. The boiler is rated at 150,000 lbs/hr of saturated steam at 1050 psig. The steam drum will contain moisture separators capable of furnishing saturated steam comparable in moisture quality to that furnished by a boiling water reactor. The steam discharge may or may not contain free hydrogen, but will not contain free oxygen. The duplication of the hydrogen-oxygen concentration normally found in the boiling water reactor steam discharge will be simulated by oxygen and hydrogen injection near the steam inlet to the superheat reactor. The chemical feed treatment for the boiler feedwater will be designed to minimize carryover of chemical compounds in the steam to the superheat reactor.

1.2 The foundation for the boiler plant will include a basin which will contain and divert to the waste system any spillage or leakage from the boiler pressure parts.

2. VSR Dump Condenser andAuxiliaries

2.1 The VSR condenser and its principal auxiliaries are shown on the simplified flow diagram, Figure II.3. The condensate-feedwater portion of the system was discussed in Section II.E.3. The system equipment includes the condenser, condensate storage tank, condensate pumps, demineralizers and filters, feedwater pump, and auxiliary cooling water system. The cooling tower for the condenser is discussed in a subsequent section, as are the buildings which house the condenser and auxiliaries.

2.2 The VSR dump condenser is a horizontal, two-pass, nondivided water box-type having a deaerating hotwell with 5 minute storage. The capacity of the unit is approximately 232,000 pounds per hour of steam from the VSR system. The maximum oxygen content of the condensate is 0.01 cc/liter. The condenser is constructed from appropriate types and qualities of steel for the shell, Admiralty metal for the tubes, Muntz metal for the tube sheet and cast iron for the water box. The duty is about 250×10^6 Btu per hour, the cooling surface provided about 14,000 ft^2, and the cooling water required about 13,000 gpm. The hotwell volume is over 300 ft^3 to provide the desired holding time. Steam jet air ejection is provided.

2.3 Two motor driven condensate pumps are used, each one of sufficient capacity to handle 60% of the condensate load under normal operating conditions. Check valves are provided in each pump discharge line to prevent back flow through the standby pump.

2.4 Two condensate demineralizers are furnished, each capable of 100% condensate flow. The demineralizers are not regenerative, but rather are designed to replace the spent and contaminated resins with a fresh charge. Two units are provided in order to allow a decay period prior to discharging the spent resins. A full flow filter of the replaceable cartridge type is in series, upstream of each of the demineralizers. Duplicate filter units are furnished for the same reasons as for the demineralizers.

2.5 A separate cooling water circulatory system is provided for the VSR cooling tower. The main circulating water flow is from the VSR cooling tower basin, through buried steel circulatory water piping, to the VSR dump condenser and back to the VSR cooling tower. An auxiliary cooling water service pump is provided to handle miscellaneous cooling loads, instrument air compressors, instrument air aftercoolers, air ejector condensers, shield cooling coils, shutdown heat exchanger, diesel generator cooling, etc. Backup to the auxiliary cooling water pump is provided by the main circulating pump. Make-up water is furnished from the existing 500,000-gallon supply tank and system.

2.6 The condensate tank is a 10,000-gallon aluminum storage tank located on the roof of the dump condenser and miscellaneous equipment building. This tank forms the surge capacity between the VSR and VBWR condensate systems, at the time when the cross connections are made and when the two systems are in joint operation. Control valving and the condensate transfer pump insure and maintain proper condensate levels in each hotwell by having a large source and storage for condensate in the VSR tank. The VBWR condensate tank will remain in the system for separate VBWR operation. Its function during joint operation is mainly that of additional capacity to the system.

2.7 The VSR condenser, the VBWR turbine generator, and the VBWR condenser are cross connected with suitable piping and control valving, designed to produce a system where maximum utility is obtained from existing equipment. Steam flow to each condenser is based on a back pressure of approximately 6 inches to limit the condensate temperature above which demineralizer resin breaks down. Steam flow to the turbine is based on turbine demand, while the excess steam produced will flow to the VSR and VBWR condensers.

3. <u>VSR Cooling Tower</u>

3.1 An induced draft redwood cooling tower removes heat from the condenser circulating water. Tower duty is approximately 260,000,000 Btu/hr with flows of 13,600 gpm. Design conditions dictate an 87°F exit tower temperature with a 125°F inlet temperature. The fans are motor-driven, weatherproof, and operate from a 480-volt, 3-phase, 60-cycle source of electrical power. The cooling tower foundation forms a basin from which the main circulating pump takes suction. The auxiliary cooling water pump also takes suction from the cooling tower basin for the auxiliary cooling water circulation system. Circulating water piping consists of buried and above-grade welded steel piping.

4. Tie In with VBWR Power Plant and Auxiliaries

4.1 Initial operation of the VSR will utilize just the gas-fired boiler and the VSR dump condenser. The connections with the VBWR power extraction system which appear on many of the drawings and in the discussions in this section and elsewhere in this report will not actually be made until a later date when the evaluations pertinent to the tie in have been completed and the necessary licensing actions taken for both the VSR and VBWR. The interconnection plan is discussed in more detail in a subsequent paragraph in this section.

4.2 The VBWR turbine-generator is a ten-stage machine rated at 5200 kilowatts at 3600 rpm. The unit is described in the document SG-VAL 2, Third Edition. The turbine was originally manufactured to utilize steam at 450 psig and about 750°F, but was modified for use with the VBWR for operation with 450 psig saturated steam. Discussions with the turbine manufacturer have indicated that it will be possible to operate the turbine continuously with 450 psig 750°F steam from the VSR without further modifications to the machine. The temperature of VSR steam to the turbine and condensers will be controlled with automatic desuperheaters.

4.3 The VBWR condenser and condensate system is described in the document SG-VAL 2, Third Edition. No modifications to the condenser are necessary for it to be able to handle steam from the turbine or bypassing the turbine, except for the addition of two desuperheaters as indicated above. The VBWR condenser is rated at about 150,000 pounds of steam per hour at an absolute pressure of 7 inches of mercury when supplied with 10,500 gpm of cooling water at 80°F. Under normal conditions, the condenser will operate at about 2-1/2 inches of mercury (86,000 lbs/hr rating) but this may increase to about 7 inches with maximum flow through the turbine bypass line.

4.4 From the VBWR condenser, the condensate flows to the deaerating hotwell, the condensate pumps, cleanup system, and reactor feed pumps back to the VBWR. As indicated previously, excess condensate is discharged to, or made up from, a common VSR-VBWR condensate storage tank. The treatment of condensate from the VBWR condenser is very much like that described for the VSR condenser. The VBWR system has two condensate pumps, one condensate booster pump, two filters, four condensate demineralizers, and two reactor feed pumps.

4.5 The VBWR condensate storage tank is a 4000-gallon tank and is used to store replacement water for the steam generating system. It is equipped with a make-up water processing system consisting of a filter and a demineralizer. Raw water for this system is supplied from the 500,000-gallon laboratory storage tank. This condensate storage tank will remain and operate essentially as originally designed for separate VBWR operation. When the VBWR and VSR are operating together, intermixing of the condensate in the two systems will occur and they will operate essentially as a single larger system.

4.6 The VBWR will continue to have an independent cooling tower and cooling water system. This system supplies cooling water to the VBWR condenser as well as to the air ejector and gland seal condensers, turbine and generator coolers, reactor recirculating and shutdown pump jackets, the reactor clean-up system heat exchanger, and to the instrument air compressor jacket and after-cooler.

4.7 As indicated above, during initial operation of the VSR, there will be no significant physical connections made with the VBWR system. During this time, each reactor will operate essentially independent of the other although they will share the same liquid waste disposal system, service air, backup instrument air, condensate makeup, fire and sanitary water supply, and control building. Also, the VBWR condenser off-gases will be discharged from the new VSR stack. It is apparent that this initial sharing of equipment cannot cause effects of operation at one reactor to be transmitted to the other reactor in such a way as to affect the safety of operation of either reactor. The tie-ins which are significant to safety will be made at a later date after some experience has been gained with operation of the VSR using just the gas-fired boiler as the saturated steam source and just the VSR condenser for utilization of the superheated steam produced. During construction of the VSR facility, it is planned to run lines to near the point of tie-in with the VBWR system and then to blank off the lines until the tie-in is approved. The addition of necessary nozzles to the VBWR system may be done in advance of the tie-in, within the scope of the VBWR license, and then these nozzles would be blanked off also. A similar procedure will be used with respect to the interconnection of VSR and VBWR instrumentation which is implied in this report. The significant tie-ins to be made later will include the VBWR steam supply to the VSR, VSR steam discharge to the VBWR turbine and condenser, the condensate systems, and the control system. For initial VSR operation, only the sharing of equipment indicated earlier in this paragraph will take place and no other interconnections will be made.

H. CONTROL AND INSTRUMENTATION

1. Layout

1.1 The VSR control room will be essentially an extension of the existing VBWR control room. The VBWR control room occupies the northeast corner of the second floor of the control building. An addition to the control building will be made as shown on the plot plan, Figure II.1. On the second floor of this addition, and adjoining the east side of the VBWR control room, will be an area of approximately 23 x 31 feet for the VSR control room. The remainder of the control building addition will be for offices, and the first floor area below the VSR control room will serve as a control access change and locker room for the VSR facility.

1.2 The panel and control console layout of the VSR control room is shown on Figure II.25. A large control console, set back about four feet from the control rod position indicator panel, contains all of the manual controls necessary for routine plant operation as well as indication for some of the more important process variables. This console and control rod indicator panel are oriented on a 45-degree bias with the panels to each side, thereby giving the operator at the console a good view of all panel-mounted instruments.

1.3 The control rod indicator panel also contains the visual annunciators. To the left of this panel, the neutron monitoring instrumentation occupies about 8 feet of linear panel space. The next two feet of the same panel contain the area radiation monitoring instruments.

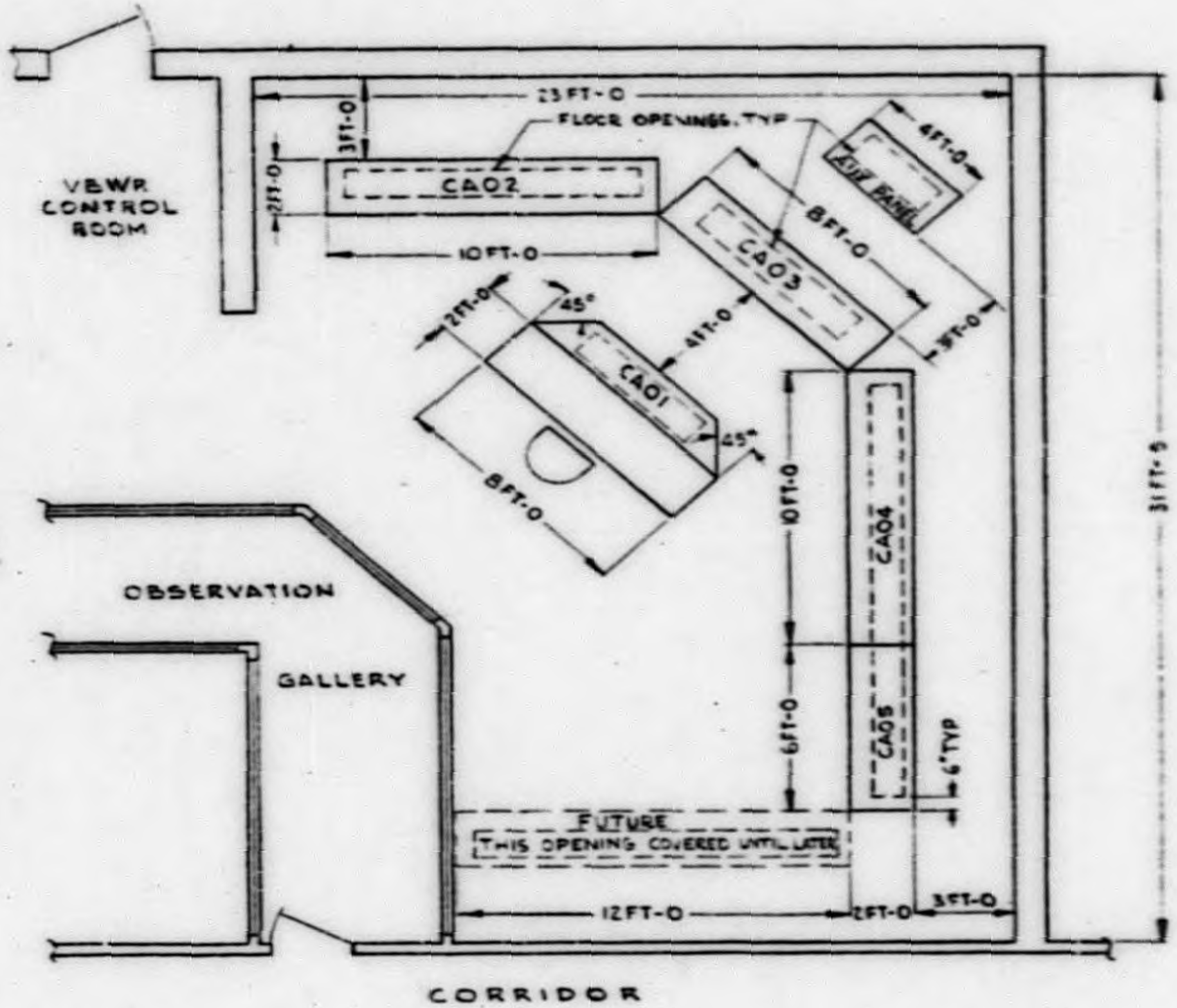
1.4 To the right of the control rod indication panel is the process instrumentation (flow, pressure, temperature, level, etc.) occupying about 10 feet of panel. The next six feet of panel to the right contains the off-gas, stack gas, and other process radiation instrumentation.

1.5 A space large enough for about 12 feet of linear panel space is provided at the south end of the control room for future instrumentation needs.

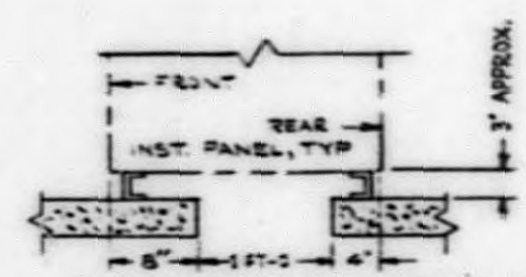
2. Neutron Monitoring System

2.1 An instrumentation system is provided to monitor the neutron level and control the reactor from startup through full power. The instrumentation covers the range in three phases startup, intermediate range, and flux level or power range.

2.2 With the initial fuel loading, a neutron source is inserted in the reactor core to assure a count rate of several counts per second. The startup instrumentation covers the range upward to about 10^{-4} of rated power. Two channels of instrumentation monitor the low level startup phase of operation. The primary neutron detectors are proportional counters. The average rate of the series of pulses is measured on a count rate meter with a logarithmic scale in order to encompass five decades of measurement. The count rate is recorded continuously. An additional circuit differentiates the log count rate and indicates the reactor period at this low level on a period meter. The approach of a short period at this low level is annunciated.



PANEL	ARRGT DWG	SYSTEM
CA01	LATER	CONTROL CONSOLE
CA02	114B5572	NEUTRON & AREA MONITORS
CA03	114B5575	CONTROL ROD INDICATION-ANNUNCIATORS
CA04	LATER	PROCESS INSTRUMENTATION
CA05	LATER	PROCESS RADIATION MONITORS



PANEL SUPPORT & FLOOR OPENING ARRGT

NOTE: BOLT CENTERS ARE 2½ FROM FRONT & REAR PANEL FACE

FIGURE II. 25

CONTROL ROOM LAYOUT

FIG. II. 25

2.3 In the intermediate power range, the instrumentation system is concerned with the rate at which the neutron flux is increasing. Two channels of instrumentation monitor and control the reactor as it rises toward full power. The primary detectors are gamma-compensated ion chambers. These log-N channels are on scale and overlap the previous startup channels. The output of the gamma-compensated ion chamber is passed to a log-N amplifier which indicates the logarithm of the input on a scale covering 10^{-7} to full power. The output of the log-N amplifier is continuously recorded. A time derivative of the output is also indicated on a meter as the reactor period. An early warning is indicated visually and audibly if either channel approaches a short reactor period. Control rods will be inserted if the reactor does get on too short a period.

2.4 At about 10^{-6} of full power, the power level instrumentation starts functioning and monitors the reactor through full power. There are three channels of instrumentation to cover this range. The primary detectors are gamma-compensated ion chambers. The output is fed into a power range flux amplifier of fast response and stable characteristics. The power level is read on a meter indicating the per cent of full power and is continuously recorded. A warning is annunciated if any of the three flux level indicators exceeds a preset percentage of over-flux. A signal to insert rods and shut the reactor down is initiated whenever two of the three power level indicators show an excess of neutron flux.

2.5 Axial tubes positioned in five radial positions will be located in the reactor core. These tubes permit wires to be inserted and removed from the reactor core during operation. Wire so irradiated may be taken to the counting room for counting and determination of the neutron flux profile over the entire length of the active fuel zone in five radial positions.

3. Process Radiation Monitoring Instrumentation

3.1 Seven liquid process monitors are employed for operational information and as a safeguard to prevent excessive release of radioactive material in liquid form. These monitors are employed at the influent and effluent lines of the condensate and clean-up demineralizers, on the line for radioactive water to storage, in the fuel storage pool, and on the auxiliary cooling water circuit.

3.2 One flow-through ionization chamber (Kanne chamber) and two gamma spectrometers are employed to continuously monitor the air-ejector off-gas radioactivity. The Kanne chamber monitors the gross beta-gamma activity. The gamma spectrometer is employed to monitor for specific isotopes of fission gases and fission iodine. Off-gas samples may be taken from the off-gas line either upstream or downstream from the charcoal filters.

3.3 The following instrumentation is employed to monitor stack-gas samples collected by means of an isokinetic probe in the stack and an air pump:

Beta-gamma moving filter particulate monitor
Iodine integrator monitor
Flow-through ionization chamber
Dual-channel gamma spectrometer

3.4 Instrumentation for monitoring gamma radiation at ten locations throughout the plant will be supplied. The gamma level at each station will be indicated and recorded at a central location near or in the control room. Adjustable trips will provide alarm actuation for each station. Each station will be capable of measuring the gamma level over three decades between .01 mr/hr and 1 r/hr. and over the energy range of .07 to 7 mev.

4. Reactor Instrumentation

4.1 Sixteen thermocouples will sense the temperature at critical locations on the outside surface of the reactor vessel. Four of these thermocouples will be selected to be recorded in the control room. The remainder of these thermocouples will be available for reading with portable instruments in the control room. Interchanging any of the thermocouples for recording on the four-point recorder will be possible.

4.2 A pressure transmitter senses the vessel pressure and supplies signals for indication and recording in the control room. High and low pressure will be annunciated from trips on the recorder. The vessel pressure will be indicated within the containment vessel. Four pressure switches sense vessel pressure and supply high pressure trips to de-energize the safety circuit in the event of excessive pressure in the vessel. Four pressure switches sense vessel pressure and supply low pressure trips to de-energize the safety circuit in the event of very low pressure in the vessel.

4.3 Two compensating columns, each approximately 180° apart with respect to the vessel, supply water legs for moderator level measurement. Eight liquid level switches, four on each column, supply low and high level trips to de-energize the safety circuit in the event of very low or very high moderator level.

4.4 One liquid level transmitter senses the liquid level in one of the above columns and supplies signals for an indicating-recording controller in the control room. The moderator make-up flow control valve is regulated by this controller. The control valve is also restricted in its opening rate to prevent excess reactivity changes due to temperature changes. Manual control of the valve will be possible from the control room.

4.5 A third compensating column on the reactor vessel supplies a water leg for level measurement from below the moderator make-up inlet to above the top of the head. A liquid level transmitter senses the liquid level in this column and supplies signals for a level indicator in the control room. This level measurement serves as a back-up to the other during operation and gives level information for flooding after shutdown.

4.6 The temperature of the outlet steam from each fuel bundle will be indicated in the control room. Two thermocouples on each tube are installed upstream from the steam collection headers. Either thermocouple can be quickly connected to the temperature indicator. Each of the thirty-two indicators has high and low adjustable trips for alarm. The thirty-two temperatures that are indicated will be recorded.

4.7 The 32 individual fuel bundle steam flows are reduced to 12 collection manifold flows. The superheated steam flow will be measured at 18 points and indicated on a local panel within the reactor containment vessel. The remote control stations for the twelve flow control valves (one for each manifold) will be located with these indicators. A low flow trip on any indicator will cause an alarm in the control room.

4.8 Thermocouples will be located to sense the bulk outlet steam temperature of each of the two outlet steam lines. Each temperature will be indicated in the control room and each indicator will be equipped with a high temperature alarm. These temperature measurements will also be used to compensate the bulk outlet steam flow measuring instruments.

4.10 Instrumentation will be supplied that will compare neutron flux to inlet steam flow; and, when the ratio of normalized flux to normalized flow exceeds about 1.25, a trip will de-energize the safety circuit. When this same ratio exceeds about 1.35, a second trip will initiate emergency cooling. Two flux amplifiers will receive flux level signals from the two log-N channel compensated ion chambers, automatically, as soon as the period scram function of these instruments has been bypassed (above about 1% of rated power). The signals from the flux amplifiers will be compared with the signals from two flow measuring instruments, providing two flux/flow signals for indication, recording, trip and alarm purposes.

4.11 Twelve thermocouples will be located within the concrete shielding around the vessel. The temperature of the shield at these twelve points will be available for measurement by portable instruments at a central location within the containment vessel.

5. Control Rod Drive Control System

5.1 The reactor has twelve electrically-positioned control rods, each individually controlled from the main control room. Mounted on the main control console are twelve switches which are positioned in a graphic pattern corresponding to the radial position of the associated control rod in the reactor core. Operation of one of the switches designates or selects a particular control rod for operation. Also provided is a single control switch to insert or withdraw the designated control rod. The controls are electrically interlocked to prevent the manual operation of more than one control rod at any time. Provision is made to expand the number of control rods to a total of 24.

5.2 The position of each control rod is continually displayed in the main control room. Twelve single-turn Selsyn indicators are positioned on the main control board in a graphic pattern corresponding to the radial position of the associated control rod in the reactor core. This pattern is identical to the rod selection switch location pattern. Associated with each Selsyn indicator are four lights:

Red Light	-	Indicates rod is fully withdrawn
Green Light	-	Indicates rod is fully inserted
White Light	-	Indicates rod is selected for operation
Yellow Light	-	Indicates control rod and traveling nut are in contact.

6. Reactor Safety System

6.1 The reactor protection system is designed to protect equipment, plant and personnel by rapidly scramming or running-in the control rods in the event of an accident or the occurrence of a potentially hazardous situation. Control signals which initiate scram originate from a variety of control and detection devices. These signals activate control circuitry to cause the reactor to be immediately shut down and initiate operation of penetration closures, emergency cooling, and whatever other devices are necessary for safe plant shutdown.

6.2 The safety system consists of two, independent, "fail-safe", safety channels both of which must be de-energized to produce a reactor shutdown or other safety system function. Each sensing element is continuously monitored so that an operation (either continuous or intermittent) is clearly indicated and identified. Whenever practical, the two channels are physically separated and clearly identified so as to minimize the possibility of maintenance personnel causing an accidental shutdown.

6.3 The plant conditions which are monitored by the reactor safety system and used to cause scram and other automatic safety measures are listed below and summarized in Table II.12.

6.3.1 High Neutron Flux: This indicates a reactor power output in excess of the safe level for continuous operation. Three channels of neutron monitoring instrumentation are furnished.

6.3.2 Short Period: This indicates an excessive rate-of-rise of reactor power. Two channels of neutron monitoring instrumentation are furnished.

6.3.3 High Reactor Pressure: This limits the rise in core power due to the small positive pressure coefficient and protects against the need for operation of the reactor safety valves. Four pressure switches are provided.

6.3.4 Low Reactor Pressure: Probable causes for low reactor pressure include failure of the steam generating source, failure of the pressure regulating valve(s) or rupture of the inlet steam line; all represent loss of cooling steam flow. Four pressure switches are provided.

6.3.5 High Enclosure Pressure: A pressure rise within the reactor enclosure indicates a major rupture of the high pressure system with possible fuel damage and release of radioactive material. Scramming the reactor minimizes the possibility of damaging the core. Since there is the possibility of release of radioactive material, isolation of the enclosure is initiated by closing critical enclosure penetrations. Four independent pressure switches are provided.

6.3.6 Low Water Level in the Reactor Vessel: This is to shut down the reactor before the moderator water level falls below the top of the reactor core and to preserve moderator for flashing to the emergency cooling system to provide the required emergency steam flow through the core. Four level control switches are provided.

TABLE II. 12

REACTOR SAFETY SYSTEM

Detectors	CONTROL	LEVEL**	Scram	Annunciate	Close Steam Outlet Valves	Open Emerg. Cooling Valves	Automatic Rod Run-In	(See Note 1) Containment Building Isolation	OTHER CONTROL
3	Neutron Flux On Scale Interlock	~3% Scale							(See Note 2)
	Neutron Flux	~.01 N							(See Note 3)
	Neutron Flux	1.1 N		X					
	Neutron Flux	1.25 N	X	X					
2	Period - Low Range	20 sec.		X					
2	Period - Intermediate Range	15 sec.		X					
	Period - Intermediate Range	10 sec.	X	X					
4	High Reactor Pressure		X	X					
4	Low Reactor Pressure		X*	X	X*	X*			
4	High Enclosure Pressure		X	X	X	X		X	
4	Low Reactor Water Level		X	X	X	X		X	
4	High Reactor Water Level	22.5 ft.		X			X*		(See Note 4)
2	Neutron Flux/Steam Flow	1.1 ratio		X					
	Neutron Flux/Steam Flow	1.25 ratio	X	X					
	Neutron Flux/Steam Flow	1.35 ratio	X			X			
4	Low Inlet Steam Flow		X*	X					
12	Steam Outlet Valves (4 Valves)	10% closed		X		X	X*		
3	High Condenser Pressure			X			X*		
4	High Seismic Disturbance		X	X					
1	Manual Action		X	X					
12	Low Scram Accumulator Pressure in one Accumulator			X			X		
4	Loss of Electrical Power to Control Rods (2 sec. delay)		X	X					
3	Emergency Dump Tank Valve Open			X			X		
12	Off-Standard Conditions in Emergency Cooling System (any of 4 conditions)			X			X		
3	Feed Pump Tripped			X			X*		
3	Low Scram Cylinder Back Pressure			X			X		
36	Steam Flow Control Valves (12 Valves)	X% closed		X			X*		
6	Steam Inlet Valves (2 Valves)	10% closed		X			X*		
4	High Radiation Level From Outlet Steam Lines (2 Lines)		X	X	X	X			

N Neutron flux at 100% reactor power
* Indicates functions which are by-passed during refueling
** Final determination pending analysis

General Notes

Note a) Loss of power to the safety system channels number 1 and 2 will scram the reactor, initiate emergency cooling, close the steam outlet valves, and close all enclosure isolation valves.

Note b) The enclosure ventilation inlet and outlet control valves will be closed and the control rods run-in whenever the reactor is scrammed.

Note c) When a control rod and its traveling nut are not in contact, the affected control rod will be automatically run-in to establish contact.

Note d) High off-gas radiation in either the VBWR condenser off-gas line or the HSR off-gas line will cause the respective off-gas line isolation valve to close before the high activity off-gases can pass the isolation valve.

Note e) Manual controls are provided to position all system valves; however, the Safety System will override any normal manual control.

Numbered Notes

Note 1) When indicated, the following "Containment Building Isolation Valves" will close: Saturated Steam Inlet, Discharge to Waste Tank (Manual Bypass), Start-up Vent to Dump Condenser, Cooling Water Outlet, Clean-up Demineralizer Water to Condenser Hotwell.

Note 2) The neutron flux on-scale interlock will prevent rod withdrawal in the Intermediate and Power Ranges unless two of the three neutron flux monitoring amplifiers indicate more than about 3% of scale.

Note 3) The period scram function will be by-passed when the neutron flux amplifier range switch is positioned for full scale indication of 6% or more reactor power.

Note 4) Reactor feedwater valve is closed when level reaches 22.5 ft. The inlet steam valve is closed when the level reaches 23 ft.

6.3.7 High Water Level in Reactor Vessel. This prevents water from being carried into the superheat fuel elements with subsequent erosion and thermal stresses. Flooding, of course, is the extreme condition this scram protects against. Four independent level switches are provided.

6.3.8 Neutron Flux to Steam Flow Ratio. This prevents a dangerous rise in fuel temperature due to low coolant flow for the power being generated. Two detectors will be provided.

6.3.9 Low Inlet Steam Flow. This provides a backup to the neutron flux to steam flow ratio scram. Four independent detectors will be provided.

6.3.10 Loss of Plant Auxiliary Power. The reactor-safety-system motor-generator sets receive their driving power from separate sources in the plant electrical system. This will assure safety circuit functioning when any electrical power is available in the plant. High moment of inertia is designed into the units so that they will ride through reasonable system disturbances without causing reactor scram. For total loss of electrical power, reactor scram will result, because of the fail-safe type of design, after the generator output has decreased below the requirements of the safety system components.

6.3.11 High Seismic Disturbance. This scrams the reactor in anticipation of possible earthquake damage. Four independent detectors are provided.

6.3.12 Manual Action. A manual control is readily available for the operator to scram the reactor or to manually initiate other safety system functions.

6.3.13 High Radiation Level From Outlet Steam Lines. This scrams the reactor and isolates the steam lines to minimize fission product transport out of the reactor system in the event of a major fuel failure.

6.4 In addition to the scram functions discussed above, Table II.12 also lists the functions which cause automatic rod run-in, which is a slightly slower method of shutting down the reactor. These functions which cause run-in do not require the same speed of action as scram, but nevertheless require reactor shutdown because it will be needed in a short time (e.g., high condenser pressure) or because a potentially unsafe condition has been created (e.g., emergency dump tank valve opens so that emergency cooling would not be available if the need should later arise). If electrical power should be unavailable to run the control rods in, a scram will occur after a two-second delay. Thus, all run-in signals are actually backed up by a scram.

6.5 The reactor safety system is a dual bus static control system consisting of logic elements, power supplies and power switches. The control elements are solid state devices using silicon diodes and silicon transistors to perform logic and switching functions.

6.6 Each channel of the dual bus logic system is powered by two parallel operating 24/12 volt DC power supplies which are sized so that either one can provide power to the safety system logic elements. The logic elements are composed of diodes arranged to perform a desired logic and to provide a dual series element transistor output. The diode logic switching elements are operated in pairs with a common input and a two-transistor series - parallel output for control of the power switches. The power switches which control the power to the control rod scram solenoids are operated in parallel and are reset independently to reduce load inrush current. During normal operation, the control input to the power switches is held at a nominal 12 volts by the logic elements. To initiate reactor scram, the control input to the power switches is dropped from 12 volts to a fraction of a volt, causing all of the power switches to cut off.

6.7 The input to the logic element is a nominal 12 volts DC supplied typically through a pressure or level switch or supplied directly from one of the neutron monitoring instruments. If all inputs are 12 volts, all logic element output transistors are held in their conducting state and the control input to the power switches is maintained at 12 volts DC. If any single input signal goes to zero, both transistors in a given logic element cut off and initiate a trip signal in the respective channel. If both channels of the reactor protection system are tripped in coincidence, the reactor is scrammed.

6.8 The dual channel reactor protection system utilizing series-parallel switching elements provides a high integrity reactor control system with a very low incidence of false operation due to component failure. Any transistor or diode can fail in either a short circuit or an open circuit, and it will not cause a false scram or cause the reactor to faii to shut down if it should. Furthermore, any logic element or power switch may be removed from the system without impairing the integrity of the reactor protection system.

7. Reactor Control

7.1 This section describes particular reactor control features which are provided for refueling, startup, and normal power operation.

7.2 Table II.12 indicates, with asterisks, those scram and control rod run-in functions which are bypassed during the refueling operations so that one of the 12 control rods can be withdrawn as "cocked safety" during the insertion of fuel bundles into the core. The bypassing is accomplished with a keylocked switch and remains effective as long as the reactor pressure is not above atmospheric and the neutron flux amplifiers are downranged to or below the 10^{-5} power scale. The latter refueling permissive is provided in addition to the "on-scale rod withdrawal permissive" to prevent inadvertent upscaling of the flux amplifiers once the rods are cocked. The scram and rod run-in functions not marked with an asterisk on Table II.12 provide the required reactor control functions during refueling.

7.3 For nuclear startup of the reactor, none of the functions shown on Table II.12 are bypassed since the plant startup procedure requires that the reactor be near rated operating conditions of temperature and pressure and have a minimum steam flow of about 25% of rated, prior to withdrawal of any control rods. The procedure is backed up by appropriate permissives to control rod withdrawal furnished in the safety system.

7.4 A simplified diagram of the steam flow control scheme for power operation of the VSR is shown on Figure II.26. The designation of controls and valves is an arbitrary one used only for the purposes of this discussion. The incoming saturated steam pressure, measured at the reactor inlet by pressure transmitter PT-7, is held constant by pressure regulating or control valves, PCV-3 and PCV-7, in the two steam lines from the VBWR reactor and the gas-fired boiler. When both sources are supplying saturated steam to the VSR, the set point of the VBWR pressure controller, PC-3, will be set higher than that of the boiler pressure controller, PC-7, to draw the maximum amount of steam from the VBWR. A low pressure limit signal measured by PT-3 upstream of the VBWR regulating valve prevents dropping VBWR reactor pressure below a set point by limiting the valve opening. (It may even close the valve since it is an overriding control.) When this point is reached, the pressure drops slightly to the set point of the boiler pressure controller which then controls its valve to maintain system pressure. In general, for joint operation, the VBWR regulating valve will be at a limit while the boiler regulating valve controls the pressure.

7.5 When VBWR alone is supplying steam, PCV-3 controls VSR pressure subject to the low pressure limit mentioned above. The VBWR has high pressure protection in the form of a bypass system, PCV-8, to its condenser. The boiler has a separate pressure regulator which controls the gas firing to maintain boiler pressure above VSR pressure. Thus, during initial operation of the reactor when the boiler is the only steam source, PCV-7 controls VSR pressure and boiler firing will vary according to demand for steam.

7.6 Outgoing superheated steam pressure, measured at an outlet header, is held constant by control valves PCV-6 and PCV-5 in the steam lines to the VSR condenser and the VBWR condenser. The steam flow to the VBWR turbine is controlled by a regulating valve PCV-4 which holds turbine pressure constant. Therefore, the speed governor dictates turbine steam flow and its variation is taken up by varying the dump steam from automatic action of PCV-5 and PCV-6.

7.7 The pressure set point of PC-6 will be set lower than PC-5 to allow loading the VSR condenser to its full capacity before steam is shunted to the VBWR condenser. When the VSR condenser pressure reaches a set limit, the vacuum controller, VC-1, will impose a limiting signal on the PC-6 controller which will limit the valve opening of PCV-6. This allows the outlet steam pressure to rise, which causes PC-5 to open the valve PCV-5. At this point, the VSR outlet pressure will be under control of PC-5.

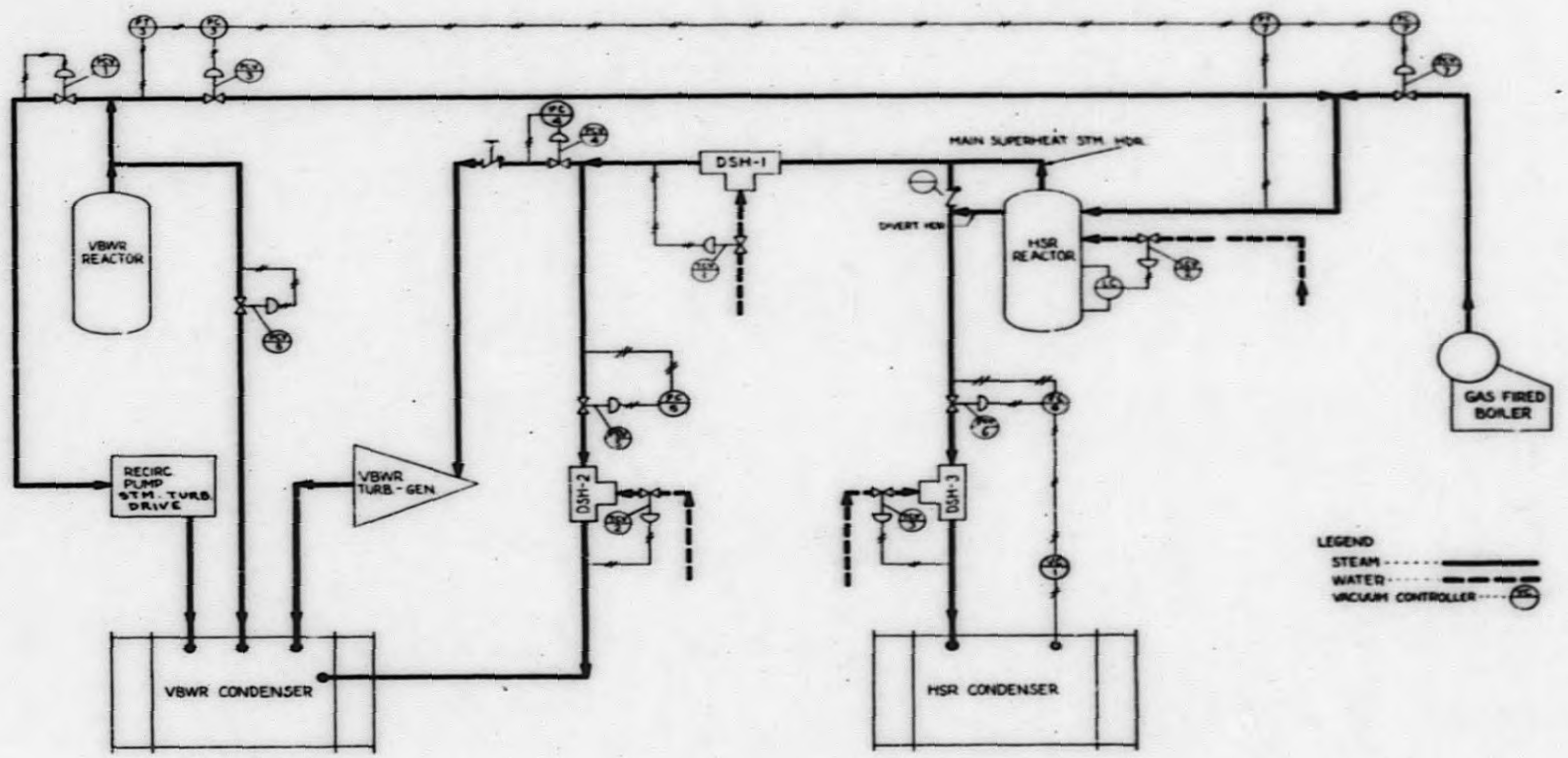


FIGURE II. 26
STEAM FLOW CONTROL SYSTEM

FIG. II. 26

7.8 Holding VSR inlet and outlet pressures constant has the effect of holding a constant differential pressure across the reactor. This has the desired effect of holding the flow in each of the core internal parallel flow channels independently constant.

4

J. WASTE DISPOSAL SYSTEMS

1. Radioactive Liquid Wastes

1.1 The liquid waste disposal system for the VSR will take maximum advantage of existing VBWR equipment. No additional collecting equipment is required except that certain piping additions and relocation are required, and a spent demineralizer resin storage and shipping facility for VSR is provided. The system is shown schematically on Figure II. 27.

1.2 Normally, the liquid wastes from the containment vessel sump, the boiler sump, the dump condenser building sump, the fuel storage pool drain and the VBWR flow to either or both of the existing 5,000-gallon waste water storage tanks. Wastes in these tanks are sampled to determine whether they should be demineralized for reuse, stored for some indefinite decay time prior to further treatment, or pumped to the 60,000-gallon tank for eventual truck disposal by a licensed radioactive waste disposal agency. If the results of this sampling indicate that the waste water is suitable for demineralizing and reuse, then that water is pumped through the existing VBWR filter and waste water demineralizer. Waste water, after demineralization, is routed to an existing 7,500-gallon tank. Here it is sampled to see that it meets condensate specifications and then is routed to either the existing VBWR 4,000-gallon condensate storage tank, or to the new 10,000-gallon VSR condensate storage tank. Otherwise it must be reprocessed. Wastes in process through the waste demineralizer may be recycled to the 5,000-gallon receiver tanks if initial conductivity is too high or when demineralizer breakthrough occurs.

1.3 Water in the shield tank above the reactor, which is used during refueling, is normally drained through the reactor cleanup system to the hotwell. From here it is processed through the condensate demineralizers and rejected by hotwell level control to condensate storage. Alternately, it may be routed to the waste system for processing. Water which must be drained from the reactor (such as at startup) follows this same route.

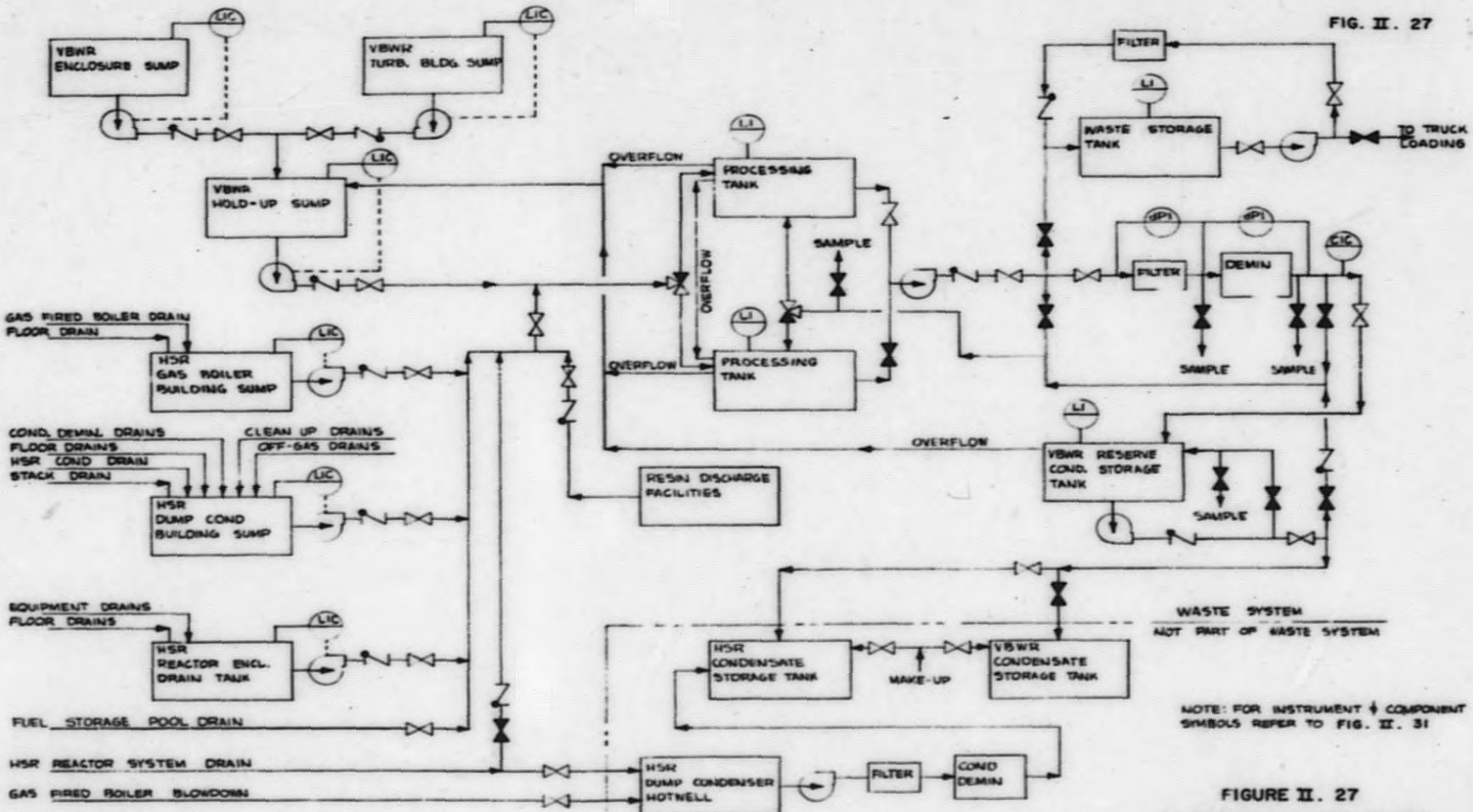
1.4 A filter recycle line at the 60,000-gallon tank is to be provided for the condition where the water to be disposed contains too much radioactivity. Recycling through the filter will tend to reduce the specific activity to a level permissible for truck disposal.

2. Radioactive Solid Wastes

2.1 A variety of radioactive solid wastes result from normal operation and maintenance of the VSR. Means for safe handling and disposal of these wastes will be provided to assure proper control and to prevent the spread of contamination.

2.2 Filter units and spent ion exchange resins from the water purification systems are the primary routinely occurring sources of solid wastes from the process. The filters in these systems are of the replaceable cartridge type and do not present a difficult handling problem. Each cartridge is a cylinder about three inches in diameter by ten inches long. Prior to replacing a filter, it is possible to take it out of service to permit a decay period while the duplicate unit is in service. Proper tools for handling the filter cartridges and suitable containers for storing them are used. The spent demineralizer resins are handled by sluicing from the demineralizers, de-watering, and transfer to suitable disposal containers.

FIG. II. 27



NOTE: FOR INSTRUMENT & COMPONENT SYMBOLS REFER TO FIG. II. 31

FIGURE II. 27

RADIOACTIVE LIQUID WASTE DISPOSAL SYSTEM

2.3 Other more-or-less routinely occurring wastes include filters from the off-gas and ventilation systems, contaminated clothing, tools, small pieces of equipment which cannot be decontaminated, and miscellaneous items such as paper and rags from contaminated areas. In addition, there may be an occasional relatively large piece of equipment which is more easily disposed of than decontaminated and repaired.

2.4 The solid wastes generated at the VSR facility will, in general, be disposed of using the methods, procedures and controls established at the VBWR. Solid wastes are transferred to a licensed radioactive waste disposal agency for off-site disposal. An area on the site is set aside as a radioactive waste collection area.

3. Radioactive Airborne Wastes

3.1 General

3.1.1 The gaseous waste disposal systems handle enclosure ventilation air and non-condensible gases from the process systems. Disposal system equipment provides for decontamination of gas streams potentially containing the relatively more hazardous radionuclides, short-term decay holdup, appropriate continuous monitoring equipment, and prevention of release to the stack if permissible discharge rates would be exceeded.

3.1.2 Release to atmosphere is via the VSR stack of 160 foot height. This stack will handle ventilation air from the VSR enclosure, VSR dump condenser off-gases, and the VBWR turbine condenser off-gases. VBWR enclosure ventilation exhaust, and VBWR turbine gland seal exhaust, will continue to discharge to the existing VBWR stack.

3.1.3 The plan for disposal of airborne wastes presented below is based on operation of the VBWR and VSR after the two systems have been tied together so that the VSR can utilize saturated steam from the VBWR. As indicated in Section II. G. 4, it is planned that the VSR stack would be used also for discharge of VBWR air ejector off-gases prior to hooking the two reactors together. The disposal plan presented would work equally well for independent operation of each reactor as it would for combined operation. For the purposes of this report, however, the disposal plan should be viewed only from the standpoint of independent operation of the VSR, but with the added recognition that VSR safety would not be affected if the VBWR were also to discharge wastes to the VSR stack. It is beyond the scope of this report to propose that the indicated off-gas system and plan be used for independent operation of the VBWR, or to evaluate the effect of such changes on the VBWR operation.

3.2 Off-Gas Rates and Compositions

3.2.1 The expected normal and maximum off-gas rates and compositions for the VBWR and VSR at full power operation are:

OFF-GAS RATES IN CUBIC FEET PER MINUTE*

	VBWR			VSR	
	Air Ejector Gases		Gland Seal	Air Ejector Gases	
	Normal	Maximum	Gases	Normal	Maximum
O_2	0.5	0.5		0.25	0.25
H_2	1.0	1.0		0.5	0.5
Air	0.5	5.5	20	0.75	7.5
TOTAL	2	7	20	1.5	8.25

* Radioactive gases are not included since their volume is negligible by comparison.

3.3 Off-Gas Process Flow

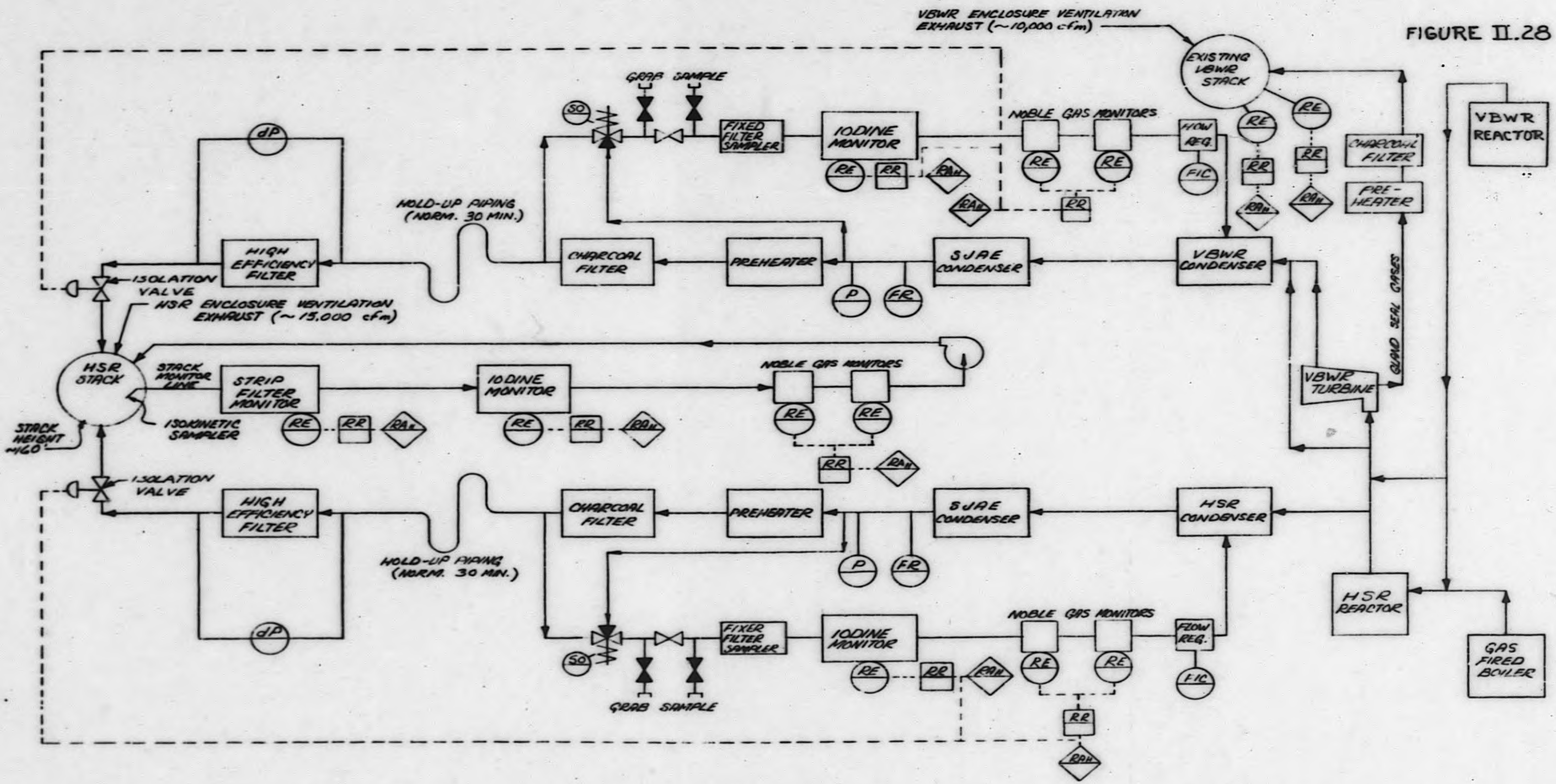
3.3.1 Figure II. 28 shows the combined off-gas and related systems associated with discharge from both the existing VBWR stack and the new VSR stack.

3.3.2 Non-condensibles from the VBWR turbine are removed by steam ejection, and moisture is then removed in a surface condenser. The resulting gas stream is routed through a preheater and activated charcoal filter to provide a substantial decontamination factor for any radio-iodines and particulates present, and is then routed through a buried holdup pipe which provides a normal 30-minute holdup before passage through a high efficiency filter and release to the VSR stack. A valve, to prevent off-gas release to the stack when required, is located on the off-gas line between the high-efficiency filter and the stack. Automatic actuation of the valve originates from the off-gas line monitors described below.

3.3.3. Process flow and equipment on the off-gas system from the VSR condenser is identical in function and feature to that for the VBWR condenser off-gas system shown on Figure II. 28.

3.3.4 No gland seal off-gas system is present in the VSR system. Gland seal off-gases from the VBWR turbine are routed via the existing VBWR off-gas holdup line (providing about 10 minutes decay) to the existing VBWR stack. As the probability of presence of radioiodines in this stream is increased when steam of VSR origin is routed to the VBWR turbine, a preheater and activated charcoal filter will be placed at the beginning of the existing VBWR holdup line.

FIGURE II.28



RADIOACTIVE AIRBORNE WASTE
FIGURE II.28

3.4 Off-Gas Radiation Monitoring

3.4.1 Existing VBWR stack monitoring equipment will be continued in service in accordance with existing specifications.

3.4.2 Each of the off-gas lines to the VSR stack (from the VSR dump condenser and the VBWR condenser) is provided with monitoring equipment which will establish the character of any radioactive materials present. Due to monitoring instrument requirements, the measurements are made following a two minute decay period, and therefore nominally 28 minutes prior to the time the monitored gases would be released to the stack. Sample flow may be taken either before or after the activated charcoal filter so that its radioiodine removal characteristics may be confirmed. The series of monitoring equipment includes (1) a fixed filter sampler for subsequent counting or analysis to determine presence of any radioactive particulates, (2) a continuous radioiodine monitor, alarm, and recorder, (3) two independent radiogas monitors, alarms and recorder, and (4) equipment for removal of spot samples of off-gas for special analyses. The radiogas monitoring equipment will provide the actuating signal for off-gas valve closure at the stack if the established maximum stack emission rate is indicated as being present in the initial portion of the holdup line.

3.4.3 The VSR stack monitoring system samples by use of an isokinetic probe located in the stack air stream at an elevation about 65 feet above the stack base, which is well downstream of all entering ducts and lines. Continuous monitoring equipment provided to back up the off-gas line monitors and to audit the release to atmosphere includes (1) a strip filter monitor, alarm and recorder for radioactive particulates, (2) a radioiodine monitor, alarm and recorder, and (3) two independent radiogas monitors, alarms, and recorder.

3.5 Permissible Stack Emission Rates

3.5.1 The presently specified permissible emission rates for the existing VBWR stack will remain in effect. Permissible stack emission rates for the new VSR stack have been determined on the basis of the stack height of 160 feet, an exit velocity of not less than 50 feet per second, the topography of the Vallecitos site and environs, and annual wind direction and atmospheric diffusion diversity factors inferred from more than three years of wind records for the site.

3.5.2 The VSR stack will be operated in accordance with the permissible emission rates tabluated below. These limits are specified individually because, due to the differing modes of exposure and differing critical organs, their simultaneous usage would produce exposures which are largely independent of each other.

	Emission Rate, μc/sec.	
	Annual Average	Initiate Valve Closure
Radiogases (based on noble gases)	200,000	2,000,000
Radioiodines	200	2,000
Iodine-131	25	
Long-lived Particulates	80	

The general limits for noble gases and radioiodines apply as measured at the stack after 30 minutes decay in the off-gas lines. Separate higher limits are established for control purposes for the individual off-gas line monitors at two minutes decay.

3.5.3 Actual annual average discharges will normally amount to only a small fraction of the permissible emission rates given above. Due to the 30-minute decay periods in the off-gas lines, short-lived activation gases from irradiation of reactor water and steam, including N-16, N-17, and O-19, are essentially completely decayed before release. A-41 emission is expected to be the order of 10 μc/sec, and therefore is negligible. Stack emission of 10 minute half-life N-13 is not expected to exceed the order of 100 μc/sec. The radiological characteristics of N-13 are such that the use of the general radiogas permissible emission rate based on the equilibrium mixture of noble gases, is conservative. Thus, essentially all of the permissible emission rates are reserved for conditions involving fission product release from defective fuel.

3.5.4 Further evaluation and justification of the stack limits indicated in paragraph 3.5.2, above, are presented in subsections 3.6 through 3.10 below.

3.6 Effects of Vallecitos Meteorology and Topography

3.6.1 The three-year wind record at Vallecitos shows:

Direction Wind From	Percent of Time 1957	1958	1959	Three Year Average
N	4	7	7	6
NE	8	6	8	7
E	6	4	4	5
SE	3	7	6	5
S	10	11	8	10
SW	32	33	32	32
W	12	12	10	11
NW	9	3	3	6
Calm	16	17	22	18

3.6.2. The significance of a stack release is dependent upon the effective height of release from the stack, wind speed, the distances to the site boundary in various directions, the ground elevations both on the site and for a reasonable distance beyond the site boundary, and recognition that the permissible off-site dose is 500 mrems per year. The topography of interest for VSR stack release, together with the effective elevations of the plume with various wind speeds are:

Direction Wind From	Distance to Site Boundary	Elevation at Site Boundary	Downwind Ground Elevations							
			1000'	2000'	3000'	4000'	5000'	6000'	7000'	8000'
N	2000'	475'	525'	475'	450'	430'	430'	480'	550'	480'
NE	4400	405	520	450	430	410	400	380	350	320
E	4000	475	540	520	470	475	440	480	520	440
SE	9200	520	640	620	630	620	640	700	680	570
S	6600	1000	800	800	800	1080	1100	1050	1060	1180
SW	5400	1200	700	920	880	1080	1240	1140	920	960
W	3800	600	600	670	680	620	610	680	620	680
NW	2000	515	540	515	500	550	560	610	640	520

Stack Elevations

VSR Stack Base Elevation ~545'
VSR Stack Top Elevation ~705'

Effective Stack Height Elevations (Exit Velocity of 50 feet/second)

Wind Speed		Effective Height
5 m/s	(~ 11 mph)	720'
3 m/s	(~ 7 mph)	735'
1 m/s	(~ 2 mph)	800'
0.45 m/s	(Calm, < 1 mph)	> 900'

3.7 Significance of Various Wind Directions

3.7.1 The data indicate that calm (less than one mile per hour wind speed) conditions prevail about one-sixth of the time. Thus, a condition of maximum wind direction diversity over the period of a year would be present if the frequency of winds, on an eight-point compass, was about 10 or 11 per cent for each direction. It is seen that all directions except S, SW, and W have frequencies significantly less than 10 per cent. Winds from S and W are prevalent just about 10 to 11 per cent of the time, while SW winds are prevalent about one-third of the time. The low frequency winds from N, NE, and E have associated favorable downwind ground elevations such that the effective height of the plume as it moves downwind will further minimize the significance of emissions in these directions compared to certain other directions. In addition, winds from these three directions will most likely be of low velocity and frequently associated with stable atmospheric conditions so that the plume would be expected to remain aloft with little, if any, diffusion to the ground at the lower downwind ground elevations.

3.7.2 While the ground elevations are not as favorable with regard to winds from the SE, the ground will be always below the expected plume height with low wind velocities, and a distance to the site boundary of over one and one-half miles is available.

3.7.3 When winds are from the western half of the compass, higher wind velocities and atmospheric diffusion almost always better than inversion will prevail. When winds are from the NW, the favorable downwind ground elevations, considering that the effective plume height will be in the range of 700 to 750 feet, minimize the effects of stack release in this direction.

3.7.4 The most limiting conditions, based on wind direction frequencies and downwind ground elevations, are judged to exist when wind is from the SW or W. Based on Neiburger's analysis as given in Appendix B of the VBWR Final Hazards Summary Report, Document SG-VAL 2, Third Edition, winds from these directions would be expected in clear weather as typical daytime directions, both in the warm and cold seasons, and in the earlier portions of the nighttime during the warm season, prior to the establishment of the surface-based inversion (which would be accompanied by wind reversal and drainage winds generally from the east).

3.7.5 During periods of wind from the SW, prevalent about one-third of the time, effluents would be carried upslope over the summits approximately one mile away at the plant boundary. During such periods, the effluents would be expected to mix vertically up to the base of the subtropical inversion (1,000 to 3,000 feet above the terrain), the concentration approaching uniformity in the vertical direction, and to spread laterally to a width of one-fourth to one-half of the distance travelled. Thus, if VSR stack release were at the annual average permissible noble gas release rate of 200,000 microcuries per second, the concentration leaving the plant boundaries would be approximately 10^{-8} microcuries per cubic centimeter. This would be equivalent to a dose rate from the cloud of about 6×10^{-3} millirads per hour; which, if continued for the third of the year in which this wind direction is expected, would yield a maximum annual dose of about 20 millirads.

3.7.6 The most critical combination of wind direction frequency and off-plant ground elevations is judged to be when wind is from the west. Considering that neutral diffusion will be the least favorable with this wind direction (C^2 is 0.01 and n is 0.25), and considering that the wind velocity may be as low as three meters per second, the downwind air concentrations based on a stack emission rate of one microcurie per second have been evaluated. For this case, the center line of the plume from the stack is expected to be approximately at the 750 foot elevation, and it has been assumed that the plume center line stays at this elevation and does not rise as the plume travels over higher terrain to the east. The expected concentrations at the ground at various distances to the east are:

Worst Case Diffusion, West Wind, Neutral Conditions

Stack Emission of 1 μc/sec at 750-foot Level

Downwind Distance		Plume Center-line Height Above Ground,	Downwind Concentration, μc/cc	
Feet	Meters	Meters	Plume Center-line	At Ground
1,000	305	46	9.5×10^{-10}	7.6×10^{-14}
2,000	610	25	2.8×10^{-10}	1.2×10^{-10}
3,000	915	21	1.4×10^{-10}	1.1×10^{-10}
4,000	1,220	40	8.3×10^{-11}	4.4×10^{-11}
5,000	1,525	43	5.6×10^{-11}	3.4×10^{-11}
6,000	1,830	21	4.1×10^{-11}	3.8×10^{-11}
7,000	2,135	40	3.2×10^{-11}	2.5×10^{-11}
8,000	2,440	21	2.5×10^{-11}	2.4×10^{-11}
15,000	4,600	0	8.2×10^{-12}	8.2×10^{-12}
30,000	9,150	0	2.5×10^{-12}	2.5×10^{-12}

The worst environs ground concentration is just off-plant at the 4000-foot distance with a concentration of 4.4×10^{-11} microcuries per cubic centimeter for an emission of one microcurie per second.

3.8 Radiogas Limit (Based on Noble Gases)

3.8.1 With a 30-minute hold-up of noble gases prior to stack release and the three meter per second wind speed, the age of the noble gas mixture at the worst off-site point determined in paragraph 3.7.6, above, will be about 3000 seconds. Assuming the least favorable equilibrium mixture of noble gases, a dose rate of one millirad per hour will be received from a concentration of 1.6×10^{-6} microcuries per cubic centimer with infinite cloud geometry, and 5.3×10^{-6} microcuries per cubic centimeter for the finite cloud dimensions expected in this case. Therefore, the one microcurie per second emission rate will deliver a dose rate of about 10^{-5} millirads per hour. Thus, the always-safe emission rate for these conditions, to deliver a dose rate of not more than 0.05 millirads per hour (500 millirads per year), is 5000 microcuries per second. This line of reasoning is conservative inasmuch as it is based on an "equilibrium" mixture, rather than the more rapidly decaying "diffusion" mixture of radiogases which has been observed experimentally at VBWR.

3.8.2 Considering that the west wind direction probability is 11 per cent, that the neutral plume will diffuse in approximately a five-degree angle, and that an eight-point wind direction represents 45 degrees, the actual wind direction diversity associated with a west wind is about a factor of 10. It is conservatively assumed that neutral conditions will prevail as much as one-fourth of the time when a west wind is blowing. The remainder of the time, lapse diffusion conditions are expected with this wind direction, with much lower concentrations at this distance. Thus, an overall diversity factor due to both wind direction and diffusion diversity of about 40 applies. Therefore, the VSR stack annual average permissible noble gas emission rate is 200,000 microcuries per second as measured at 30 minutes decay. The instantaneous permissible noble gas emission rate from the stack is taken as a factor of 10 higher, or two curies per second.

3.9 Radioiodine Limit

3.9.1 In a boiling water reactor, a natural protection against iodine release to the stack is available by means of the primary water to primary steam decontamination factor, and the scrubbing effects of the main condenser and the air ejector condenser. The first such protection mechanism is not available in a superheater reactor, and therefore iodine traps are provided in the off-gas lines. Based on the study of diffusion and topography given above for the noble gases, an emission of about 100,000 microcuries of the equilibrium mixture of iodines per second would be limiting with regard to direct radiation from the atmospheric cloud. However, much lower limits are appropriate considering direct inhalation by man with a permissible limit due to thyroid deposition of 500 mrems per year for general public exposure. This dose would result from continuous exposure to a concentration of equilibrium mixture iodines of about 2.5×10^{-10} microcuries per cubic centimeter. Based on the worst case trajectory with a west wind, the permissible stack emission of iodines is about 200 microcuries per second as measured at 30-minutes' decay. The Iodine-131 component of this permissible emission rate is about 25 microcuries per second. Based on the iodine removal features of the process, the I-131 emission should be less than 0.1 μc/sec when the stack is emitting the average permissible release rate of noble gases.

3.10 Emission Limit for Particulates

3.10.1 The appropriate limit for particulates in air for general public continuous exposure is 1×10^{-10} microcuries per cubic centimer as long as the Strontium-90 contribution does not exceed 1×10^{-11} microcuries per cubic centimeter and as long as the alpha emitter contribution is small. From considerations outlined above, this leads to a permissible average stack emission rate of 80 microcuries per second. If a longer-lived particulate emission of this magnitude is ever indicated by the stack monitoring system, periodic detailed analysis of the principle contributors would be made.

3.11 Control Limits for Off-Gas Line Monitors

3.11.1 Control of stack emission is provided, nominally 28 minutes prior to release to the stack, at the off-gas line radiogas and radioiodine monitors. Based on equilibrium fission product mixtures of noble gases and radioiodines, the off-gas line activity flow rates at two minutes decay equivalent to the permissible stack emission limits are:

	Off-Gas Line Activity Flow Rate, μc/second	
	Annual Average	Initiate Valve Closure
Radiogases	325,000	3,250,000
Radioiodines	250	2,500

3.11.2 It is expected that control will be based on the presence of gamma energies from certain of the fission products. As an example, if control for radiogases is based on the 0.42 Mev gamma available in 100% of the disintegrations of Xe^{138}, and as Xe^{138} is 14% of the equilibrium noble gas mixture at two minutes decay, a flow rate of Xe^{138} of about 45,000 μc/sec would be equivalent to the permissible annual average stack emission rate.

3.11.3 Because parallel flow through the two similar off-gas systems occurs, alarm levels on each system will be established at one-half of the permissible annual average flow rate, so that operator attention will be directed to any release which might approach the permissible long-term average. Due to the small probability that flow rates above the permissible annual average would occur in both lines simultaneously, and as operator attention would be constant under such conditions, it is not considered necessary to apply a safety factor of two on the level which would automatically initiate closure of the appropriate holdup line isolation valve. Plant operations will be controlled so that the permissible annual average emission rates are not exceeded.

3.12 Environs Monitoring

3.12.1 All of the above permissible emission rates are based on a combination of theoretical and practical considerations. The true indication of the long term acceptability of these emission limits is given by a proper site and environs monitoring program. Such a program must prove that the actual doses at points of general public exposure are within the acceptable licensing limits.

3.12.2 Since noble gases are the radionuclides which may contribute the more significant portion of possible environs dose, careful continuous monitoring for this mode of exposure is essential. As radioiodines, particularly Iodine-131, may be second in importance, this type of monitoring also is planned. Radiogas monitoring is performed with pairs of detached ionization chambers, either gamma- or beta-gamma-sensitive with a capacity of about 10 mrads, placed at suitable locations in the approximate five-mile radius from the point of emission. Radioiodine monitoring is performed by vegetation sampling at the same network of positions. The natural background shown by these monitoring methods will be established a reasonable period in advance of operation.

3.12.3 Present plans are to establish radiogas monitoring at approximately 20 to 30 positions in the five-mile radius, establish the background, and continue operation of the network during facility operation. Following establishment of radioiodine background, sampling will be continued at a frequency commensurate with the degree of need indicated by the off-gas line and stack radioiodine monitors.

3.12.4 Integration of monitoring network results with the stack emission continuous monitors and the plant site wind direction recorder provides the essential information for adjustment of the permissible stack emission control limits upward or downward as is appropriate to provide a high degree of assurance that general public radiation exposures are always well within permissible annual dose regulations.

K. MISCELLANEOUS PLANT SYSTEMS

1. Electrical System

1.1 A 1500 kva indoor 12 kv transformer furnishes 480-volt, 3-phase, 60-cycle power for most of the plant auxiliaries. Lighting is supplied through step-down transformers. Power for the reactor instrumentation and controls is shown on Figure II.24.

1.2 The transformer is fed by tapping the VBWR 12 kv feeder using an overhead pole line to the VSR fence boundary north of the transformer location, and from there underground to the load center in the dump condenser and miscellaneous equipment building.

1.3 The 1500 kva transformer capacity was selected on the basis of a maximum normal load of about 1400 kva. The total connected load is approximately 1900 kva. A battery and battery charger will supply d-c for some controls. A 75 kw diesel-generator is furnished to supply a-c power to essential services on loss of main power supply. The loads are described in Section II.F.4.

2. Service Air System

2.1 The service air system planned for the VSR plant will be an extension of the VBWR system, and no new compressors or receivers are planned to be installed for service air.

2.2 The VSR instrument air supply system will have its own air compressor and receiver, and will be provided with a backup supply through interconnection with the VBWR instrument air supply system.

3. Ventilation System

3.1 The control room is provided with an air conditioning unit to provide dust-free atmosphere of even temperature for the instrumentation as well as for operator comfort.

3.2 The containment vessel will be provided with a ventilation system of approximately 15,000 cfm air flow, which will result in several air changes per hour. The ventilating system will consist of a supply blower, an evaporative cooler, a gas heater, containment isolation valves on inlet and outlet exhaust lines, and an exhaust blower. The exhaust air will enter the new VSR off-gas stack and be discharged approximately 160 feet above grade. This part of the system is described in more detail in Section II.J.3.

3.3 The dump condenser and miscellaneous equipment building contains conventional industrial ventilation blowers.

4. Fire Protection System

4.1 Fire protection equipment and procedures established for the VSR are similar to the Conventional Industrial Plant fire protection equipment and procedures which have been established for the existing Laboratory site, including the GETR and the VBWR. Equipment, buildings, procedures, etc., are in accordance with Company-wide standards, state and local regulations, and the recommendations of insurance agencies.

4.2 A 6-inch fire main loops the VBWR site and is extended for the VSR installation. Fire hydrants and hoses are strategically located on the extended loop.

4.3 A 500,000 gallon raw water storage tank located on an adjacent hillside approximately 100 feet above the reactor level is provided; 100,000 gallons of which are reserved for fire protection.

4.4 Conventional portable fire extinguishers are located in the buildings and in the containment vessel, and are properly labeled as to their use.

4.5 A fire brigade has been established at the Laboratory, made up of plant personnel who are trained in the use of the equipment for the specific hazards which exist at the plant. A fire truck designed specifically for the site is part of the equipment in addition to the conventional extinguishers and hoses.

4.6 In the event of a fire, the shift supervisor of the facility is in charge of directing the fire protection effort. The fire brigade reports to him for orders, and the local firemen work under his direction should the situation require outside help.

5. Communication System

5.1 A communication and paging system is provided which has loudspeakers and sensitive microphones. Switches at a master station in the control room permits the operator to page each of the stations selectively, or all the stations at once. Stations are established in the boiler area, the dump condenser building, the control rod drive room, the containment building operating floor area, and at the change room.

L. PLANT STRUCTURES

1. General Layout

1.1 The overall site layout is shown on Figure II.1. The VSR facility will be located to the east of the present VBWR facility. The principal plant structures are a reactor containment vessel, a dump condenser and miscellaneous equipment building, an off-gas stack, an additional cooling tower, a fresh fuel vault, and an extension of the VBWR control building. An outdoor-type gas-fired boiler is provided also. The reactor containment building will be approximately due east of the present VBWR turbine building. The off-gas stack will be approximately north of the new reactor containment vessel. The gas-fired boiler will be east of the off-gas stack. The dump condenser and miscellaneous equipment building will be south of the gas-fired boiler and east of the reactor containment vessel. The new cooling tower will be east of the dump condenser and miscellaneous equipment building. The fresh fuel vault will be located to the northeast and will be more than one hundred feet from any normally occupied building on the site.

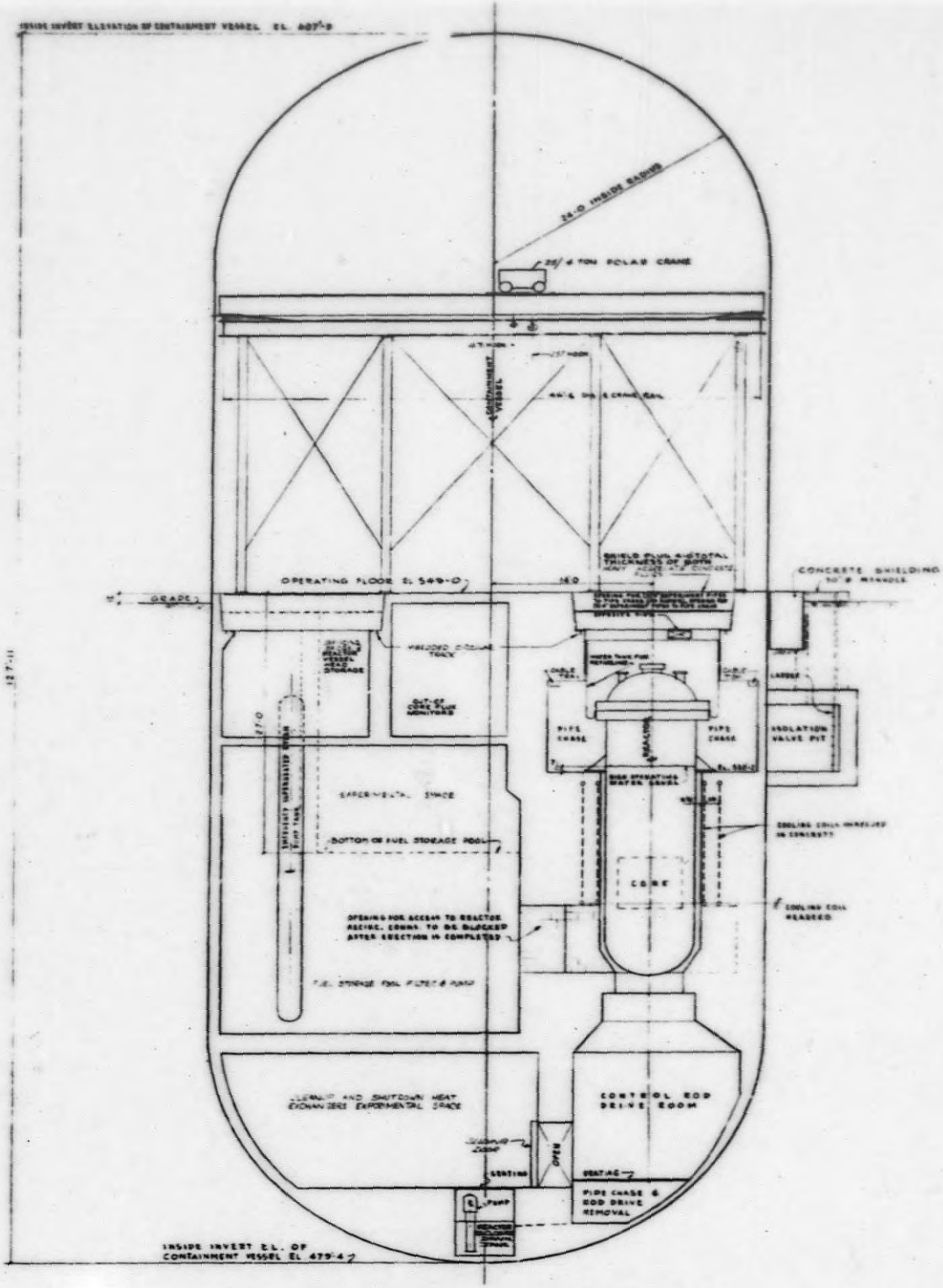
1.2 Space has been provided for additional capacity to be added to the gas-fired boiler, dump condenser, and cooling tower.

2. Containment Building

2.1 The containment building (reactor enclosure) is an all-welded steel capsular pressure vessel approximately 128 feet in overall height and 48 feet in diameter, consisting of an 80-foot straight cylindrical section having hemispherical heads at the top and bottom. The enclosure is provided with two pressure tight air locks as well as penetrations for piping and electrical cables as required. Approximately 70 feet of the reactor enclosure is below grade level. A vertical section and plan view of the containment vessel are shown on Figures II.29 and II.30.

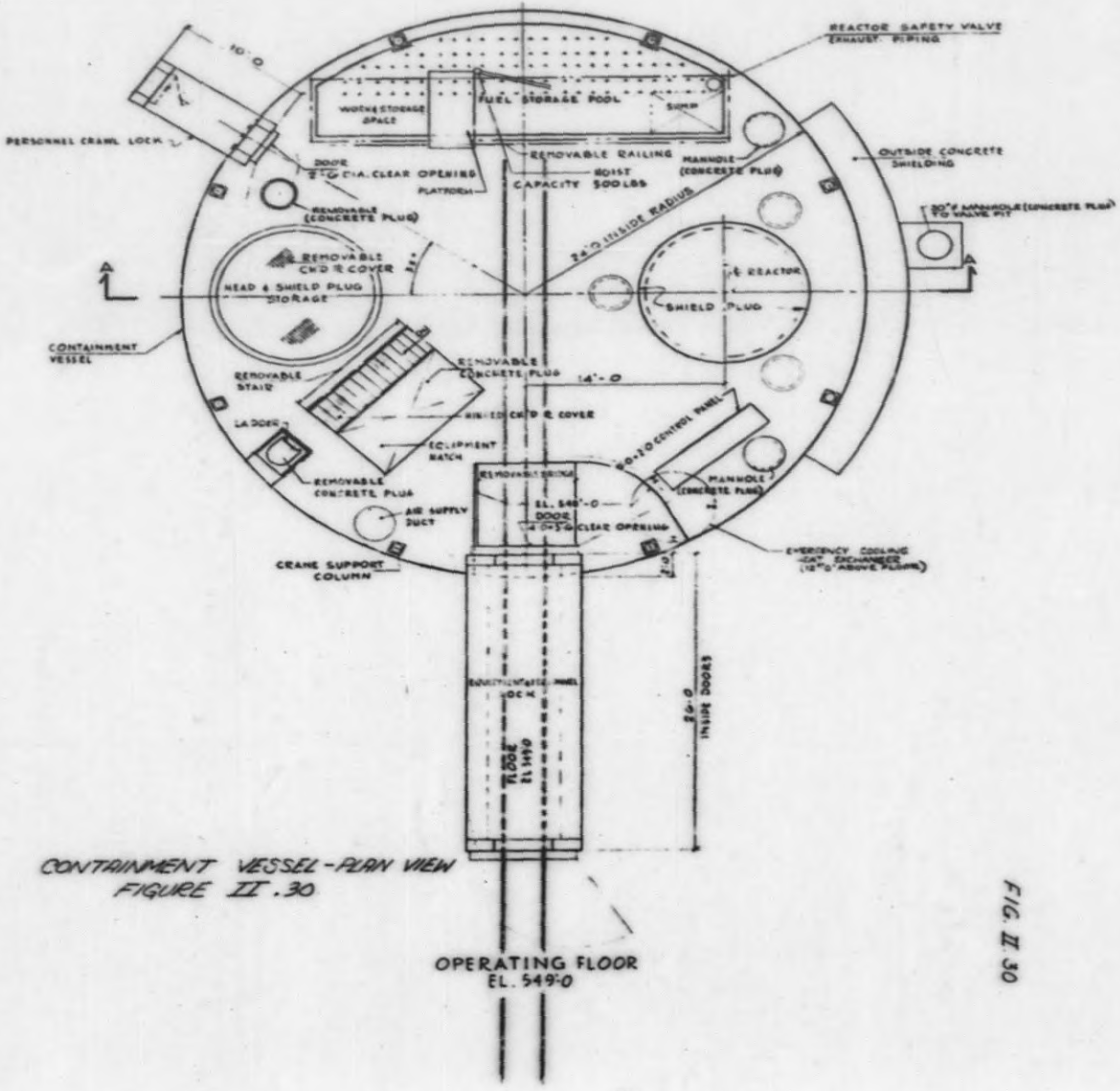
2.2 The containment vessel is designed for 58 psig with a coincident temperature rise of 250°F. This design pressure provides a margin above the minimum design pressure dictated by the design criteria and "maximum credible accident" analysis presented in Section IV of this report. The margin is available for possible changes in system or containment volumes, or for future modification to the reactor. Design, materials, fabrication, erection and inspection shall conform with the requirements of the ASME Boiler and Pressure Vessel Code, Section II, Material Specification, Section VIII, Unfired Pressure Vessels, and Section IX, Welding Qualifications, as amended by Code cases applicable to reactor containment vessels. The ability of the vessel to withstand a pneumatic test of 1.15 times the design pressure will be demonstrated, as will the maximum leak rate of not more than 1%/day of the contained volume at the design pressure.

2.3 Three floor levels in addition to the operating floor are provided inside the containment for equipment location, operation and monitoring. Construction below and including the operating floor is principally of reinforced concrete, for the floors, reactor shielding, equipment shield walls, columns, pilasters, etc. Provisions are made to install an additional floor level at some future date. Removable



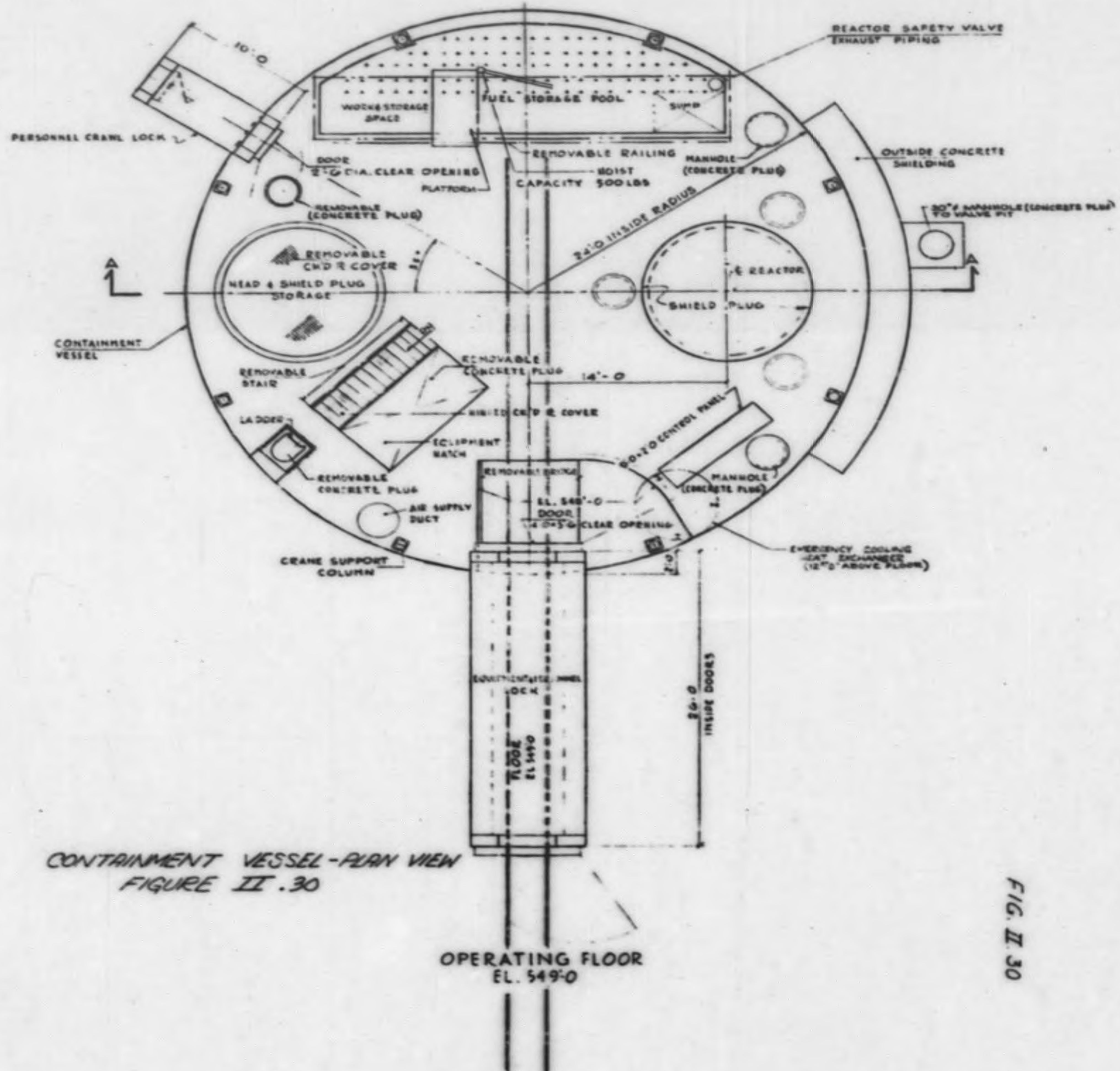
CONTAINMENT VESSEL- VERTICAL SECTION
FIGURE II .29

FIG. II 2



CONTAINMENT VESSEL - PLAN VIEW
FIGURE II .30

FIG. II 30



CONTAINMENT VESSEL-PLAN VIEW
FIGURE II.30

FIG. II 30

steel stairways provide personnel access to the various levels. The stairways are removable to provide for the infrequent movement of large equipment without sacrificing additional floor space. A steel ladder serves as access during periods when the stairways are removed.

2.5 Above the operating floor, structural steel framing supports a 25-ton polar crane which services the reactor for installation as well as the many crane services required by refueling and maintenance operations (Section II. E. 7). The crane has a four-ton auxiliary hook for miscellaneous services. A fuel storage pool servicing bridge and an associated electric hoist facilitates fuel movement within the pool that is provided inside the containment vessel.

2.6 Additional principal structural items associated with the containment vessel include the segmented shield plug for the reactor vessel well, a reactor vessel head and shield plug storage well, a circumferential pipe chase around the reactor vessel outlet steam header, vertical pipe chases and vertical access ports to areas below the operating floor, reactor enclosure drain tank and sump, and the containment isolation valve pit which is below grade level just outside the containment vessel.

2.7 The principal equipment located in the containment includes the reactor vessel and outlet steam headering system, control rod mechanisms including scram system and seal coolant system, reactor instrumentation (partial), reactor cleanup demineralizer and shutdown system (partial), fuel pool filter system, and the emergency cooling system. Space has been allocated for future equipment including reactor coolant recirculating pumps, in-core flux monitors, and other special test equipment and instrumentation.

2.8 The containment vessel will be provided with a ventilation system of approximately 15,000 cfm air flow as described in Section II. K. 3. The exhaust air from the ventilation system will be discharged from a stack approximately 160 feet above grade.

2.9 Piping and cables penetrate the containment with reinforced leak-proof fittings, meeting the vessel pressure and leakage criteria. Piping penetrating the containment is provided with automatic valving which insures the leak tight integrity of the enclosure if it should be required. Piping which connects directly to the reactor primary system and penetrates the containment shell or which opens into the interior of the containment vessel, has two isolation valves in series. At least one of the two valves closes automatically from an appropriate safety system signal. It is possible to close the other valve by manual actuation from a location that can be occupied during the immediate aftermath of any accident. Piping which does not connect directly to the reactor primary system but which, as a result of pipe failure inside the shell, might discharge radioactive material outside must have at least one shutoff valve which can be closed by manual actuation from a location that can be occupied during the immediate aftermath of any accident. The only exception to this latter requirement is the vent piping from the secondary side of the emergency heat exchanger which is exempted because this piping and the shell of the emergency heat exchanger are considered to be an extension of the containment shell for the purposes of isolation requirements. The emergency exchanger and vent piping are adequately protected against being broken in the event of a reactor primary system rupture.

2.10 One personnel and equipment air lock and one personnel crawl lock are provided. The larger lock is of the autoclave type, with both doors opening outward from the ends of the body of the lock, while the smaller lock is of the pressure-seal type which has both doors opening toward the inside of the containment vessel. Doors of each lock are operated manually and are interlocked so that the integrity of the containment vessel is maintained at all times. The larger lock requires power for operation and is provided with an emergency power source also, but the crawl lock is operated without the need for any power assist. An alarm system which operates on containment air pressure is used to indicate any gross violation of containment integrity such as having both doors of an airlock open at the same time. This alarm backs up the interlocking provisions of the air locks.

2.11 A containment spray cooling system is included in the plant whereby water is sprayed inside the containment vessel from a series of nozzles fixed to a circular header located above the operating floor adjacent to the containment wall. The pump which is provided takes suction from the 500,000 gallon Laboratory raw water storage tank. The containment spray is not required for pressure reduction of the containment vessel contents following a primary system rupture accident, but may be used if desired to aid the natural cooling and pressure reduction effects and to facilitate washdown of radioactive materials which may have been released inside the containment vessel.

2.12 All the welds made on the containment vessel will be radiographed. All unsatisfactory radiograph negatives will be rejected and the weld again radiographed. All welds which fail to meet Code specifications will be cut out, rewelded, and reradiographed.

2.13 Upon completion of the containment vessel, a soap bubble test at about 5 psig will be applied to all welds and seals. The test will be applied to each door of the air locks with the other door open.

2.14 After completion of the soap bubble test, and correction of any weld leaks which may be found, a high-pressure test will be made at a pressure equal to 115% of the design internal pressure of the vessel. Each door of the air locks will be tested separately and the equalizing valves of each lock will be tested.

2.15 After completion of the initial pneumatic test, a leakage rate determination will be made at a pressure not exceeding the design pressure. The leak rate test will utilize a leak-tight reference system within the containment vessel to provide temperature compensation during the test. The test data will be interpreted in terms of the percent per day of the vessel contents which would leak at the design pressure. In order to meet the safeguard design criteria for the reactor, this leakage rate must be determined not to exceed 1%/day.

2.16 When plant construction is essentially complete, a final leakage test will be conducted at about 10 psig internal pressure using soap bubble or halide leak detection methods to check all previously checked points of potential leakage that may have been disturbed during construction and all new welds and seals which are points of potential leakage.

3. Stack

3.1 The VSR ventilation and off-gas stack is constructed of reinforced concrete, and is of free standing, unlined design having the following size and capacity:

Height above grade	160 feet
Inside diameter at top	2-1/2 feet
Volume of exhaust gas	15,000 cfm
Maximum temperature of exhaust gases	150°F
Minimum thickness of concrete	6 inches

3.2 Lightning protection is provided. A galvanized steel ladder with safety cage is provided for the full stack height. A cast iron inspection access door is provided at the base of the stack. Provisions are made for radiation monitoring instrumentation inserts approximately 65 feet above grade. The Federal Aviation Agency has been contacted to determine whether there is a necessity for aircraft warning lights.

4. Other Structures

4.1 Some of the details of the other plant structures have been given in other sections of this report. The control building addition was described in Section II. H. 1. The gas-fired boiler, dump condenser and miscellaneous equipment building, and the VSR cooling tower were discussed in Section II. G.

M. SITE AND ENVIRONMENT

1. Site Location

1.1 The VSR will be located immediately adjacent to the VBWR facility on the site of General Electric's Vallecitos' Atomic Laboratory in Alameda County, California.

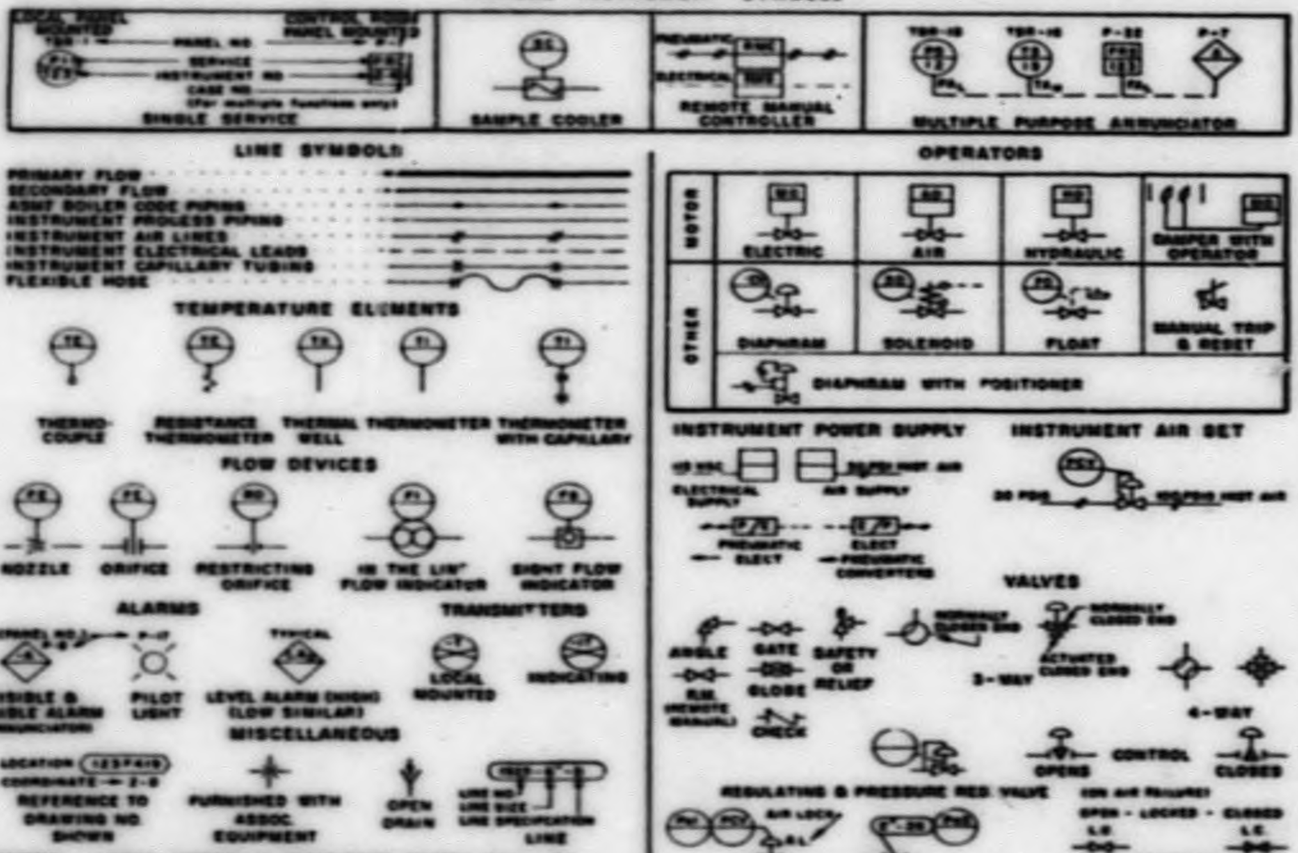
2. Site Characteristics

2.1 The significant features of the site and environment, required to be reported in accordance with the Regulations, have been described in the document SG-VAL 2 Third Edition, dated November 30, 1959, which is the current Final Hazards Summary Report for the VBWR facility. To avoid repetition, reference is made to that document.

2.2 The evaluations of the site with respect to the VSR facility are found principally in Section II. J. 3 and Section IV of this report. Section II. J. 3 describes the procedures and limits for safe disposal of radioactive air-borne wastes utilizing the good diffusion available with discharge of these wastes from the 160-foot-high VSR stack. Section IV describes the many features of the facility which minimize the possibility of a serious incident involving significant release of radioactivity to the environment, and evaluates the radiological effects in the environment of the occurrence of a "maximum credible accident" to the reactor.

INSTRUMENT FUNCTION / MEASURED VARIABLE		CONTROLLING: RECORDING	CONTROLLING: INDICATING	CONTROLLING: BLIND	CONTROLLING: CONTROL VALVE	MEASURING: RECORDING	MEASURING: INDICATING	MEASURING: OBSERVATION GLASS	MEASURING: PRIMARY ELEMENT	MEASURING: TEST POINT	MEASURING: TRANSMITTER	MEASURING: INTEGRATOR	MEASURING: AMPLIFIER	MEASURING: SAMPLER	SWITCH	SWITCH: INDICATING	SWITCH: BLIND	ALARMS: ALARM	ALARMS: ALARM LOW	ALARMS: ALARM HIGH	POWER SUPPLY
		-RC	-IC	-C	-CV	-R	-I	-G	-E	-X	-T	-Q	Am	Sm		-IS	-S	-A	A_L	A_H	
AIR	A													AS_m							
CONDUCTIVITY	C	CRC	CIC		CCV	CR	CI		CE	CX				CS_m		CIS	CS	CA	CA_L	CA_H	
DENSITY	D	DRC	DIC	DC	DCV	DR	DI			DX						DIS	DS	DA	DA_L	DA_H	
DIFF PRESS	dP	dPRC	dPIC	dPC	dPCV	dPR	dPI				dPT					dPIS	dPS	dPA	dPA_L	dPA_H	
FLOW	F	FRC	FIC	FC	FCV	FR	FI	FG	FE	FX	FT					FIS	FS	FA	FA_L	FA_H	
HUMIDITY	H	HRC	HIC			HR	HI		HE												
pH	pH	pHRC	pHIC	pHC	pHCV	pHR	pHI		pHE	pHX			pHA_m	pHS_m							
LEVEL	L	LRC	LIC	LC	LCV	LR	LI	LG			LT					LIS	LS	LA	LA_L	LA_H	
OXYGEN	O_2					O_2R	O_2I		O_2E												
PRESSURE	P	PRC	PIC	PC	PCV	PR	PI			PX	PT					PIS	PS	PA	PA_L	PA_H	
POSITION	Po					PoR	PoI				PoT						PoS				
RADIATION	R					RR	RI		RE	RX			RA_m	RS_m			RS	RA			RPw
SPECIFIC GRAVITY	Sg																				
SPEED	Sp	SpRC	SpIC	SpC		SpR	SpI		SpE												
TEMPERATURE	T	TRC	TIC	TC	TCV	TR	TI		TE	TX	TT					TIS	TS	TA	TA	TA	
TURBIDITY	Tu																				
VISCOSITY	V																				
VIBRATION	Vb																VbS	VbA			
WEIGHT	W																				
W[illegible]T FACTOR	Wf	WfRC	WfIC		WfV																
[illegible]ATER	Wa																				
															Suffix sub H or L may be added to designate High or Low						

TYPICAL INSTRUMENT SYMBOLS



NOTES: 1. FIRST LETTER OF INSTRUMENT SYMBOL SHALL BE PROCESS VARIABLE.
2. A SWITCH IS A DEVICE CAPABLE OF OFF - ON ACTION.
3. AN ALARM IS THE ANNUNCIATOR, SIGNAL HORN OR OTHER DEVICE, ACTUATED BY SWITCH CONTACTS.

FIGURE II. 31 VALVE AND INSTRUMENTATION LEGEND

PART III - OPERATING AND EMERGENCY PROCEDURES

A. INTRODUCTION

1. This section outlines the administrative and procedural controls used in operation of the plant, the general methods of conducting various phases of normal operation, the general approach to experimental operations such as will be associated with superheat fuel development, and the emergency procedures to be followed in the event of a serious accident to the reactor involving a significant radiation hazard. While at the present stage of design the operating procedures are necessarily very preliminary, an attempt is made to emphasize those procedures which have particular safety significance for a superheat reactor of the type under design, e.g., the procedures for safely flooding or unflooding the superheat steam passages of the core. The principal purpose of presenting procedures at this stage of design is to impart the necessary perspective to the descriptive and safety analysis sections of the report.

B. BASIC OPERATING PRINCIPLES

1. The basic operating principles for the VSR include the following:

1.1 Before being placed into regular operation, the plant will be subjected to thorough individual component and systems tests and an initial start-up program.

1.2 Operation and control of the reactor and most of the process equipment is to be centralized in the control room.

1.3 The control room will be manned at all times during operation by at least one licensed operator.

1.4 The control room staff will direct or have prior knowledge of any main process functions performed at remotely operated panels and valve racks not in the control room.

1.5 Startup, normal shutdown, and all other repetitive operations will be performed in accordance with specified check lists.

1.6 With some exceptions, the reactor containment building and other plant buildings will be habitable for limited periods of time during normal operations. Radiation monitoring by fixed or portable instrumentation will be provided for entry to all radiation zones.

1.7 Maintenance of most facilities outside the primary shielding may be undertaken by contact methods after a short shutdown period.

1.8 All tests and routine maintenance of protective devices and critical operating equipment will be done in accordance with a firm schedule.

1.9 All personnel leaving a radiation zone, as well as equipment being taken from a radiation zone, will be appropriately surveyed to insure control of contamination.

1.10 Irradiated fuel is transferred between the core and the fuel storage pool, making use of a shielded transfer cask. To assure the protection of the environs against a refueling accident, containment integrity provisions will be in effect during all fuel transfer operations.

1.11 Operation of the radioactive waste handling system will assure that disposal of radioactive materials will not result in the exposure of any persons on or off the plant site in excess of permissible limits. Liquid wastes are handled in discreet batches to facilitate control. Gases and airborne wastes are continuously monitored and discharged from a high stack to facilitate atmospheric dispersion. Solid wastes are removed to a designated area and transferred to a licensed radioactive waste disposal company.

1.12 The plant is so protected by automatic safety devices that no single operator error or a reasonably conceivable combination of operator errors could cause a severe accident.

1.13 All unexpected incidents, unsafe acts, or incidents of excessive exposure to radiation will be fully investigated to effect remedial procedures to prevent recurrence.

1.14 In the event of any situation which may compromise the safety of continued operation, it will be a required procedure to shut the plant down as quickly as the situation calls for and to take other planned emergency action to protect persons and property.

2. The balance of Section III of this report amplifies these principles and discusses the procedural measures incorporating them.

C. OPERATING STAFF

1. Organization

1.1 The testing, startup, and operation of the reactor will be accomplished by a carefully selected organization composed of personnel with special training and experience. A tentative organization chart is shown on Figure III.1 to illustrate how the operation probably will be organized. The VSR organization for startup and initial operation is expected to be essentially the same as that of the VBWR and GETR.

1.2 The Manager and personnel of the VSR organization will be provided necessary supporting services from existing functional groups at the Vallecitos Atomic Laboratory. Also readily available will be technical and operating advice and assistance from GETR and VBWR technical and operating personnel, from experienced consultants, engineers and scientists at the Vallecitos Atomic Laboratory, and from the design engineering and manufacturing facilities of General Electric's Atomic Power Equipment Department centered in San Jose, California.

1.3 Technical audit of reactor operations performance will be provided by the Nuclear Safety organization which currently performs this function for VBWR and GETR. In addition, the Laboratory Safeguards Group and the General Electric Technological Hazards Council will provide continuing counsel and review of the reactor operation.

1.4 The Laboratory Safeguards Group consists of senior members of the Laboratory selected on the basis of their experience, knowledge, and judgment. This committee's function is to counsel the operating organizations of the Laboratory in regard to the safety aspects of proposed reactor operations and experiments. This includes review and approval of all new or proposed revisions to the standards for the reactor and experiments.

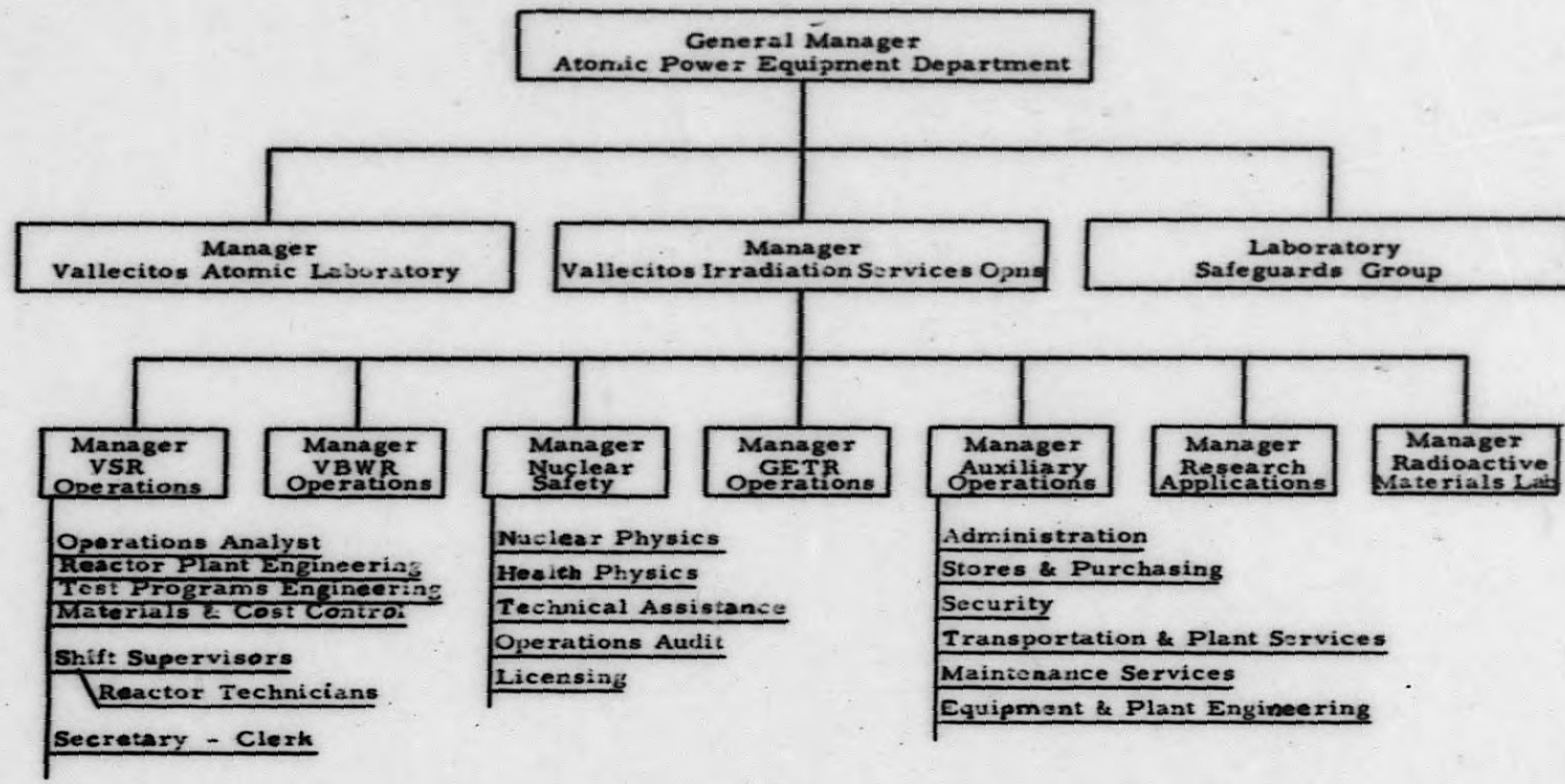
1.5 The General Electric Technological Hazards Council consists of representatives from each department of the Company's Atomic Products Division and several other components of the Company who are particularly suited for this assignment by reason of their experience and knowledge in this field. The functions of this Council include furnishing advice to Company managers on all matters relating to reactor safeguards, reviewing and recommending reactor safeguard design criteria and operational limits, and participating in appropriate safeguard studies conducted or sponsored by Government agencies or industry-wide activities.

2. Responsibilities

2.1 The Manager - Vallecitos Superheat Reactor Operation will be a man with extensive reactor operating and management experience. He is responsible to the Manager - Vallecitos Irradiation Services Operation for the safe, efficient operation and control of the reactor; for the handling of fuel elements; for the disposal or shipment of radioactive materials; and for

FIGURE III.1

TENTATIVE FUNCTIONAL ORGANIZATION CHART



the establishment and control of operating and radiation control procedures for all personnel working within the reactor environs. He has proprietary responsibility for the upkeep of the reactor, instrumentation, controls, auxiliary equipment, experimental equipment, and containment building. The Manager - VSR Operation will staff the organization with operating personnel responsible for carrying out these functions and responsibilities on a 24-hour-a-day basis. He will provide the necessary planning, coordination, training and instruction. The following key personnel are included in the VSR organization:

2.1.1 Shift Supervisors, responsible to the Manager - VSR Operation, provide overall supervision of the reactor operation on shifts and weekends, administer planned work and handle emergencies. These supervisors will have reactor operating experience, supervisory abilities, and good technical backgrounds. They are responsible for safety and continuity of all reactor plant operations on their assigned shifts.

2.1.2 The Operations Analyst investigates and analyzes performance of reactor systems and methods of operation and recommends corrective action where appropriate for efficient and safe operation, provides advice and assistance to Shift Supervisors, develops operating procedures for approval, coordinates the training program and performs various liasion and administrative functions as directed.

2.1.3 The Reactor Plant Engineer is responsible for scheduling and coordinating the maintenance of the overall reactor plant, for performance of engineering work required for the maintenance of reactor process equipment, and for support in planning alterations or expansions of the nuclear facility.

2.1.4 The Specialist - Materials and Cost Control is responsible for procurement, storage and shipment of all materials including reactor fuel, operating supplies, equipment parts, liquid and solid radioactive waste. He is the official SF material custodion and is responsible for property accountability. He is also responsible for cost analysis and control and performs various administrative duties as directed.

2.1.5 The Test Programs Engineer programs the use of the reactor and its experimental facilities for maximum utilization of the installed capacity. He receives and reviews all customer inquiries and accepts commitments within approved limits for use of irradiation space and reactor facilities. He is responsible for an engineering evaluation of proposed test facilities and irradiation experiments. He determines design safety and compatibility of proposed experiments. He also performs any required technical liasion with customers to supply technical advice on proposed experiments.

2.2 As indicated by the organization chart, services are furnished by other components of the Vallecitos Irradiation Services Operation organization, under the direction of the reactor management. These services include the following:

2.2.1 Nuclear Physics evaluates the physics aspects of operation for safe, efficient control of the reactor process on a continuing basis. He schedules the positioning, recycling, and replacement of fuel elements in the reactor and maintains records of fuel burnup and performance. He predicts startup reactivity requirements for the use of operating personnel and determines flux received by installed experiments or to be received by planned experiments.

2.2.2 Maintenance maintains and repairs plant and equipment under the direction of the Reactor Plant Engineer.

2.2.3 Health Physics establishes radiation protection standards, recommends radiation controls, provides radiation monitoring, consults on possible radiation hazards, and audits results of health physics programs.

2.2.4 Security provides a system of checks to assure that the security of plant facilities, fissionable materials, and classified information is adequately protected.

2.2.5 Fire Protection provides auxiliary fire fighting forces and training programs in fire prevention and control.

2.2.6 Analytical laboratory service and consultation is furnished by the Vallecitos Atomic Laboratory organization.

3. Licensing and Training

3.1 An intensive training program will be initiated in 1961 to train personnel for the initial operation of the facility. This program will continue, on a lesser scale, as required to insure well-trained replacements and to maintain a knowledgeable staff. Education of personnel on the operating characteristics of the reactor and training each man to accomplish his assignment in a safe, efficient manner is the most important single task to be accomplished in preparation of the staff for initial startup and operation. Key members of the organization will have comprehensive reactor experience. It is expected that these would be people now holding operator licenses at another reactor facility. These people will formulate and conduct a training program for all those new to the operation and will provide a retraining program for those who have had previous experience elsewhere. All personnel will be thoroughly indoctrinated in the safety policies and process standards for operation. As part of this training program, individuals will be given assignments to develop operating procedures, check sheets, and various other controls which will assure operation within process standards. Reactor personnel will participate in the pre-operational test phases of the facility activation. This will enable them to become familiar with the plant systems as a whole and with individual characteristics of the various pieces of plant equipment. The final training of reactor personnel will occur during the start-up and testing of the plant with the advice and consultation of fully qualified General Electric personnel.

3.2 Operators' licenses will be acquired in accordance with Title 10, Chapter 1, Part 55, Code of Federal Regulations.

D. ADMINISTRATIVE PROCEDURES

1. Operating Policies, Standards, and Limitations

1.1 The safety of operating personnel and of people in surrounding areas is a vital consideration of the General Electric Company in operating reactors at the Vallecitos Atomic Laboratory site. This safety policy is implemented by means of the following measures:

1.1.1 Strict adherence to the technical and operating limitations which become a part of the operating license issued by the AEC.

1.1.2 Establishment of process standards which specify the limits and manner in which nuclear, experimental, and process variables must be controlled for safe operation. The process standards must fall within the license limitations.

1.1.3 Establishment of operating procedures which supplement the process standards and outline the pertinent steps to be followed in operating the reactor and experimental facilities. These procedures are designed to assure operation within the limits set by the process standards.

1.1.4 Assignment of overall responsibility for the safe operation of the physical plant and experiments to a single organizational component.

1.2 The above policy statements will be administered in the following manner:

1.2.1 The reactor organization prepares process standards and submits them to the Laboratory Safeguards Group for review and approval. Following such approval, the authority for the establishment of a process standard is vested in the Manager - Vallecitos Irradiation Services Operation. Once established, process standards will be observed without exception until modified by the same approval authority.

1.2.2 The reactor organization prepares and issues operating procedures, subject to review by the Laboratory Safeguards Group. Authority for the establishment of operating procedures is vested in the Manager - VSR Operation.

1.2.3 All experiments will be operated by the reactor organization to maintain close control and assure compliance with standards and procedures. The reactor organization will make a thorough study of all proposals for experiments or for changes in operation, in order to determine that they can be accomplished safely.

1.2.4 The Laboratory Safeguards Group is readily available for consultation and advice on matters relating to nuclear safety and will be referred to freely by the reactor organization as the need arises.

1.3 The observance of process standards will be implemented by:

1.3.1 Close monitoring of reactor and experimental conditions and variables by both operator and experimenter observation, as well as by the automatic alarm and recording systems.

1.3.2 Indication in the reactor control room of critical experiment variables.

1.3.3 Automatic reactor shutdowns when conditions exceed process limits or become such that safety of personnel, the reactor, or an experiment is questionable.

1.3.4 Careful training of personnel in the necessity for strict observance of process standards.

1.3.5 Regular auditing of performance by the Nuclear Safety organization and management.

1.4 The subject matter covered by process standards consists of process controls and safety limitations imposed upon the operation of the reactor facilities and experiments. These standards take the form of fixed numerical values within which process variables must be controlled as well as specifying the modes of operation where such are important to the safety of the reactor, experiments, or personnel.

2. Maintenance

2.1 All maintenance or repair to the internal parts of the reactor will be under the direct guidance of qualified supervision. The shutdown and preparation of the pressure vessel for maintenance will be accomplished in the same manner as for refueling. Every precaution will be taken to prevent unnecessary exposure to radiation during the conduct of work. Radiation exposure will be meticulously controlled by the proper use of portable and personnel monitoring devices under the supervision of the Health Physics group.

2.2 Following maintenance work on critical reactor components, controls, and instrumentation, functional tests will be performed to assure proper and reliable operation prior to restarting the reactor. It is expected that the replacement or removal of major components within the reactor enclosure will be accomplished through the equipment air lock, using the rotary crane without disturbing the integrity of the enclosure vessel. Should this not be feasible, access for equipment will be provided through a temporary opening provided in the vessel wall. The opening would be closed and the integrity of such closure would then be re-established and tested prior to reactor operation.

3. Security

3.1 Access to the reactor is limited to operations personnel properly cleared by prior security investigation and to visitors under security escort. Access to the Laboratory site is under the control of Laboratory guards. Access to the reactor area will be restricted by approved security fencing and controlled gates.

3.3 Before control rod withdrawal is scheduled to start, the security officer will check to ascertain that all personnel are out of the containment vessel except those essential to the reactor startup. During normal reactor operation, access to the containment vessel will be controlled. All physical operation of experiments, including minor adjustments, will be limited to operators directly under the supervision and control of the reactor operating organization. Critical reactor and experiment controls will be kept locked except when they are under the surveillance of an operator.

3.4 The security officer will require all persons entering designated radiation areas to wear radiation monitoring badges. He will arrange for surveys to be made by qualified personnel of areas where radiation levels or conditions may have changed prior to admitting other people. He will regulate the activities of any personnel doing maintenance or construction work in the reactor area so as to prevent any unauthorized activity which could compromise the integrity of the reactor safeguards.

3.5 New fuel elements will be kept in a locked shielded vault, where fuel storage racks physically separate the elements so as to prevent any possibility of a critical assembly. Each operating supervisor will be responsible for all fuel and fuel inventory on his shift. One member of the organization will have overall responsibility for fuel inventories and for making all changes to the reactor core. All fuel loadings will be recorded on a loading sheet and verified that they are exactly in accordance with the approved loading plans.

4. <u>Radiation Protection Policies and Standards</u>

4.1 Planned radiation exposure of personnel will be maintained at a minimum consistent with practical operation of the reactor. Standards are established at the Laboratory which are consistent with applicable Federal and State laws, N. B. S. Handbooks, and recommendations of the International Commission on Radiological Protection. Generally, substantial safety factors are applied to the above limits.

4.2 Environmental monitoring stations, surrounding the Laboratory, provide a continuous indication of radiation levels and variations in concentration of particulate airborne radioactive materials. Water which is to be discharged from the site is retained and analyzed for radioactivity content prior to discharge. Surface and ground water, soil, and vegetation samples from the laboratory site and vicinity are analyzed to determine any deviation from background radiation levels. A small meteorological station supplements the environmental monitoring program.

4.3 Laboratories are established for bioassay, radiochemistry, and counting work to support the health physics program. Specially designed survey meters and radiation detection devices for alpha, beta, gamma and neutron radiation are available for use in the standard calibration of health physics instruments. An emergency vehicle is equipped to provide limited laboratory services under field conditions.

4.4 Thorough records are maintained of all analytical, monitoring, and personnel exposures data. Educational programs are conducted for orientation of employees and the training of personnel to perform radiation monitoring. The Health Physics organization provides consulation, advice, and auditing by professional Health Physicists for each operation within the Laboratory.

4.5 A survey program will be effected at the VSR facility which will include routine periodic monitoring, as well as special surveys which will be conducted whenever unusual conditions are present. Surveys will be made by direct instrument readings and checking of smears.

4.6 Continuous air monitoring will be performed for determination of particulate radioactive material, and when necessary, equipment will be available for radio-gas monitoring. The containment vessel air exhaust will be continuously monitored for both radioactive gases and particulate matter. Continuous radiation monitors will be installed to alert personnel and for supervision of high radiation levels.

4.7 Official written standards and procedures have been established for control of radiation hazards and exposures. The performance of operating components is audited by the Health Physics group.

4.8 Contamination and radiation control activities in the reactor area will follow the instructions, standards, and procedures. In order to minimize contamination and radiation exposures, it is anticipated that the area will be divided initially into four zones of control. Zone 1 would include the reactor equipment rooms where access is limited to reactor shutdowns. Zone 2 would include the remainder of the containment vessel. Zone 3 would include auxiliary equipment areas outside the containment vessel, which have potential for contamination. Zone 4 would include the control room and office building, and the remaining portion of the grounds. Admission to zones 1, 2, and 3 will normally be limited to personnel whose duties require access to these locations. There will be special instructions and restrictions for each zone of control consistent with the hazard presented.

5. Incidents

5.1 Accidents or incidents causing injury to personnel or significant damage to equipment as well as incidents where there was potential for serious personnel injury or equipment damage will be formally investigated. The objective of these investigations will be to determine the cause and to recommend action to be taken to eliminate the cause or reduce its effects should it recur. Investigations will be made by a technically qualified group appointed by management.

5.2 In the event that unexplained occurrences beyond the experience of operating personnel take place, the reactor will be shut down and the situation carefully evaluated by experienced supervisory and technical consultants as required. The inability to go to critical within a reasonable percentage of predicted control rod position is an example of such an occurrence. Because of the experimental nature of the plant, phenomena may possibly be encountered which are poorly understood, and this possibility will receive careful consideration in the planning of all reactor and experiment operations.

E. PRE-OPERATIONAL TESTS

1. General

1.1 A program of pre-operational testing of equipment and systems will be carried out. It is the purpose of this program to demonstrate that the plant has been built according to specifications and that it is ready for fuel loading and initial start-up. This program is intended to cover all tests during the construction period and prior to plant operation.

1.2 The remainder of this section is a summary of the pre-operational test program. This information is presented to indicate the nature and scope of the program. Details may be modified as final planning and execution progress.

2. Power Plant Auxiliaries

2.1 Instrumentation and controls will be checked and exercised for proper installation of primary elements, transmitters, interconnecting piping and special devices such as reservoirs, orifices, and relays.

2.2 The condensate demineralizer equipment will be operated in normal modes to demonstrate proper functioning and accessibility of equipment, valves, and instrumentation.

2.3 The condensate pumps will be checked for operation in accordance with the manufacturer's instructions, for proper miscellaneous connections, and for proper control.

2.4 The boiler feedwater pumps will be operated with discharge valves closed and flow recirculated through low-load bypasses to the condenser hot-well. The pumps will draw suction through the strainers provided.

2.5 Switchgear, relays, devices, interlocks, and scram circuits will be tested using simulated signals.

2.6 All vents, drains, controllers, feedwater piping and instruments on the desuperheaters will be tested to insure proper installation and operation.

2.7 Pre-operational testing of the gas-fired boiler will be performed and guaranteed by the boiler vendor as part of the acceptance tests, to prove compliance with all regulatory and insurance agencies and performance guarantees. This includes hydrostatic testing, boiler washing, leak tightness, and safety valve testing.

2.8 The cooling tower basin, condenser, and circulating water piping will be filled from the raw water storage tank. The circulating system will be operated and tested for running current, leakage, vibration, and flow rate. The cooling tower fans will be checked for rotation, vibration, starting and running current. The dampers and louvers will be examined for operability, air leakage and air flow.

2.9 With the cooling water circulating pumps in operation, the water boxes, valves and joints of the dump condenser will be tested for leakage. A system vacuum will be held to insure acceptable air in-leakage. The air ejector will also be tested.

3. Reactor Auxiliaries

3.1 Each primary element, transmitter, receiver, indicator, control element and other instrumentation device will be checked, as well as all inter-connecting piping and wiring. Adjustments and calibrations of each device in accordance with manufacturer's recommendations will be made. Operating conditions will be simulated for each device to the fullest extent possible. Temperature devices using thermocouples for the primary element are checked by applying a known d-c voltage to the system to simulate output of the thermocouples. Limit switches for indicating lights, alarms, and safety system operation will be checked by actual or simulated operational conditions. All control devices will be exercised to assure proper operation with the accuracy and response characteristics required by the system and specified by the manufacturer. Set points for all devices will be checked and adjusted as necessary.

3.2 Each circuit of the reactor safety system is tested to see that a scram signal can be initiated through its particular channel. An appropriate test signal is simulated mechanically to test the vacuum sensors and a simulated scram signal is supplied to each of the neutron flux sensor amplifiers to check the scram function of these devices. By combinations of sensor testing, all safety system coincidence features are thoroughly checked out before initial loading. The fail-safe feature of this system is also checked for proper action. In such a manner, a reliable, properly adjusted safety system is assured for subsequent protection of the reactor plant.

3.3 The emergency steam dump tank of the emergency cooling system will be tested by having the reactor vessel at normal coolant operating level, filling it with steam at operating pressure from the gas-fired boiler, and checking steam flow, pressure gradients, etc, upon operation of the quick-opening valves to the dump tank. The valves, makeup feed system and level controller of the emergency cooling heat exchanger will be examined for proper operation and system tightness.

3.4 Each control rod drive will be operated over its full travel, observing operation of the carriage and lead screw for vibration, rubbing and run-out. Motor speed, motor currents, position indication, and position of limit switches will be checked also. Scram testing of all rod pistons will be carried out. Instrumentation is provided to check the scram time for each rod. Tests for any vibration or indication of any interaction during group scramming will be made. With the vessel full of water and the head off, each rod will again be run over its full travel to check for binding due to poison section or fuel channel misalignment. Tests will also be repeated with the vessel up to temperature but with no fuel present.

3.5 The liquid poison system will be filled with water and checked for operation of pumps, control valves, relief valves, check valves, level switches, and alarms. When the system operates satisfactorily, it will be filled with the liquid poison material. The liquid poison system will be loaded and operable for initial fuel loading.

3.6 The water flow through the shield cooling coils will be checked to assure proper operation.

3.7 The fuel pool will be filled with water and checked for leakage. Dummy fuel elements will be run through a complete cycle from the new fuel storage vault into the reactor core to train the fuel handling crews. This procedure is repeated as necessary to assure the capability of the fuel handling crew in all phases of the work, including vessel head removal, operation of the fuel handling tools, and use of the orifice changing equipment.

3.8 Reactor cleanup demineralizer pumps will be tested for noise, total differential head, leakage and cooling; the nonregenerative heat exchanger for leakage, pressure drop, cooling water flow, and control valve operation. All piping and valves in the system will be tested for leakage and the demineralizers for flow, pressure drop, and water purification effectiveness. The resin sluice system will be operated for satisfactory action in emptying and recharging of demineralizers.

3.9 The primary system steam relief valves will be inspected visually. They will have been set previously, prior to installation on the reactor system.

3.10 To insure a properly clean reactor system, every effort and control will be used during the fabrication, shipment, and installation to maintain a high degree of cleanliness. Cleaning procedures prior to initial operation will be established to obtain appropriate cleanliness.

4. <u>Service Equipment</u>

4.1 All auxiliary systems not covered in the foregoing will be operated and checked to assure satisfactory condition and compliance with procurement and erection specifications, and drawings with respect to performance, cleanliness, accessibility of valves, controls, instruments, and specialties. These systems include the following:

Ventilation System
Cooling Water
Waste Disposal
Service Water
Make-up Water
Instrument Air
Service Air
Containment Vessel Access Door Interlocking System
Containment Isolation Valve Closing and Interlocking

F. INITIAL CORE LOADING AND CRITICAL TESTING

1. General

1.1 Accurate knowledge of reactor characteristics is required for safe operation of the plant. Consequently, an extensive program of tests and measurements is planned for execution during and immediately following the initial fuel loading. The loading and critical testing program will begin when the special initial loading instrumentation and the necessary reactor equipment have been thoroughly checked and found to be in a safe and operable condition.

2. Basic Test Conditions

2.1 At the start of loading, the reactor vessel will be filled with water to a minimum of 3 feet above the active portion of the core. The normal equipment and techniques are, in general, used for the installation of fuel. Minor modifications are made as required; for example:

2.1.1 The water level may be below that required for normal operation since the fuel has not been irradiated to any appreciable extent.

2.1.2 The fuel may be moved and stored without the fuel cask, since the fuel is new.

2.2 A neutron source of sufficient strength to provide adequate readings at all times on the neutron sensors will be inserted before fuel loading begins. Neutron sensor - source geometric relationships are maintained such that the sensors measure neutron multiplication by the fuel.

2.3 During fuel loading, the control rod scram circuit will be operated by at least two neutron sensitive channels in non-coincidence scram logic. This instrumentation as well as the scram circuits and control rods are checked at frequent intervals. Each control rod is checked for proper functioning and for neutron absorption as it is encompassed by the fuel loading.

3. Core Loading

3.1 The first stage of core loading will be to a "just critical" water reflected array. Using criticality calculations as a guide, fuel will be installed in a selected area within the control rod pattern. Loading will be accomplished in a stepwise manner with neutron multiplication measurements between steps. The magnitude of each fuel addition will not exceed two-thirds of the estimated amount required to attain criticality, with the minimum addition being one fuel bundle.

3.2 When testing is complete on the minimum critical array, fuel loading will again proceed in a stepwise manner, with the reactivity effect of each added increment determined to be less than one-third of the shutdown control margin. The maximum core size will be that determined to yield the desired excess reactivity, if this is less than the 32 bundle loading.

4. **Critical Test Program**

4.1 The physics measurements planned during the critical test phase of the program are described in some detail in Section II. C. 8 of this report. Briefly, these measurements include:

4.1.1 The calibration of control rods and measurement of core excess reactivity.

4.1.2 The evaluation of fuel bundle worths.

4.1.3 The measurement of void and temperature coefficients of reactivity.

4.1.4 The mapping of neutron flux and power distributions within the core at very low power.

4.1.5 The evaluation of flooding and unflooding reactivity effects as a function of temperature and control rod pattern.

4.2 Procedures for accomplishing the above measurements will be such that reactivity transients will not occur during critical operation. For example, the determination of the flooding reactivity effect at an elevated temperature would take place as follows:

4.2.1 The unflooded core would be pre-heated to the desired temperature with all rods inserted (sub-critical) using an external heat source (non-nuclear).

4.2.2 The reactor would then be made critical by withdrawal of control rods. The rod positions at critical would be noted and, if necessary, the rods would be re-calibrated at this temperature to determine the excess reactivity of the assembly.

4.2.3 The reactor would then be made sub-critical by insertion of control rods; the process tubes would be flooded by raising the water level, and pre-heating would be resumed to achieve the desired temperature again.

4.2.4 When the desired temperature is reached, the reactor would be brought to critical with the same control rod pattern as before. The rods would be re-calibrated, if necessary, positions noted, and the excess reactivity once again determined. The difference in excess reactivities between the two runs represents the flooding reactivity effect at that temperature and for that rod pattern.

4.3 By procedures such as the above, it should be possible to safely measure the pertinent reactivity effects and coefficients without incurring unnecessary reactivity transients at any time when the reactor is critical.

G. STARTUP AND POWER OPERATION TESTS

1. General

1.1 Following the initial fuel loading, but prior to nuclear power production startup, it will be necessary to complete the pre-operational test program of items not required for the loading or critical testing operations. An example of such a test is the steam flow distribution test, discussed below. Only when all pre-operational tests are completed, the initial critical tests completed and the data found satisfactory with respect to reactor safety, will the startup and power operation test program be undertaken.

2. Flow Distribution Tests

2.1 After the fuel has been loaded in the reactor, it will be possible to study the steam flow distribution through the core from the steam outlet headering and control system. The 12 steam flow bias control valves will be set at their predetermined settings for the flow distribution desired. Saturated steam will be admitted to the VSR, and flow will be established at several successive levels up to rated flow. Instrumentation will be checked, and vibrations and thermal expansions will be measured at each level. The flow indication for each of the bundles or groups of bundles will be checked against the calculated values and adjustments made where required. This will calibrate the outlet steam valves with saturated steam.

2.2 With the steam flow bias control valves set to represent a normal operating situation, the emergency cooling system will be actuated together with a containment building isolation trip. The flow decay with time will be noted and checked against the minimum flow requirements through each fuel bundle or groups of fuel bundles. This will be done with the reactor vessel up to pressure and the moderator at saturated temperature.

3. Initial Nuclear Startup

3.1 The plant will be readied for power operation startup as indicated in the cold startup procedures in Section III.H. The reactor will be brought up to initial power in small steps, the first step at about 5% power and 25% steam flow. At this lower power rating, the fuel clad temperatures will be very low. If, for some reason, flow were to short circuit one of the fuel bundles, calculations indicate that fuel melting would not result. Hence, operation at this power level will permit extensive checkout and calculation checks without endangering the safety of the system. Any significant discrepancies between calculated and measured parameters will be investigated before increasing power further.

3.2 As much as 1/8 of the core, but no less than one fuel bundle assembly will be fully instrumented with thermocouples. This will permit measuring the outlet temperature from each of the nine fuel elements of the bundle at low power and correlating these to the calculated values. This will serve as a check on the accuracy of the overall power distribution and internal orificing calculations.

3.3 While at about 5% rated power, nuclear instrument calibrations will be checked, wire irradiations made to check axial power distribution at 5 core-length positions, and radiation surveys conducted to check the adequacy of reactor shielding.

4. Power Operation Testing

4.1 When the first power step (5% power) has been satisfactorily completed, flow and power will be increased in increments up to the design rated value for the core loading, taking due consideration of all operating limits as well as data taken at each power step. A minimum of 5 steps, none of which will exceed about 3 thermal megawatts, will be taken to reach rated power. Meaningful tests will be conducted after each power increase to ascertain the safety of proceeding to the next higher power level. Superheat steam temperatures and flows, and the relationship to fuel cladding temperature, will be of particular interest during these tests. Steam activity levels will be carefully observed. Other pertinent measurements to be made and noted at appropriate intervals include reactor vessel temperatures and the associated thermal stresses, temperature in the reactor shielding concrete, radiation levels throughout the plant, power distribution measurements from wire irradiations, and transient stability tests. This period of testing will serve also as a checkout of the auxiliary equipment and reactor controls in general. For example, the proper functioning of the desuperheaters will be demonstrated at this time.

4.2 At an appropriate time during the power testing program, an emergency cooling test will be conducted. This will be done at cladding temperatures well below the present design limit of 1250°F. Individual bundle steam outlet temperatures will be noted as well as steam flows from bundles or groups of bundles.

4.3 Before proceeding above about 50% of rated core power, it is planned to run at a steady power for sufficient time to impress a power distribution on the fuel, and then to shut down and gamma scan at least part of the core to check the power distribution calculations. The power distribution will be correlated with fuel bundle steam outlet temperatures from fully instrumented fuel bundles to verify the ability to predict peak cladding temperatures from outlet steam temperatures.

4.4 Design power levels and cladding temperatures will be approached cautiously. Consideration will be given to the desirability of approaching the maximum cladding temperatures by further throttling of steam flow in a single bundle as well as by power increase. This would permit reaching the maximum claddding temperature at a lower reactor power as well as limiting the fraction of core involved in the initial high temperature testing.

4.5 Substantial experience with operation of the reactor within the initial design limits would be gained prior to undertaking further increases in power or peak cladding temperatures.

H. NORMAL OPERATIONS PROCEDURES

1. General

1.1 The general procedures for a few selected operations which are peculiar to the VSR are outlined below in this section. The reactor startup and shutdown procedures are included. Procedures for other routine operations such as refueling, waste system operation, and maintenance procedures are not included at this time. These latter types of procedures are, in general, very similar to those used at the VBWR and other facilities, they are not peculiar to a superheat reactor, and the general approach to such operations has been indicated in the equipment descriptions in Part II of this report as well as by the basic operating principles and administrative procedures presented earlier in Part III of this report.

1.2 The generalized procedures which are presented below are intended to indicate the nature and scope of the required operations. These may be modified as the final planning and execution progress. The detailed procedures will be developed by the operating staff prior to initial operation. The procedures used for initial startup will incorporate testing experience prior to such operation. Procedures may then be modified further depending upon the initial operating experience.

2. Cold Startup Procedure

2.1 A cold startup after a period of operation or refueling differs from the first cold startup made in connection with power-operation testing in that the excess reactivity will have changed and the startup critical position is affected accordingly; it will be necessary to proceed on the basis of predicted criticality in such a case. The reactivity of the core will be confirmed by a criticality check before resuming power operation. Care is taken in withdrawal of the control rods to approach criticality in a conservative manner.

2.2 Before actual reactor startup, a complete detailed check list is completed to assure that all equipment is in proper condition for operation, that safety circuits and interlocks are operable, that all valves, control settings, reactor water level, vents and drains are set properly.

2.3 Prior to reactor startup, the steam source, the VSR condenser, VSR cooling tower, and the associated plant equipment will be checked and placed in operation. The steam source will supply a percentage of rated steam flow to the VSR condenser through the steam by-pass line. The "steam source" could be either VBWR or the gas-fired boiler. Insofar as VSR startup is concerned, the procedures are basically the same whichever steam source is used.

2.4 Steam will be slowly bled into the VSR moderator. The moderator will be heated at a predetermined rate based on detailed pressure vessel stress analysis. Critical vessel temperatures will be monitored. The reactor drain will be controlled to maintain water level as steam is added. The moderator pressure is raised to a predetermined value, which may be near the full-rated operating pressure, before any attempt is made to blow the water out of the flooded superheat steam passages of the core.

2.5 Having achieved the required pressure, the preheating steam valve will be closed and the main steam inlet valve to the reactor set to maintain that pressure. Next, the individual steam flow bias control valves will be completely closed and the isolation valve downstream of these valves will be opened while the main pressure reducing valve to the condenser is set well below the reactor pressure.

2.6 The first fuel bundle or bundles will then be blown free of water by first cracking and then slowly opening the bias valve. The latest tests conducted under the AEC Superheat Program indicate that the core can be blown dry in this manner, substantiating theoretical calculations. To assure that water will not be trapped in low velocity regions above the core, further tests on a prototype assembly (complete with risers and downcomers) will be conducted to confirm the required steam flow rate and flow time needed. The first bias valve will be closed and the process repeated for the other eleven valves. The control rod system will not be energized during blow out of the bundles.

2.7 Having blown the core free of water, the reactor is now prepared for criticality. All bias valves are adjusted to their previously calibrated positions for the desired mode of operation. The main outlet pressure regulating valve is set to give about 25% flow. All flow nozzles are checked to assure proper flow distribution.

2.8 At this point, the entire process system is again checked for proper operation. The process control system can now be tied into the safety system. The reactor is brought up to criticality by control rod withdrawal, and the critical rod positions checked. Power is then slowly raised to about 5%. All outlet temperatures are monitored to insure that the core is indeed dry and to detect if any leakage into the steam circuit is present. The reactor is maintained at this condition for a sufficient length of time to ascertain that no difficulties are present. This low power level is sufficiently high to obtain meaningful readings of steam outlet temperatures and yet low enough so that fuel meltdown would not occur even if core flow were suddenly lost.

2.9 To take on full load, the flow is slowly raised to 100% of rated flow which will cause the steam outlet temperatures to drop to nearly saturated steam conditions. All the flows are checked, and, based on the previous calibrations, the bias valves are adjusted to give the proper flow distribution for saturated flow. The power is raised from 5% rated to some fraction of rated power at a predetermined rate. The total steam flow will drop due to the increase in outlet temperature, and the pressure regulating valve is readjusted to give rated total flow. The power is again increased another step and the flow readjusted. Thus the full power condition is reached in a series of steps. Upon reaching full power, the bias valves may be adjusted to give the precise conditions desired in the individual fuel bundles. The number of steps required to reach rated power is being established from part load calculations of the core, effect on flow distribution, and other pertinent factors.

3. Hot Startup Procedure

3.1 A hot startup is used following scram should the cause of the scram be discovered and corrected quickly and it is desired to get the plant back on the line as soon as possible. The procedure is essentially independent of the cause of shutdown, assuming that the cause of the shutdown is recognized and any non-standard condition corrected. An appropriate check list for such a situation will be provided for the operators to complete prior to withdrawal of control rods.

3.2 The reactor would then be started up beginning with 25% flow, and repeating the procedures described in paragraphs 2.7 through 2.9 of the "cold startup procedure".

4. Full Power Operation

4.1 Normal power operation of the plant, including adjustment to limited load changes, is controlled manually.

4.2 The principal functions of the operating personnel during normal power operations are as follows:

4.2.1 Manipulation of control rods to accommodate major changes in load demand and also adjustment of control rods and steam flow bias control valves to secure the desired operating conditions.

4.2.2 Surveillance of plant and experimental equipment for proper functioning and performing of necessary adjustments and repairs. These functions include routine preventative and overhaul maintenance; process-steam sampling; observing and recording of information provided by plant instrumentation; routine tests of control rod functioning; and periodic tests of critical station valves.

4.2.3 Receipt and preparation of new fuel for refueling and the preparation of irradiated fuel for off-plant shipment.

4.2.4 Evaluation of any abnormal conditions and emergency action as required to minimize the effects of equipment difficulties.

4.2.5 Operation of the waste disposal system.

5. Normal Reactor Shutdown

5.1 A normal reactor shutdown of short duration can be accomplished while maintaining essentially rated reactor pressure. Power is reduced slowly, and the saturated steam will continue to flow through the fuel process tubes for the period of time that the reactor is shutdown.

5.2 Normal plant shutdowns of long duration such as are required for refueling, require operations in addition to those for shutdowns of short duration. After control rods have been inserted, system pressure is reduced by progressive readjustment of the pressure regulator setting in order to insure controlled cooling of the fuel within a minimum time. The determining factor in system pressure reduction is the allowable cooling rate for the pressure vessel and associated equipment. The saturated steam flow is continued to the reactor, with the steam bled to the condenser at a controlled rate, until such time as the core can be flooded. Flooding will not take place until the fuel clad temperatures are sufficiently lowered that severe thermal stresses in the clad are avoided upon flooding of the process tubes. Flooding of the core is accomplished by slowly raising the moderator level after the system pressure has dropped below 500 psia. After flooding, the continued removal of decay heat from the core will be accomplished by heat transfer to the moderator and circulation of the moderator water through the shut-down cooling heat exchanger.

6. <u>Emergency Reactor Shutdown</u>

6.1 In the event of an emergency shutdown which involves a reactor scram accompanied by a containment isolation, the problem becomes one of insuring that sufficient saturated steam is permitted to flow through the process tubes for some period of time. This is accomplished by the emergency steam cooling and superheat steam dump system, wherein moderator water inside the reactor vessel is flashed and the steam thus generated flows through process tubes into this system. The emergency cooling system operation is described in more detail in Section II. F. 1.

J. EXPERIMENTAL OPERATION

1. General

1.1 The VSR will be built as a developmental tool principally to be employed as a test bed for superheat reactor fuel. Consistent with this objective, the reactor is designed to be able to accommodate the irradiation of large numbers of fuel samples for "screening" purposes, as well as fuel for long burn-up and other tests, under a variety of temperature, power and flow conditions. Design features provided to promote the usefulness of the reactor as a fuel test facility include the individual fuel bundle flow orificing provisions, outlet subheaders with 12 flow control bias valves, individual bundle steam outlet temperature measurements, flow measurements on individual bundles or small groups of bundles, a "divert header" for outlet steam flow from four bundles to the VSR dump condenser (especially for testing defected fuel), provisions for in-core irradiation of wires during reactor operation, and many other design and instrumentation features guided by the basic objective of the reactor.

1.2 This report does not contain a specific proposal to the Commission for using the reactor to investigate and develop a diversity of superheat fuels. As indicated in Section II. B. 3, the safeguard analyses in the present report are limited to consideration of operation of a Mark I core up to "design" conditions. When the VSR development program and supporting safety evaluations are sufficiently developed, it is anticipated that these will be submitted to the Commission for consideration in issuance of the operating license for the reactor. Depending upon the results of analyses and preliminary information available by the time of VSR startup, the program could include the addition of specific developmental fuel bundles to the startup core as well as indicating the longer-range program for fuel testing. With respect to the latter item, it is anticipated that the VSR will be used to investigate and develop superheat fuel much in the same way that VBWR has been used to develop boiling water reactor fuel. This implies a program based on step-wise test procedures for introducing new fuel types into the reactor, for bringing them gradually up to more and more severe operating conditions, for introducing additional similar fuel bundles, etc.

1.3 The remainder of this section is devoted to a general discussion of the kinds of fuel development which are currently considered will be undertaken in the VSR. The intent is to provide a preliminary indication of the scope of the development program for the VSR.

2. Superheat Fuel Cladding

2.1 The investigation of cladding temperature limits will be of prime interest for any cladding material investigated. The clad temperature limits have a direct bearing on the maximum degree of superheat which is attainable and therefore have a direct bearing on the economy of a large superheat facility.

2.2 Cladding materials of interest for use in superheat reactor service include the 300 series stainless steels, Inconel and Nichrome. The possibility of low cycle fatigue failure of these materials is of interest because of the plastic strain (plastic deformation) which occurs as the fuel material expands and contracts when the reactor is started up or shutdown or the power output changed. Because of the nature of operation of a test facility such as the VSR, a relatively larger number of cycles of deformation are produced in a given period of time than in a typical power reactor. Thus, the VSR is well suited to testing clad stability with respect to low cycle fatigue failure.

2.3 In a nuclear superheat reactor the clad thickness must be minimized in order to attain favorable neutron economy. This introduces the plastic deformation discussed in the paragraph above since the cladding is not "self-supporting" at operating conditions of temperature and pressure. The thin cladding emphasizes also the importance of corrosion effects. The effects of the steam environment, as well as radiation effects, on corrosion rate are of interest.

2.4 The erosion effects of high velocity steam will be of interest not only from the effects of erosion on corrosion resistance, but also due to the carryover of radioactive surface scale which may be deposited in the turbine or other parts of the system.

2.5 Other areas of investigation for long-term irradiation of cladding materials include the effects of normal applied stresses and the possibility of cladding-fuel interactions due to escape of fission fragments into the cladding material.

3. Properties of Fuel Material

3.1 Fission gas release from the fuel to the space within the cladding, as well as release from defective fuel cladding, is of prime interest in the design of superheat reactor fuel and reactor systems. The effects of fuel material chemical and physical properties, as well as exposure level, will be of interest in investigating fission gas release.

3.2 Tests with purposely defected cladding will permit evaluation of system operation, in general, as well as investigation of possible effects of permitting excess oxygen to enter the defect and affect the UO_2 fuel material.

4. Process Tube Development

4.1 The process tubes are required to separate the water moderator from the superheat fuel. Process tubes with the required stability under irradiation and cycling operation, resistance to corrosion and erosion, and strength or support against collapse are required. To meet these requirements with a design which minimizes neutron absorption and heat loss to the moderator will require substantial development and testing in the VSR.

5. Fuel Geometry

5.1 The VSR could be used to test single-pass as well as double-pass fuel element geometries. Other important aspects of the fuel geometry include fuel support, vibration, wear and fretting corrosion.

K. EMERGENCY PROCEDURES

1. The reactor and its instrumentaion and controls have been designed to shut the reactor down and take other required immediate actions quickly and automatically, without the need for operator action or the following of prescribed procedures. Once the reactor has been safely shut down, whether by the automatic devices or by the operator's response to alarms or other indications, the operators will assess the cause for emergency shutdown and instigate whatever additional actions may be required by the particular situation. Emergency procedures will be prepared in advance for the operator to follow. In some cases the emergency procedures will provide actions to take in order to avoid an emergency shutdown when an undesirable condition begins to develop.

2. The objective of the emergency procedures in all cases is to minimize the hazards to personnel and to restore the plant facilities to a safe condition as quickly as possible. Emergency procedures are an important part of each plant operating procedure, and will be developed in detail by the operating organization when the detailed operating procedures are prepared prior to initial plant startup.

L. DISASTER AND EVACUATION PLANS

1. Disasters may be placed in the following general categories:

   (1) Fires
   (2) Explosions
   (3) Mechanical or operational failures resulting in severe hazards to personnel.
   (4) Enemy air attack
   (5) Earthquakes

   Any of these disasters could involve significant radiation problems. The primary aim is to minimize injury to personnel. The General Disaster Plan for the Vallecitos Atomic Laboratory is described below. Drills are conducted at least annually.

2. Upon first indication of an incident involving the release of radioactive material, all personnel will be alerted. Upon a signal to evacuate, all operations will be stopped and ventilation shut down. Action will be taken to make the facility safe. All personnel will go to a prearranged area within each building and an emergency team will go into action. A messenger will be dispatched with information for all persons in remote areas of the site to cover those situations where the alarms may not be heard. The emergency team will simultaneously determine whether or not anyone had been exposed to the extent of medical concern, the extent and magnitude of radioactivity on the site and in the vicinity of the site, and meteorological conditions. A preliminary evaluation will be made to determine the magnitude of the incident and to scope the necessary operations. If necessary, the Laboratory Manager or his alternate will notify civil authorities. If there had been serious exposure to personnel, the exposed or injured individuals will be evacuated to medical facilities, after having established that it is safe to leave the buildings. All other personnel will then be evacuated. An emergency team will continue to monitor and decontaminate until the situation is under complete control. A committee will conduct a complete investigation, evaluation, and report of the incident.

3. If the accidental criticality monitoring system is triggered, the emergency warning system will sound continuously in a pulsed manner and all personnel in the alarmed area will evacuate by the nearest exit, taking with them any portable radiation monitoring instruments found in the evacuation path. Initial evacuation of personnel will be upwind a distance of approximately 1,000 feet from the affected area. Re-entry to the area will be accomplished under controlled conditions.

4. The primary alarm will consist of horns and bells in each of the principal areas. A coded signal is used which denotes the magnitude and locations of the incident. In the event of accidental criticality, the emergency alarm system in the other areas remain unaffected and additional coded alarms may be given in the normal manner in these other locations.

5. If an air raid occurs, the information will be transmitted by telephone. Additional communication between buildings will be provided by the telephone system. It has been assumed that any disaster will disable commercially supplied power but that the battery-operated telephone system will continue to function. In the event that the telephone system is also inoperable, additional information will be transferred by messenger.

6. Assembly areas for the principal buildings are as follows:

   Building 102 - near the reception desk
   Building 105 - in the entrance hall
   Boiling Water Reactor and Superheat Reactor - lunch room and control room
   General Electric Test Reactor - control room
   Maintenance and Storage Building - receiving area
   Building 103 - Main entrance area

7. Portable radiation measuring instruments which have been properly calibrated and are in operating condition will be located throughout the facility at strategic points. Self-contained breathing apparatus plus approved high efficiency filter type respiratory protective devices, as well as suitable quantities of protective clothing are also available at each facility. Stretchers, first aid kits, and personnel suitably trained in first aid and rescue work are available.

8. The health physics truck is equipped with a supply of protective clothing and respiratory protection, portable instrumentation, some laboratory type instrumentation for water and air sample analysis, and a portable air sampler.

9. The highest ranking person at the scene of the incident will be responsible for the safety of all personnel in the building, for making an initial evaluation of the magnitude of the incident, and for taking any emergency action he deems necessary until such time as he is relieved by a responsible member of management. He will be responsible for informing management immediately that an incident has taken place, and giving his estimate as to the seriousness of the situation and type of incident that has occurred. He will direct the action of the emergency crews in his building with the aid of a qualified health physicist. In the event the building must be evacuated, any individual seeing a portable radiation monitoring instrument should take it with him.

10. The Manager of the Laboratory, or his designated alternate, will act as site disaster director. It will be his responsibility to notify civil authorities if necessary, and to provide them with specific information on evacuation routes, private homes or offsite facilities to be evacuated, roads which must be blocked, and other data pertinent to the particular situation. It will be the responsibility of Radiation Safety to determine the magnitude of radiation hazards, to direct the action of the emergency team, to obtain meteorological data, and to advise the disaster director on evacuation and all other pertinent matters. If the disaster is a fire involving radiation problems, Radiation Safety personnel will meet offsite firemen at the main gate, will escort them to the site of the fire, and will advise the fire chief of the radiation hazards involved.

11. Exposure of rescue personnel and damage control personnel will, wherever possible, be limited to normal weekly exposure limits or to limited additional exposure after obtaining the approvals which are normally required, with the following exception:
In extreme emergency involving human life or material loss of great magnitude, a planned emergency exposure not in excess of 25 rem, whole body exposure, and 100 rem additional to the hands and forearms, and feet and ankles may be taken. (Note: These limits are in keeping with the recommendations of the National Committee on Radiation Protection, National Bureau of Standards Handbook 59, September 24, 1954). It is desirable to give several people small exposures rather than one person the full 25 rem exposure. Where practicable, exposure will be limited to half of these emergency values. Whenever possible, such exposures will not be effected without prior approval of the Laboratory Manager. Under no circumstances will it be planned to expose an individual more than once to a dose of this magnitude.

12. If it is necessary to evacuate the site, Radiation Safety will advise the disaster director on the route based on weather information and the guard at the gate will be informed. In the event that weather data from the site stations are not available, the principle evacuation route will be in a general up-wind direction. Transportation will be primarily by personal cars. Each car will be loaded to maximum capacity and will leave when loaded. All persons other than the driver will wait inside the building until a car is at the loading location. No one should remain outside unless absolutely necessary. Visitors will be the responsibility of the person they are visiting. Maximum evacuation time from the site is estimated to be 35 minutes.

13. Tentative initial destination of the evacuation caravan will be either near the Alameda County Fairgrounds in Pleasanton, near the Rodeo grounds in Livermore, or near the Fremont City Hall in the Mission San Jose District of Fremont. No one should leave the rendezvous point until Radiation Safety has made personal contamination surveys.

14. Injured persons will not be taken to the rendezvous point but will be taken to the nearest hospital in the direction of the evacuation route, being certain that Radiation Safety is notified of the hospital so that adequate surveys may be performed.

15. The site disaster director will notify the local authorities to evacuate nearby inhabitants, as required by the magnitude of the accident. Maximum evacuation time for the inhabitants near the site boundary is estimated to be 1.5 to 3 hours.

16. Following is a list of hospitals:

Direction.	Name and Adress	Approximate Travel Time
East	Veterans Admin. Hospital Arroyo Road Livermore, California	15 minutes
East	St. Paul's Hospital 813 "J" Street Livermore, California	15 minutes
West	Washington Township Hospital 2000 Mowry Road Fremont, California	20 minutes
Northwest	Eden Hospital 20103 Lake Chabot Road Castro Valley, California	30 minutes
North	Kaiser Foundation Hospital 1425 South Main Street Walnut Creek, California	40 minutes
South	San Jose Hospital 14th & East Santa Clara Sts. San Jose, California	40 minutes

PART IV - SAFETY EVALUATION

A. INTRODUCTION

1. Part IV of this report presents the basic safeguard philosophies guiding the design of the VSR facility and reflected in the specific design features and procedures described in Parts II and III. Part IV presents also discussions of the reactor safety in the event of occurrence of various off-standard conditions, equipment malfunctions, or operator error; a summary of several detailed analyses of specific hypothesized reactor accidents; and the radiological effects of the "maximum credible accident" to the reactor.

B. SAFEGUARD OBJECTIVES

1. The general safeguard objectives, or goals, toward which the safeguard design criteria are aimed, and toward which the design strives, are as follows:

1.1 Normal operation of the plant must not result in the exposure of any persons on or off the plant premises to radiation in excess of established permissible limits.

1.2 Safety against a nuclear accident that might release dangerous amounts of radioactive materials must be preserved even in event of equipment malfunction, operator errors, or other credible contingencies.

1.3 Confinement must be provided for any significant quantity of radioactive materials that might be released from the reactor, including the unlikely event of a reactor system rupture, to prevent radioactive contamination of the environs.

C. GENERAL SAFETY FEATURES

1. Inherent Safety

1.1 The prompt negative Doppler fuel temperature coefficient associated with the UO_2 fuel used in the reactor provides a substantial inherent safety mechanism against nuclear power excursions under any reactor operating condition. The Doppler effect provides also a degree of self-regulation for steady-state operation of the reactor.

1.2 Under normal operating conditions, the boiling water moderator provides the inherent safety of negative temperature and void reactivity coefficients such as are found in current BWR designs. However, these coefficients are less effective against power transients than in BWR's due to the insulating effect of the superheat steam space and process tube between the fuel cladding and the moderator water.

1.3 The reactor will have a negative "unflooding" or "water blow-out" reactivity effect in the hot condition. This provides inherent safety against a hot startup accident prior to unflooding the superheat steam process tubes, and an inherent means of reversing the reactivity addition effects of accidental flooding of the process tubes during power operation.

2. Designed-In Safety

2.1 The plant control systems are designed to minimize the possibility that operator errors or equipment malfunctions could lead to an unsafe operating condition. The control rod system provides for manual withdrawal of single rods at a very slow rate. The steam flow control system provides for automatic control of the total desired flow and manual control of the 12 very slow moving flow control bias valves.

2.2 A reactor safety system is provided to detect potentially unsafe operating conditions and to initiate automatically the appropriate countermeasures to prevent the condition from becoming too severe. The system includes sensing devices for all reasonably conceivable unsafe operating conditions that might arise from operator error or equipment malfunction.

2.3 Two separate and independent neutron poison systems are provided to assure shutdown of the reactor. The normally used system consists of 12 control rods which are capable of fast automatic "scram", simultaneous automatic rod "run-in", or manual insertion. The safety system provides signals to the control rod system for automatic action on a "fail-safe" basis. The second neutron poison system is a manually-actuated slower-acting emergency system which injects a poison-bearing solution directly into the reactor moderator.

2.4 Two systems are provided for emergency removal of reactor heat. The first system is the normal full-power heat removal system, which can be used in many emergency situations. This system consists of inlet steam supply from the VBWR, the gas-fired boiler, or both, and dissipation of exhaust steam flow to the VSR condenser or the VBWR turbine-condenser. The alternate supply and exhaust provisions make this a potentially very reliable system. The second system is the "reactor emergency cooling system", which is provided within the containment vessel for use in the event of isolation of the reactor from either the normal steam inlet or exhaust lines. This system takes its steam supply from the reactor water moderator and discharges to a dump tank and condenser designed to provide adequate emergency cooling flows.

3. Radiological Control

3.1 The gaseous and liquid waste disposal systems are designed to control the release of radioactivity to unrestricted areas to within permissible limits through the use of adequate monitoring of the radioactive content of such wastes and absolute control of their release based on measured results. A good example of this is the constant monitoring of condenser off-gases and automatically-controlled isolation of the off-gas discharge line to the stack if the monitor should indicate a potentially excessive rate of activity release.

3.2 Sources of radiation are so located or shielded to the extent necessary to ensure that personnel exposure can be held within permissible limits in the performance of normal operating tasks in the plant.

4. Containment

4.1 Although the possibility of an accident involving the dispersion of substantial quantities of fission product radioactivity is extremely remote by virtue of the foregoing safety features and the application of strong procedural control, the protection of the health and safety of the public is further assured by housing the reactor and many of its auxiliaries within a high-integrity, essentially vapor-tight containment vessel.

D. SAFEGUARD DESIGN CRITERIA

1. Reactivity Coefficients

1.1 The moderator void coefficient averaged over the interior of a fuel channel (the steam generating area) must be zero or negative whenever the reactor is critical. If reactor pre-heating and interlocks are used to avoid criticality under conditions when the moderator void coefficient may be positive, the magnitude of such positive coefficient must be sufficiently small that significant core damage would not result if criticality were achieved accidentally under conditions normally avoided by the interlocks.

1.2 The moderator temperature coefficient must be zero or negative at full operating temperatures and should be as negative as possible at lower temperatures consistent with good over-all design. If the moderator temperature coefficient may be positive under any conditions of planned criticality, the magnitude of such positive coefficient must be sufficiently small that its effect could not credibly cause fuel damage. If reactor preheating and interlocks are used to avoid criticality under conditions when the moderator temperature coefficient may be positive, the magnitude of such positive coefficient must be sufficiently small that significant core damage would not result if criticality were achieved accidentally under conditions normally avoided by the interlocks.

1.3 The reactivity effect associated with unflooding the superheat steam passages must be zero or negative whenever the reactor is critical. If reactor preheating and interlocks are used to avoid criticality under conditions when the unflooding reactivity effects may be positive, the magnitude of such positive reactivity effect must be sufficiently small that significant core damage would not result if criticality were achieved accidentally under conditions normally avoided by the interlocks.

1.4 The reactivity effect associated with flooding the superheat steam passages must be zero or positive whenever the reactor is critical (by virtue of paragraph 1.3 above). The magnitude of this positive effect must be minimized as much as possible consistent with the requirements in paragraphs 1.1, 1.2 and 1.3, above, to optimize reactor safety with respect to conflicting requirements. Interlocks (or other devices) must be provided to prevent core flooding with the reactor critical, and the magnitude of any positive flooding reactivity effect must be sufficiently small that significant core damage would not result if flooding occurred accidentally under conditions normally avoided by the interlocks.

1.5 The prompt Doppler temperature coefficient of the fuel must be negative.

2. Control Rod System

2.1 The control rod system must provide the assured capability of fast automatic shutdown of the reactor upon loss of the signals from the safety system which indicate that no potentially unsafe operating condition exists.

2.2 The reactivity worth of individual control rods should be minimized to the extent possible, e.g., by using a reasonably large number of control rods.

2.3 The control rod system must provide sufficient shutdown margin so that the reactor can be shutdown and/or remain shutdown with any control rod fully withdrawn from the core. Calculations must indicate that in such a case the reactor would be shutdown by at least 0.01 Δk for the worst reactor condition. In the cold, flooded condition (when criticality must be avoided), subcriticality must be demonstrated with the strongest control rod fully withdrawn. In the hot, unflooded condition, a shutdown margin of at least 0.003 Δk must be demonstrated with the strongest control rod fully withdrawn and the strongest rod adjacent to it withdrawn at least 6 inches. The 0.003 Δk margin is to be demonstrated using the falling period technique.

2.4 During core alterations, one control rod in the vicinity of the alteration must be withdrawn approximately half-way and must be maintained available for rapid scram insertion. The control rod system must be interlocked so that only one such "cocked safety" rod can be withdrawn from the core at any time when criticality is not permitted (Section IV. D. 1. above).

2.5 The control rod system must be designed to permit withdrawal of only one rod at a time, and then only at a safely limited rate.

3. Liquid Poison System

3.1 The poison system must be manually actuated and must have two independent barriers against inadvertent operation.

3.2 The system must be capable of injecting poison into the moderator under all reactor conditions, and must be operable except when the control rods are fully inserted and the controls locked out. The poison system must be provided a source of emergency power.

3.3 Within about 10 minutes after actuation, the poison system must supply sufficient negative reactivity for complete shutdown of the cold, clean core with all control rods removed.

4. Emergency Cooling System

4.1 The emergency cooling system must provide sufficient steam flow through the core to limit cladding temperature rise to 200°F following reactor scram and isolation from the normal steam flow.

4.2 The emergency cooling system must be "fail-safe" with respect to automatic actuation from the safety system, and must be able to operate unattended for at least one hour.

4.3 Additional design safety criteria are listed in Section II F. 1. 8.

5. Reactor Safety Valves

5.1 A minimum of two safety valves must be provided for high pressure protection of the reactor vessel. The relief settings for these valves should be above the pressure which could be produced at the VSR by the VBWR or gas-fired boiler when they are at the relief settings of their own safety valves, but should be within the relief settings specified by the Code for a vessel designed for 1250 psig. The combined capacity of the vessel safety valves should exceed the steaming rate of the VSR at the relief pressure under the extremely hypothetical assumptions that the reactor is operating above rated power by an amount related to the positive pressure coefficient of the reactor, that all of the heat produced by the core goes into steam production, and that feedwater flow to the reactor has been stopped.

5.2 An additional relief valve should be located on the superheat steam outlet piping and discharge to the fuel storage pool. This valve should have a relief setting which would provide relief prior to discharge of the main safety valves. While this valve is not intended as a backup to emergency cooling, it would provide the initial relief path through the core, and would have the added advantage of minimizing contamination within the containment vessel. The capacity of this valve should exceed the steaming rate of the moderator at the relief pressure assuming that the reactor is operating above rated power by an amount related to the positive pressure coefficient of the reactor.

6. Reactor Safety System

6.1 The reactor safety system must be designed to detect potentially unsafe operating conditions and to initiate automatically the appropriate countermeasures where it would not be reasonable or prudent to rely on operator action. The system may be of the dual-bus type, to minimize the number of false operations, but also must be of "fail-safe" design and have adequate redundancy of components to assure that it will indicate a potentially unsafe condition if one should develop.

6.2 The primary function of the safety system is to initiate very fast insertion of all control rods ("scram") in the event of occurrence of potentially unsafe conditions requiring rapid shutdown of the reactor. The occurrence of high neutron flux or low steam flow are examples of conditions requiring fast automatic reactor shutdown.

6.3 Conditions which require plant shutdown but which do not require the speed associated with scram may cause simultaneous run-in of all control rods using their electric motor drives. If power should be unavailable to the drives for 2 seconds, scram must occur.

6.4 The containment vessel ventilation isolation valves must be closed by the safety system whenever the reactor is scrammed. All containment isolation valves must be closed on signals indicative of a reactor system rupture. The steam line isolation valves must be closed upon a signal of low pressure (to preserve moderator pressure for emergency cooling) or a signal of high radioactivity in the outlet steam lines.

6.5 The safety system must initiate emergency cooling in all cases when the outlet steam line isolation valves are closed and on any other condition indicative of inadequate cooling.

6.6 Section II. H. 6 of this report presents a detailed listing of the conditions to be monitored by the safety system and the automatic actions it will be designed to initiate in each case.

7. Special Control Features

7.1 In order to limit the amount of energy contained in the reactor system at any time, the reactor must be shut down and feedwater flow stopped when the moderator level rises to 22.5 feet, and the inlet steam flow stopped at the 23 foot level. A permissive should permit opening these valves when reactor pressure is below 500 psia.

7.2 In order to safely bypass the safety system controls which are not needed under refueling conditions, a keylocked switch shall be provided for the "refueling" mode. The bypassing shall be effective only so long as a pressure permissive indicates that the reactor is at atmospheric conditions. The "refueling" control shall activate additional controls needed during refueling. These include a control which permits "cocking" any single control rod, and a control which causes rod run-in if the micromicroammeter range switches are not set below about 10^{-3} rated power.

7.3 When a control rod and its traveling nut are not in contact, the affected control rod must be run-in automatically to establish contact. All control rods must be run-in automatically whenever the reactor is scrammed.

7.4 The micromicroammeters must prevent control rod withdrawal in the intermediate and power ranges unless they indicate a definite on-scale signal.

7.5 The micromicroammeters must provide for bypassing the period scram function above about 1% of rated power.

7.6 The low steam flow scram sensor must be adjustable down to about 20% rated flow in order to serve as a startup interlock. Together with the low pressure scram, this will backup procedural controls to provide assurance that the reactor is not started up (control rods withdrawn) until the reactor is hot and pressurized, unflooded, and a minimum steam flow established.

8. Emergency Power

8.1 The "fail-safe" design of the reactor safety system will place the reactor in a safe condition in the event of complete failure of the normal station power. A few vital loads will require emergency power in order to continue to observe the reactor neutron flux, to inject liquid poison if necessary, to run-in the control rod drive nuts, and to close backup isolation valves. In addition, it would be desirable to provide emergency power for the reactor system instrumentation, alarm system, communication system and critical lighting.

8.2 An automatic-starting emergency generator or battery power constitute satisfactory sources of emergency power.

9. Containment

9.1 The design pressure and temperature of the containment vessel (reactor enclosure) must equal or exceed the peak pressure and temperature which would accompany the "maximum credible accident" to the reactor (Section IV. L). The vessel must be designed also to resist appropriate wind loads, live loads, external pressure and earthquake loads.

9.2 The containment vessel must be designed and tested in accordance with the ASME Code as amended by Code cases applicable to reactor containment vessels. A pneumatic test must be conducted to demonstrate that the vessel leakage rate at the design pressure does not exceed 1% of the contained air per day.

9.3 The containment design must embody automatic provisions for complete isolation and lesser degrees of isolation as appropriate. The more detailed criteria for isolation provisions are given in Sections II. L. 2. 9, IV. D. 6 and IV. D. 8.

9.4 As part of the isolation provisions of the design, interlocked double-door airlocks must be provided for routine access to the containment vessel. The personnel crawl lock must have both doors of the inward-opening pressure seal type, and must be operable without any power assist. Larger equipment locks may utilize power assists and may utilize an outward opening outer door (autoclave type lock design) providing the seal and door design give assurance of proper sealing comparable to that with a pressure sealing door. The two doors of each lock must be interlocked so that only one door can be opened at a time. In addition, an alarm system based on containment air pressure must be provided to indicate any gross failure of containment integrity such as having both doors of an airlock open at the same time.

9.5 Containment integrity must be provided when the reactor is in operation, during refueling, or during other potentially hazardous periods. At any time when containment integrity is not provided, the reactor must not contain an array of fuel that would be critical if all control rods were withdrawn. Fuel in the reactor at such a time could not be repositioned or added to, but could be removed, without first assuring that containment integrity had been re-established.

9.6 A water spray system must be provided within the containment vessel to facilitate post-accident pressure reduction of the containment contents and washdown of radioactive materials. Containment pressure should be reduced to about 5 psig within 24 hours after a primary system rupture accident. Since this can be accomplished with just natural cooling in the VSR design, the heat removal capacity and reliability of the containment spray system are not critical.

10. Missile Considerations

10.1 To minimize the possibility of creating missiles which might damage the reactor system or containment vessel, only the equipment which it is essential to contain is located within the containment vessel. For example, the turbine-generator and its auxiliaries are located outside the containment vessel, and as an added precaution, the containment vessel is so located as not to intersect the plane of rotation of the turbine.

10.2 The reactor vessel and other high pressure equipment located inside the containment vessel are surrounded by concrete which serves as a barrier to protect the containment vessel from potential missiles in the event of a system rupture or mechanical failure.

10.3 The reactor vessel design with respect to fast neutron embrittlement is based on a limit of 5×10^{19} nvt integrated flux over the lifetime of the vessel (design value is actually about 1×10^{19} nvt in 20 years). Based on current knowledge, this is considered a reasonable design limit to avoid an increase in the nil ductility transition temperature of the vessel material which could be of significance at the normal reactor operating temperatures. Thus, in the event of a reactor vessel rupture, a ductile type failure not conducive to the generation of missiles would be expected. Samples of the vessel material will be irradiated as the reactor is operated to permit periodic checking of the change in material properties due to irradiation.

E. REACTOR STABILITY

1. The nuclear stability of the reactor was discussed in Section II. C. 6 and the hydraulic stability of the moderator in Section II. D. 5. Factors which tend to preclude instabilities of any kind include the small size of the core, the low conductivity of the fuel, the low steam void fraction in the moderator, the high system pressure, and the low reactivity in voids in the moderator.

2. If, contrary to expectations, the reactor should experience some form of xenon or nuclear-hydraulic induced instability, it would be principally an operational problem rather than a safety problem. If the oscillations produced were beyond the control of the operator, the high neutron flux sensors or flux/flow sensors would shut the reactor down safely.

F. SYSTEM RESPONSE TO OFF-STANDARD CONDITIONS

1. Transient Load Pickup and Drop

1.1 Transient response data to small changes in superheated steam flow (i.e., the "load") have been obtained by simulation with an analog computer. In all of the transient runs, the reactor showed a marked degree of stability by leveling reactor power at a new steady-state value after a short power transient.

1.2 Figure IV.1 shows the response of an average fuel element to a step change in steam flow from 100% to 90% and back to 100%, starting at 100% rated power. On the decrease in flow the power drops about 4%, the average fuel center temperature rises about 8°F and then drops to a new equilibrium value about 5°F below rated. The inner annulus and outer annulus fuel cladding temperatures rise as indicated on Figure IV.1. When steam flow is returned to 100%, the reactor power and temperatures return to the original values. The analog model neglects pressure and void reactivity effects.

2. Power Failure

2.1 The system response to electrical power failures is of interest because the interruption of power to some or all of the electrically operated or controlled components is a definite possibility. The effect of any power failure on the reactor will depend on the affected distribution circuits and on their service relation to the reactor control and cooling systems. In general, the electrically operated components of the plant are so designed that interruption of power would result in action which produces a safe shutdown of the reactor and prevents the occurrence of a potentially hazardous condition.

2.2 If the power failure occurred during power operation with both the VSR and VBWR operating, but with the turbine bypassed, the following sequence of events would occur automatically:

2.2.1 Both reactors would scram when power was lost.

2.2.2 The flow of saturated steam from VBWR and the gas-fired boiler would be cut off due to closure of the regulating valves PCV-3 and PCV-7 (see Figure II.26 for identification of valves and controllers used in this discussion).

2.2.3 The flow of superheated steam from the VSR to both condensers would be cut off due to closure of pressure regulating valves PCV-5 and PCV-6.

2.2.4 The emergency cooling system trip valves would have opened at the time of power failure, to automatically maintain sufficient shutdown steam flow.

FIG. IV. 1

NOTE: TEMPERATURES ARE CORE AVERAGES
STEP CHANGE IN STEAM FLOW RATE FROM 100% TO 90% TO 100%
DOPPLER COEFFICIENT = 66¢/1000°F

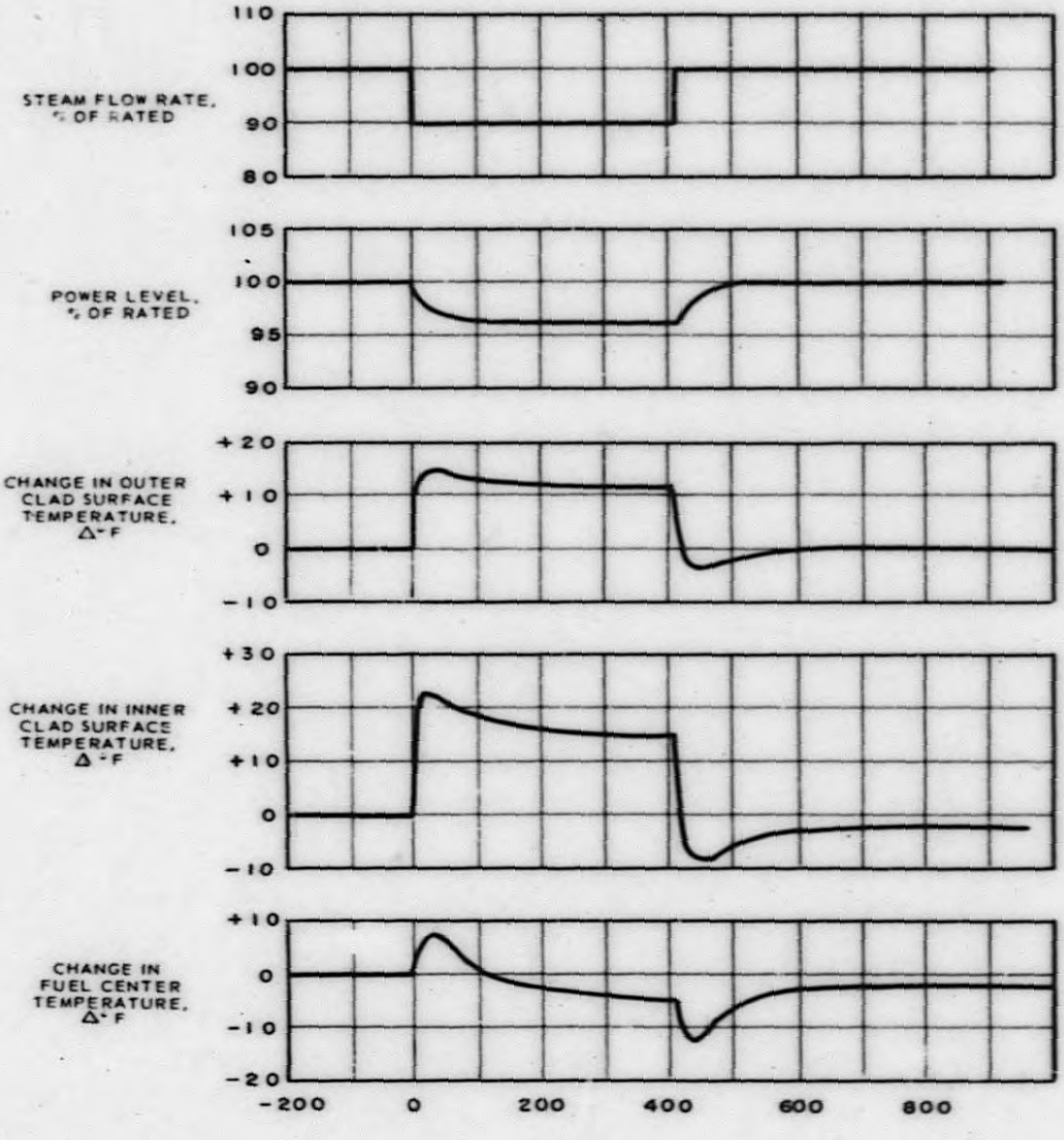


FIGURE IV.1
RESPONSE TO STEP CHANGE IN STEAM FLOW

2.2.5 The power failure would cause isolation valves to close, the condensate and feedwater pumps to coast to a stop, and the ventilation systems to stop.

2.2.6 The emergency diesel-generator would come on the line within a few seconds, to supply the emergency loads indicated in Section II.F.4.

2.2.7 As indicated in Section II.F.4, the control rod position indicators and one neutron flux channel are operated normally from the station battery and would be available without any interruption during the power failure.

2.2.8 The emergency cooling system would operate for at least one hour without replenishing water to the emergency condenser. Automatic make-up is then provided to maintain a minimum level in the condenser.

2.3 If the situation were the same as in Paragraph 2.2 above, except that the turbine were in operation, the principal difference would be the somewhat more rapid decrease in steam flow due to the fast-acting turbine trip valves. Again, all responses would be adequately safe.

2.4 If the reactor were shutdown when the power failure occurred, the principal point of interest would be continued cooling of the reactor. If the reactor were on shut down steam flow for cooling, the system would still be pressurized and the emergency cooling system would take over. If the reactor had been flooded and was being cooled by the shutdown heat exchanger, then decay heat would cause a slow rise in system temperature. If the reactor vessel was open and the power failure lasted for a considerable period of time, slow evaporation of water from the vessel could occur. This water could be made up.

3. Change in Pressure Controller Set Point

3.1 In all cases, the pressure control systems tend to compensate for changes in controller set points with no instability, safety hazard or core damage. In the worst cases, or where the set point is drastically altered, the VSR is safeguarded by scram at pressure and flow levels which are designed to prevent core or vessel damage. For identification of valves and controllers referred to in the discussion below, see Figure II.26.

3.2 If the reactor is operating with the gas-fired boiler as the only source of steam flow, the pressure controller PC-7 acts on a signal from PT-7 to control VSR reactor pressure by throttling the steam flow from the gas-fired boiler. When VBWR is not supplying steam to VSR, a decrease in the set point of PC-7 produces a decrease in flow to the VSR and a drop in reactor pressure to the new set point. Flow decreases because reactor outlet pressure is being held constant. If the flow decrease is large enough, scram occurs; otherwise, the new, but still safe, flow (and temperatures) will be held. Just the opposite sequence of events occurs if the set point is increased, except that there is no high flow scram. Scram would occur from high pressure if the pressure change was large enough.

3.3 When VBWR and the boiler are both supplying steam to VSR, PCV-3 will be at the load limit and PCV-7 will be regulating pressure. If the set point of PC-7 is decreased, PCV-7 closes until the new set point is satisfied. The lower flow will be held, if safe. If the set point of PC-7 is increased, PCV-7 will open until the new set point is satisfied. The higher flow will be maintained. Increasing the set point of PC-3 will not do anything because PCV-3 cannot open beyond its load limit. The set point of PC-3 has to be decreased to below that of PC-7 before PCV-3 starts to close. PCV-7 opens as PCV-3 closes to maintain PC-7 set pressure. The action continues until either PCV-3 is completely closed (PCV-7 regulating), or PCV-7 is completely open (PCV-3 regulating). With PCV-3 regulating, the flow will decrease. The bypass valve PCV-8 opens as PCV-3 closes to protect VBWR from high pressure.

3.4 Controllers PC-5 and PC-6 control the superheated steam pressure at the reactor outlet with PCV-5 and PCV-6 throttling valves to the VBWR and VSR condensers. Controller PC-4 regulates turbine pressure through throttling valve PCV-4. For relatively low flow, PC-6 alone will be controlling since PCV-5 will be closed. Increasing the set point of PC-6 causes PCV-6 to close until the new point is satisfied. The opposite action will result from a decrease in set point. The resultant flows are held, if safe. For larger flows, PCV-6 will be at an open limit determined by the vacuum controller VC-1, and PC-5 will be controlling. Changes in set point of PC-5 results in similar action as for PC-6. The resultant flows, in both cases, are affected slightly by the amount of flow to the VBWR turbine. Changing the set point of PC-4 changes the turbine inlet pressure but should have a negligible effect on VSR steam flow since the turbine is under governor control which determines the turbine steam flow.

4. Control Rod Run-In

4.1 The results of an analog computer study of the reactor response to a control rod run-in is shown on Figure IV. 2. In the study it was assumed that reactivity was removed at a constant rate of about 12¢/second, with the total reactivity removed equal to about $3.60. Neutron flux drops to 10% in about 20 seconds, indicating that rod run-in will provide a very rapid shutdown of the reactor.

4.2 The only possible problem indicated by the study is the rapid cooling of the core and outlet steam piping if flow is not reduced. In about 30 seconds, the outlet steam temperature and inner clad temperature in an average fuel element drop about 200°F. Other temperature responses in an average element are shown on Figure IV. 2. The effect of the transient is very similar to that which occurs when the reactor is scrammed.

5. Control Rod Run-Out

5.1 Figure IV. 3 shows the reactor response to a control rod run-out at 5¢/second, starting at rated reactor power, and terminated by a high flux scram at 125% rated power. The 5¢/second ramp rate corresponds to the maximum attainable rod run-out rate for the reactor. The average fuel temperature rise amounts to only about 1°F.

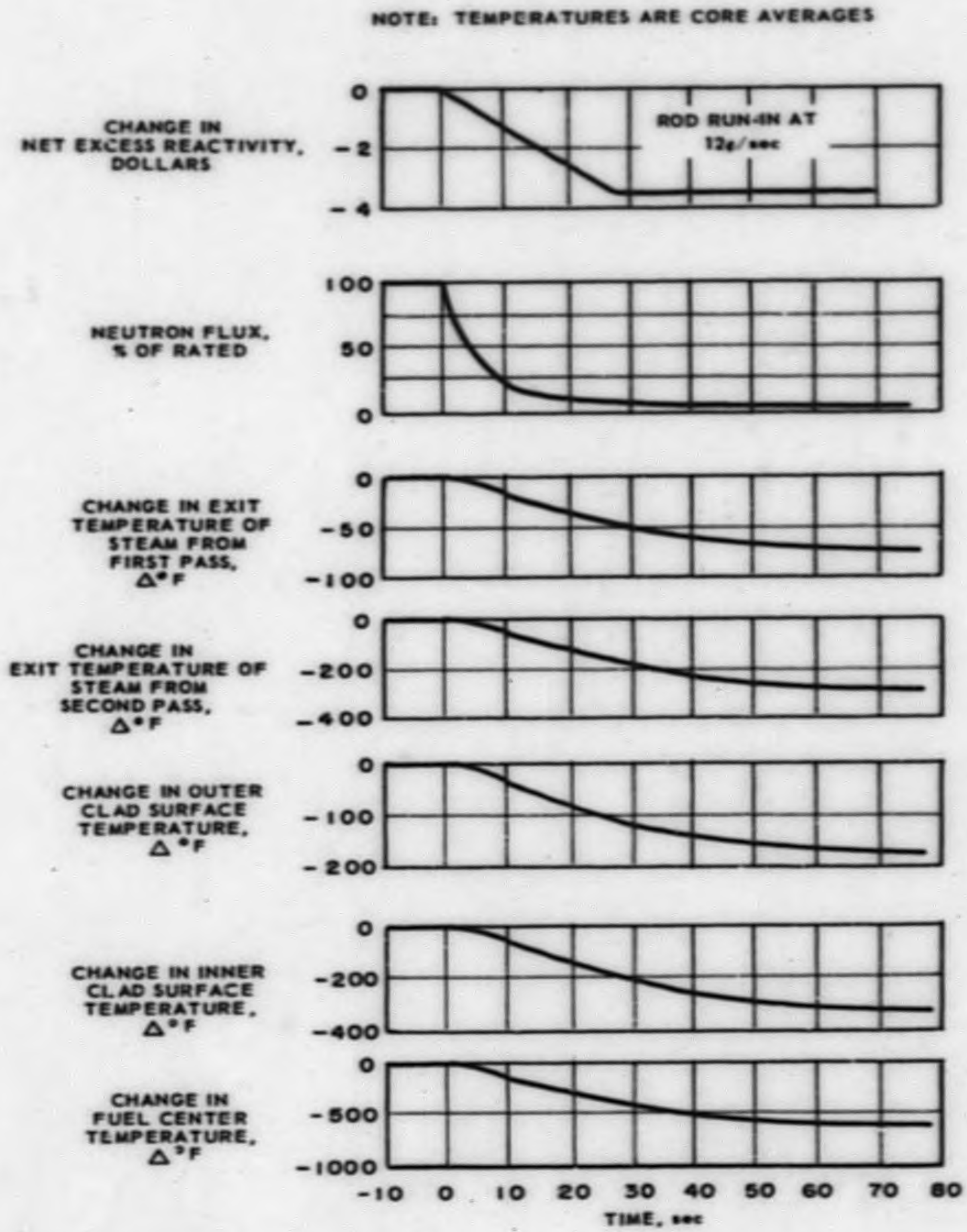


FIGURE IV.2
CONTROL ROD RUN-IN AT RATED POWER

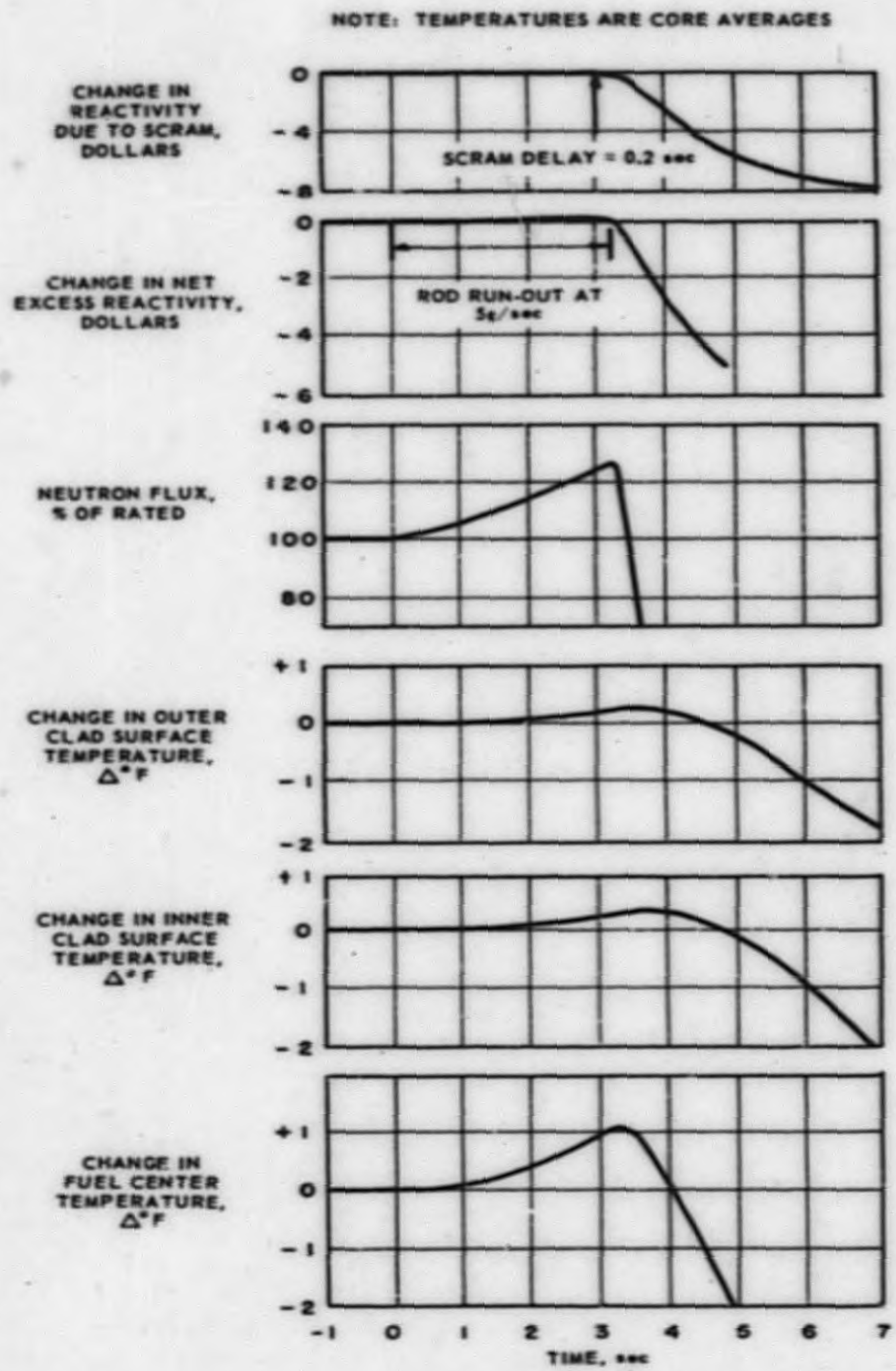


FIGURE IV.3
CONTROL ROD RUN-OUT AT RATED POWER

5.2 A greater temperature rise would occur if the reactivity ramp were assumed to occur at a lower initial power level. However, the scram level also changes if the initial power is sufficiently lower than rated for the flux instrumentation to be on a lower range. The most severe accidents of the rod run-out type, which assume multiple failures of equipment and procedures, are presented in Section IV. J.

6. Change in Superheat Steam Flow Bias Control Valve Settings

6.1 The flow through one to four fuel bundles connected to a common superheat steam headering line is controlled by a "bias" valve in each line. Operation of one of these valves changes the flow through the fuel bundles connected to its line. If one of these valves is operated (manually) during operation, the flow through the rest of the core is transiently changed by a small amount, but the flow control system returns the core pressure drop to the original value and thus restores the original flow through the rest of the core. The resulting reactor pressure and flux transients are negligible.

6.2 If the flow through any fuel bundle is reduced toward a value which would cause overheating of the clad, an alarm sounds on high outlet steam temperature from the bundle affected, to warn the operator to take necessary corrective action. Low flow trips are provided also on many of the individual bundles and on groups of bundles. In addition, the bias control valves are designed for very slow movement, and cause reactor shutdown (control rod run-in) if any one of the valves is closed beyond a minimum allowable position.

7. Turbine Trip

7.1 It is presently anticipated that the VSR condenser will always receive at least 50% of the steam flow from the reactor. Thus, the VBWR turbine would receive 50% or less of the flow. In the event of a turbine trip, the flow to the turbine would be reduced very rapidly. The control valves to the condenser would respond very quickly to compensate for the reduced core flow. Depending upon the initial fraction of total flow to the turbine, the response time of the control valves, the location of the low flow scram sensor (reactor inlet or outlet), and the low flow scram setting, the flow reduction on turbine trip might or might not cause a low flow scram before flow could be brought back to normal by the fast-acting control valves. If the control valves worked normally, the scram would be unnecessary. Without a scram on low flow, calculations indicate that the reactor pressure rise would be less than 15 psi and the flux increase due to the pressure rise would be less than 10%.

7.2 If the reactor flow control system should fail to increase flow to the normal value following the turbine trip, the reactor would scram on a high flux/flow ratio of 1.25 and activate emergency cooling at a 1.35 ratio. These flux/flow settings are based on an assumed maximum allowable cladding temperature rise of 200°F for such accident situations.

8. VBWR Transient Conditions

8.1 If the VBWR power is increased when the gas-fired boiler is not operating, the excess steam produced will be bypassed automatically to the VBWR condenser. If the boiler is operating, the pressure regulator will reduce steam flow from the boiler to hold the total flow to the VSR constant. The increase in boiler pressure would reduce the gas firing as necessary.

8.2 If VBWR power is decreased, flow from VBWR would decrease. If the gas-fired boiler were not operating to make up the difference in total flow, the VSR inlet steam pressure and flow would decrease. If the VBWR power decrease were large enough, the VSR would scram from low pressure or high flux/flow ratio.

8.3 The effect of a pressure change in VBWR would be the same as for power changes described above. The effect of a VBWR scram would be the same as noted above also. In fact, all VBWR transients would be reflected at VSR in terms of flow and pressure changes. The VSR system response will be adequate to compensate for the transients, if small, or to scram the VSR if the flow and pressure effects are large.

G. SAFETY IN EVENT OF EQUIPMENT MALFUNCTION

1. Pump Malfunctions

1.1 Condensate pump failure will result in cavitation of the reactor feedwater pump suction, and rapid loss of feedwater flow (with the consequences noted in the paragraph below). Indications of condensate pump failure are circuit breaker pilot light indication, pump discharge pressure loss indication and feedwater flow loss indication. Two condensate pumps are provided, each having automatic switchover to the standby pump if the second pump is not in operation.

1.2 Loss of the reactor feedwater pump results in loss of moderator makeup to the reactor, loss of feedwater to the gas-fired boiler, loss of control rod drive seal water, and loss of desuperheating water flow. Indications of failure are available from the circuit breaker, discharge pressure and flow indication. Continued loss of flow would result in a reactor scram on low water level. Prior to this, the reactor would be shut down by a rod run-in signal which is initiated when the feedwater pump is tripped.

1.3 Failure of the circulating water pump results in loss of cooling water to the VSR condenser and subsequent loss of condenser vacuum. Indication of the failure is available from the circuit pilot light, circulating water pressure and condenser vacuum. If the failure were due to loss of electrical power, reactor scram might occur for the same reason. If not, the reactor would be shut down by rod run-in when condenser vacuum dropped sufficiently. Closure of the steam flow control valve to the condenser, resulting from low condenser vacuum, would decrease reactor steam flow and increase pressure. Either of these latter signals could scram the reactor.

1.4 Failure of the auxiliary cooling water pump results in loss of cooling water to the shield cooling coils, to the scram system air compressor, and to the cleanup and shutdown heat exchanger. Indications of failure are from the circuit pilot light, pressure guage, temperature of compressor outlet air, shield cooling water outlet temperature, and shutdown cooling heat exchanger temperature indication. Continued loss of auxiliary cooling water will cause alarms but will not shut the reactor down. For example, an alarm will occur if the shield cooling circuit loses pressure or if the cleanup system exceeds a temperature limit. Accumulators are provided for scram air for each drive, so that failure of cooling to the compressor is not serious. If any accumulator should lose pressure at this time, however, this would cause automatic run-in of all control rods.

1.5 Failure of the condensate transfer pump results in possible loss of the ability to maintain the desired levels in the condenser hot wells when operating in combination with VBWR. Besides pump pressure and circuit pilot lights, indication of failure is provided by level indication on the condenser hot wells. Continued failure of the condensate transfer pump could eventually lead to loss of suction to the condensate pumps. The effects on the reactor would then be the same as described for failure of the condensate and feed pumps.

1.6 The fuel storage pool filter pump will sound an alarm in the event of failure. There would be no effect on the reactor and no hazard from failure of this pump.

1.7 The liquid poison pumps are provided in duplicate, have very little duty, and should therefore be very reliable. Only one of the pumps is required to inject poison at the desired rate. The pumps are provided an emergency source of power from the diesel-generator. The possibility of failure of both pumps at any time when they are required is extremely remote.

2. Valve Malfunctions

2.1 The direction of motion of any valve upon loss of air pressure or power is determined, in general, by the valve design. Valves are designed to fail open, fail closed, or to hold the last position before power failure. However, under some conditions of failure of the associated control circuit, a valve may not move when it is supposed to or may move in the opposite direction from that intended by the designer. The discussion in subsequent paragraphs about the effect on the reactor system of various valve malfunctions therefore considers failures in both the open and closed directions. Failures in the "as is" position are considered less severe than the worst of the "open" or "closed" cases, and will not be discussed. The discussion considers the more significant valves which have a direct influence on the reactor. Many of the control valves are referred to by the numbering system shown on Figure II. 26.

2.2 The pressure control valve PCV-3 from VBWR is designed to fail open. If it fails open, the steam flow to the VSR will increase, VSR pressure will increase, and VBWR pressure will decrease. The gas-fired boiler pressure controller PC-7 will reduce boiler steam flow to maintain VSR pressure and flow at the desired values. The pressure and flow transients at VSR are expected to be small. If VBWR alone is supplying steam, the VSR pressure and flow will increase to an equilibrium with VBWR. The VSR may scram from high pressure and the VBWR from high flow. The effect of VBWR scram has been discussed in Section IV.F.8.

2.3 If PCV-3 fails closed, VSR pressure and flow decrease transiently, but the action of PC-7 will re-establish pressure and flow if the boiler has enough reserve. If it does not, VSR will scram from low pressure or high flux/flow ratio. VBWR pressure will increase, but it is protected by its steam bypass valve PCV-8. If VBWR alone is supplying steam, VSR will scram due to failure of PCV-3 to the closed position. The effect on VBWR is the same whether the boiler is operating or not.

2.4 The steam outlet pressure control valve PCV-4 to the VBWR turbine is designed to fail closed. If PCV-4 fails closed, the decrease in VSR steam flow is made up by PCV-6 or PCV-5 which discharges steam to the condensers. The pressure, flux and flow transients should be below trip levels. If PCV-4 fails open, the turbine governor-control valve will act to maintain turbine flow.

2.5 Control valve PCV-5, which dumps steam to the VBWR condenser, is designed to fail closed. If it fails closed, the resulting low flow will scram the VSR. PCV-6 cannot make up the loss in flow by dumping more steam to the VSR condenser because it will be operating at load limit whenever the VBWR condenser is being used to dump steam. If PCV-5 fails open, VSR flow will increase transiently until PCV-6 restores the original flow. Since the flow transient will not be large, the VSR is not expected to scram on low pressure or high flux (caused by lowered fuel temperature and the resulting positive Doppler reactivity effect.

2.6 The VSR condenser steam dump valve PCV-6 is designed to fail closed. If it fails closed, VSR will scram from high flux/flow signal since PCV-5 will be unable to make up the loss of flow unless PCV-6 was operating at a low enough flow (PCV-5 closed because of lower pressure set point). The worst case of PCV-6 failing closed is when only the VSR condenser is being used to dump steam. An analysis of the transient which results in such a case is summarized in Section IV. J.

2.7 If PCV-6 fails open, the resulting high flow will be reduced by PCV-5 if its flow at the time of failure is in excess of the increase in flow through PCV-6. However, the resulting overload on the VSR condenser will probably cause control rod run-in from a signal of high condenser pressure. Scram from low pressure or high flux due to low temperature (Doppler) is less likely.

2.8 The pressure control valve PCV-7 from the gas-fired boiler is designed to fail open. If it fails open, VSR flow and pressure will increase transiently until PCV-3 closes to restore normal conditions. PCV-3 will then be regulating pressure. However, if PCV-3 cannot decrease flow sufficiently, the VSR will scram on high pressure or high flux caused by rising pressure. If PCV-7 fails closed, VSR flow and pressure will decrease and cause a low pressure or high flux/flow scram. PCV-3 will be unable to make up the flow because it will be at load limit.

2.9 The VSR level control valve LCV-2 is designed to fail closed. If it fails closed, the lack of feedwater will cause the water level in VSR to drop. A low water level trip will cause scram at a level which is still safe in all respects. If LCV-2 fails open, the maximum rate of feedwater will cause the water level to rise with subsequent scram at a safe high water level trip point.

2.10 Two automatic inlet steam isolation valves are provided in series; one is designed to fail closed and the other to fail as is. If either one fails closed, the VSR will shut down (automatic rod run-in) when the valve is still almost fully open. If the reactor pressure drops low enough, emergency cooling will be initiated also. The pressure at VBWR and the boiler will rise. VBWR is protected by its steam bypass system. The boiler is pressure-regulated, but if analysis shows that it is unable to prevent boiler pressure from reaching the safety valve setting in this case, a pressure-regulated steam bypass line to the VSR condenser will be provided for the momentary relief required. If either inlet steam isolation valve fails open, the containment vessel will still be isolated automatically by the other valve if isolation is called for (both the main and backup valves are automatic on this line).

2.11 On each of the two outlet steam lines, two isolation valves are provided in series; one valve is designed to fail closed and the other to fail as is. As in the case of the inlet steam isolation valves, both the main and backup isolation valves are automatic. If any one of these valves fails closed, the VSR is shutdown by rod run-in when the valve is still almost fully open. Emergency cooling is also initiated. The VBWR and the boiler are protected in the same way as indicated in the paragraph above. If any one of the outlet steam isolation valves fails open, containment isolation is still automatic if it should be required.

2.12 There are two emergency cooling valves in parallel to the emergency cooling steam dump tank. Both valves are designed to fail open. If one of the valves fails open during reactor operation it will cause the reactor to be shut down by automatic rod run-in. If one of the valves should fail closed, this would not impair emergency cooling since essentially full flow is possible through either valve.

2.13 There are two emergency cooling valves in parallel to the emergency condenser. Both valves are designed to fail open and only one must open in order to provide the desired emergency heat exchanging capacity. The situation is similar to that for the emergency cooling steam dump tank except that the reactor need not be shut down automatically in the event one of the valves fails open during operation since the capacity of the condenser is not impaired as long as it is supplied with sufficient cooling water.

3. Instrument Failure

3.1 Each of the nuclear period channels consists of a compensated ion chamber, dual (plus and minus) high voltage power supply and log-N period amplifier. Loss of signal current to the amplifier could be due to failure of the chamber (open circuit of connections from chamber electrodes to the connectors), open circuit or short in cables or connectors, or failures of the plus high voltage supply. The manifestation of the loss of signal is a downscale or negative log-N indication which is detected by comparison with the other log-N meter. Loss of the minus high voltage will show up as a higher log-N indication if there is significant gamma flux. No "false" period scram should occur unless the failure, in the process of occurring, induces a transient short period signal.

3.2 The log-N period amplifier can fail in two ways: (1) in a way in which the trip circuitry or internal power supply fails "safe" to scram the reactor or (2) in a way in which the amplifier section fails, resulting in no indication as in "loss of signal" case discussed in the paragraph above. In any case where log-N indication is lost, a period trip from the faulty channel is not available if a short period should occur. However, the second period channel gives full period protection since only one period trip is required to indicate scram on both busses of the safety system. Furthermore, the three flux micromicroammeters offer close back-up protection since in the period or intermediate power range they are in their low ranges, monitoring the neutron flux, with trip points set at 125% on each scale.

3.3 The period scram function is bypassed once the power range is reached and failures in the period channels will no longer be able to cause a spurious reactor shutdown.

3.4 Each of the power range neutron flux monitoring channels consists of a compensated ion chamber, a dual high voltage power supply and a micromicroammeter. The same failures can be postulated to produce loss of trip or spurious trip as for the period channels. Since two out of the three flux channels have to trip for scram, a failure which causes a trip will not cause scram. A failure which causes the indication to go to zero will be annunciated from the on-scale rod withdrawal permissive. With either type of failure, the faulty unit can be taken out for servicing leaving one bus of the safety system in a tripped condition. In this condition the reactor is fully protected by the remaining safety system bus but is more susceptable to unneeded shutdowns from any spurious trip in the remaining bus.

3.5 All the scram functions involving level and pressure (including vacuum), seismic disturbance, and each steam isolation valve, have four contact-type trip sensors. Either one of two trip contacts in each safety bus will trip that bus. The other safety bus must also trip to produce a reactor scram. It is possible for a trip contact to fail open or closed. If it fails closed, then it is unable to cause trip. However, the second sensor in that safety bus will still be able to trip that bus and there is no compromise of safety.

3.6 Each flux/flow scram unit consists of a compensated ion chamber, dual high voltage power supply, a flux amplifier, a flow sensor, and a ratio amplifier. The same failures can be postulated to produce loss of trip of spurious trip of the flux unit as for the period channels. The ratio amplifier would be unable to trip with loss of the flux signal. However, the second ratio amplifier gives full protection since only one flux/flow trip is required for scram. Loss of the flow signal, on the other hand, would cause an immediate trip and scram.

3.7 The most likely failures of the pressure and level controllers and sensing elements would cause the controlled valves to become fully opened or fully closed. Such valve conditions have been examined in Section IV.G.2. Where the failure results in loss of control, the valve stays in a fixed position or moves erratically without going to its limits. The affected variable (level, flow, pressure) will change eventually to an alarm or scram point. The operator may be able to switch to remote-manual control should he notice the failure in time. In any case, the reactor safety system gives full protection.

3.8 Any failure in VBWR instrumentation will be reflected in steam flow and pressure changes at the VSR. The VSR response to such cases is discussed in Section IV.F.8.

4. Stuck Control Rod

4.1 If a control rod should become stuck in the inserted or partially inserted position, this would become evident when an attempt was made to withdraw the control rod. When the motor driven nut-carriage is withdrawn slightly and the control rod does not follow it, a separation switch indicates the lack of following and causes the motor to move the nut-carriage in the rod insertion direction until engagement is again established. The control rod shaft is provided with latching fingers to prevent rod withdrawal if the separation devices should fail and the stuck rod should become unstuck.

4.2 If a control blade should become separated from the drive shaft inside the reactor vessel, and should become stuck in an inserted position, procedural controls are required to detect the situation before the drive shaft could be withdrawn sufficiently so that a severe accident could result from the blade becoming unstuck and dropping from the core. The principal procedural controls are the use of rod patterns which minimize the reactivity worth of individual control rods, and the constant observation of neutron flux for anticipated response to small increments of rod withdrawal. Appropriate procedures will be included in the normal operating procedures for the plant. An analysis of the worst case of a control blade dropping out of the core is presented in Section IV. J.

4.3 If a control rod should become stuck in the withdrawn or partially withdrawn position, safety of the reactor would not be impaired. The control rod system is so designed that the reactor can be fully shutdown in its most reactive condition with any one control rod in its fully withdrawn position.

5. Cladding Failure

5.1 Analytical predictions of cladding failure for the annular fuel elements is made difficult by the lack of definitive knowledge in several areas including the interplay between the fuel and cladding, the complex thermal stresses, high temperature properties of the materials, and cladding plasticity. It is to investigate such areas as these that the great need for superheat fuel testing exists. However, based on the calculations that have been made, it appears that if cladding failures were to occur they probably would be due to plastic cycling of the cladding in regions of high stress concentrations. If such were the case, the failures would be in the form of fine fissures on the clad surface, forming gradually or following a severe transient. As the fissure began to form and grow, the activity in the outlet steam would be expected to increase.

5.2 The failure at a bellows in a superheat fuel element under test in the SADE loop of the VBWR caused release of fission gases at the rate of about 200 microcuries per second. Some carryover of radioactivity into the outlet steam line occurred also. The VSR system is designed to accommodate failures of this magnitude without necessarily interrupting operation. The longer-term effects of system contamination for such cases are of interest as part of the VSR development program. In fact, it is planned to irradiate fuel that has purposely defected cladding in order to investigate these problems. Thus, isolated cases of cladding failure will not compromise the safety of the reactor. The principal safety measures in the event of cladding failure are the condenser systems and the off-gas system, which retain most of the biologically significant activities and release the rest within established permissible limits. The off-gas lines to the stack are isolated automatically if releases to the environment in excess of the established limits might otherwise occur.

5.3 In hypothetical cases of severe steam flow starvation to the fuel, where cladding temperatures may rise above the normal operation limits, the failure points are not well established. The stainless steel cladding loses strength very rapidly as temperatures are increased above 1500 to 1600°F. Higher temperatures would probably cause distortion of the cladding, making the fuel unsuitable for continued operation, but probably would not cause any gross failure. As long as the system is pressurized, the pressure external to the cladding will help prevent cladding failure from internal gas pressures. Temperatures near the cladding melting point of about 2600°F are probably required in order to cause release of the fission gases contained by the cladding. Even under conditions of gross clad failure or melting, the release of fission gases should be relatively small. Experiments with UO_2 fuel pellets have shown that for UO_2 operating temperature below about 4000°F, the release of gaseous fission products is less than 1%. If the same accident which caused the cladding to melt also caused the UO_2 temperatures to reach 4000 - 5000°F, then the release of gaseous fission products would be substantially greater.

5.4 The attainment of cladding temperatures near the melting point suggests complete loss of steam cooling flow. If failed cladding should release fission products when steam flow has stopped, then the fission products would be contained within the reactor system within the containment vessel. If there were sufficient flow for fission products to be carried into the outlet steam lines, the radiation monitors on the steam lines would cause reactor shutdown and isolation of the outlet steam lines if the radioactivity levels were high enough. Any excessive amounts of radioactive materials that had passed the steam line isolation valves would be effectively confined within the condenser and off-gas systems by automatic closure of the off-gas line valves. These safety features effectively protect the health and safety of the public even in the event of a very gross failure of fuel cladding. Section IV.J presents analyses of loss of flow accidents and nuclear transients that indicate the safety against attainment of cladding temperatures that could result in significant release of fission products. The discussion above has dealt only with the safety against cladding failure if it should occur irregardless of the many measures incorporated in the design to prevent it.

5.5 Consideration of cladding failures involving cladding melting temperatures suggests the possibility of violent chemical reactions between the molten cladding and the cooling steam. However, the cladding material for the Mark I core is 304 stainless steel, which is generally regarded as a very unreactive material. Thermodynamic considerations indicate that it is doubtful that stainless steel will undergo a rapid or violent reaction with steam, and that if it should react rapidly that the energy release would be an order of magnitude less than for other commonly used cladding materials such as zirconium. Calculations indicate that the maximum probable energy release from the reaction of stainless steel with water vapor is about 250 Btu/pound. To obtain any significant degree of reaction it would be necessary for the cladding to be molten and very finely divided. As there will be no mechanism to provide dispersion of a substantial portion of the cladding, a 100% reaction is extremely improbable if the reaction is possible at all. Each fuel element contains about 3.2 pounds of stainless steel in the cladding and velocity booster tube. Even a 100% reaction of this amount of steel would represent only a very small energy release. In the case where cladding melting might occur primarily from flow stoppage, a metal-steam reaction would be limited by the amount of stagnant steam available within the process tubes. A conservative calculation indicates that the reaction would be limited to less than 5% of the steel for such a case.

6. Activation of Emergency Cooling System During Operation

6.1 Should the emergency cooling system be activated while the reactor is operating normally, an interlock is provided to run the control rods in automatically. Inadvertent actuation of the emergency cooling system could occur as the result of a valve malfunction as indicated in Section IV.G.2.12. The reactor is shutdown in this case primarily because, as a result of the spurious trip, the emergency cooling steam dump tank would be unavailable to provide emergency cooling if it were actually needed. When the emergency cooling system valves have been reset and the dump tank conditions returned to "normal", the reactor can be started up again.

6.2 Since a signal for emergency cooling is generally accompanied by a condition indicating inadequate flow, or as a result of or accompanying a steam line isolation valve signal, the steam flow increase through the core will usually be less than in the case of inadvertent actuation of the emergency cooling system. The emergency cooling system is designed to provide about 80% flow for about 4 seconds, decaying to 5% in about 100 seconds. The core will therefore receive up to about 180% of rated flow, assuming it is running initially at full power, with a subsequent steam temperature drop of about 100°F. The pressure in the VSR will also begin to drop in this case. This will cause the pressure regulating valve at the outlet to close, thereby cutting off steam flow to either the turbine or condensers. At most, the pressure in the vessel could drop sufficiently to cause the reactor to scram. It is planned to study the transient effects of this situation in more detail with the aid of an analog computer as detail design of the emergency cooling system progresses.

7. Loss of Condenser Vacuum

7.1 If the condenser vacuum is lost, the main heat sink for the VSR is also lost. If no action were taken by an automatic system protection circuit, condenser pressure would increase to a point where the condenser rupture diaphragm would fail. To eliminate this possibility, a vacuum-sensing device is used to transmit a rod run-in signal to the reactor upon loss of a few inches of vacuum. As the vacuum continues to fall, a subsequent signal calls for closure of the pressure regulating valve at the VSR condenser inlet. If the reactor had not shut down due to the high condenser pressure signal, it would be scrammed from signals associated with low flow or high reactor pressure as a result of closure of the pressure regulating valve.

7.2 Upon closure of the pressure regulating valve, steam flow through the core would have to be routed to another heat sink to continue to provide the required shutdown cooling. If the VBWR condenser had been operating with the VSR system, it would automatically pick up the load. If not, the emergency cooling system would have to be actuated. An appropriate signal for actuation of emergency cooling in this case will be incorporated into the reactor safety system.

8. Piping or System Ruptures

8.1 This section deals with the safety afforded in the event of possible mechanical failure of the various pipes providing flows to or from the reactor, or to the reactor vessel itself. The primary protection against the occurrence of such accidents is the use of accepted Codes to provide a conservative design, proper installation, testing, and periodic inspection. If a failure in the system nevertheless occurs, the containment and isolation systems have been designed to minimize the uncontrolled release of radioactivity. The most severe case of a system rupture is the "maximum credible accident" to the reactor, described in Section IV. L. The discussion which follows is largely qualitative; it is planned to analyze these situations in greater detail as a part of the detailed design safeguard effort.

8.2 If a break occurred in the incoming saturated steam line inside the containment vessel, the size and exact location of the break would determine the severity of the accident. If the break were small, the leakage would be made up by the VBWR or gas-fired boiler. If the break were large, and were between the isolation valve and reactor inlet check valves, the reactor would lose inlet steam flow. The inlet check valves would prevent loss of moderator so the reactor could safely scram and be cooled with the emergency cooling system. The steam issuing from the inlet steam line would probably activate area radiation monitor alarms, particularly if the steam were from the VBWR, and would cause an increase in containment pressure. The latter signal would cause closure of the inlet steam line isolation valves, which would stop inlet flow in about an additional 10 seconds. During this time the containment pressure would rise only a few psi above the "high containment pressure" scram and isolation set point (which probably will be set for about 2 psig). The only release of radioactivity would be that from the incoming steam and all this would be held within the containment vessel.

8.3 If the incoming steam line break occurred just at one of the two reactor vessel inlet steam nozzles, the check valve located at that point would not be effective in preventing depressurization of the moderator. The reactor would scram and emergency cooling would be actuated and would operate for a few seconds until moderator pressure dropped too low. Steam flow through the core would then stop and cladding temperatures would rise, but probably would not reach the clad melting point. When the moderator was fully depressurized, it is expected that the moderator would still cover the active core region if make-up water continued to flow. However, cladding temperatures near the melting point would be expected and some release of fission product gases into the containment vessel might occur. The response of the inlet steam line and isolation valves would be the same for this line break as described in the paragraph above. The peak containment pressure achieved in this accident would approach that of the "maximum credible accident", but the radiological consequences would be less severe.

8.4 If the superheat steam outlet line broke outside the containment vessel, the rate of steam release to atmosphere would be limited to about 100 pounds per second due to the relatively high resistance to flow through the core and steam line headering. A low reactor pressure signal would scram the reactor, close the steam line isolation valves and initiate emergency cooling. The reactor would be safely shut down, moderator level and pressure would be maintained at safe values, and the fuel would not sustain any damage or release fission products. Up to about a ton of steam might escape to atmosphere before the isolation could be completed. The normal nitrogen and oxygen activities associated with this steam would have negligible radiological significance by the time it reached the site boundary. If the reactor were being operated with defective fuel, it is anticipated that any release of fission product activity that would be satisfactory for continuous operation of the reactor would not constitute a severe environmental hazard if released for the short period until isolation could be completed. If the line break occurred downstream of a pressure control valve, the valve would quickly throttle to try to hold upstream pressure and thus less steam would be released to the atmosphere.

8.5 If the outlet steam line break occurred upstream of the isolation valve inside the containment vessel, it would not be possible to isolate the reactor system. The situation would be very similar to the case of inlet line break discussed in paragraph 8.3, above, except that the fuel would be cooled better while the moderator is depressurized in this case. The fraction of fuel cooled during this period, and the rate of moderator depressurization, would depend on whether the break was in the main header steam line or the divert header steam line. No cladding melting or gross fission product release would be expected in this case as long as moderator makeup continued to keep the level up to the top of the active core. The peak containment pressure in this accident would approach that of the "maximum credible accident", but the radiological consequences would be less severe.

8.6 A break in the feedwater line upstream of the check valve would not constitute a serious accident. The effect on the reactor would be very similar to failure of feedwater from such causes as pump or valve failures, discussed in a previous section. The feedwater is cool and low in activity content so it would pose no hazard inside the containment vessel. If the line broke between the vessel nozzle and the check valve, then the failure would be serious. The effects would be similar to those discussed in paragraph 8.3, above, except cladding melting and fission product dispersion inside the containment vessel would be more likely since it would not be possible to makeup the moderator level. The feedwater nozzle is located above core level but a drop in level below this point would occur due to flashing of the moderator during depressurization through the feedwater nozzle.

8.7 Four large nozzles near the bottom of the reactor vessel are provided for possible future use. A break in the caps sealing these nozzles would cause an accident approaching the severity of the "maximum credible accident". Any assumed system rupture that would cause the reactor vessel to lose essentially all of the contained water, and make it impossible to keep makeup water in the vessel, will result in substantial melting of the fuel in the core and dispersion of fission products into the containment vessel. This type of hypothesized situation forms the basis for the "maximum credible accident".

8.8 The case of a break in an individual fuel bundle outlet steam line has been analyzed in detail because it is the break which imposes the largest pressure differentials across the process tubes and other portions of the fuel and internal superheat steam piping. This accident provides a design basis for many of these items. The process tubes are of particular interest in this case because, if the process tubes could be collapsed during reactor operation, a reactivity addition would occur. (Even then, however, it would be necessary to have simultaneous collapse of most of the process tubes in order to cause a severe transient.) To insure against such an accident, the process tubes must be designed to withstand the highest external pressure imposed when steam is passing through the fuel elements and passing out the broken nozzle. The normal maximum pressure differential across the 25-mil process tubes is about 13 psi. The calculations indicate that breaking a fuel bundle outlet steam line of the largest size provided would cause a maximum pressure differential of about 72 psi across the process tubes in that bundle. The detailed results of the calculation are shown on Figure IV.4. They indicate also that a high pressure differential will exist across the fuel (across the outer and inner pass) and that the outlet steam lines inside the reactor vessel will have to withstand almost a full 1000 psi pressure differential in order not to collapse and cut off flow through the fuel bundle in the assumed accident. The process tubes are designed to withstand an external pressure of at least 200 psi before collapse. Even considering possible errors in the calculation of the 72 psi maximum accidental pressure differential, the tube strength is sufficient to provide a safety factor of at least 2 against collapse.

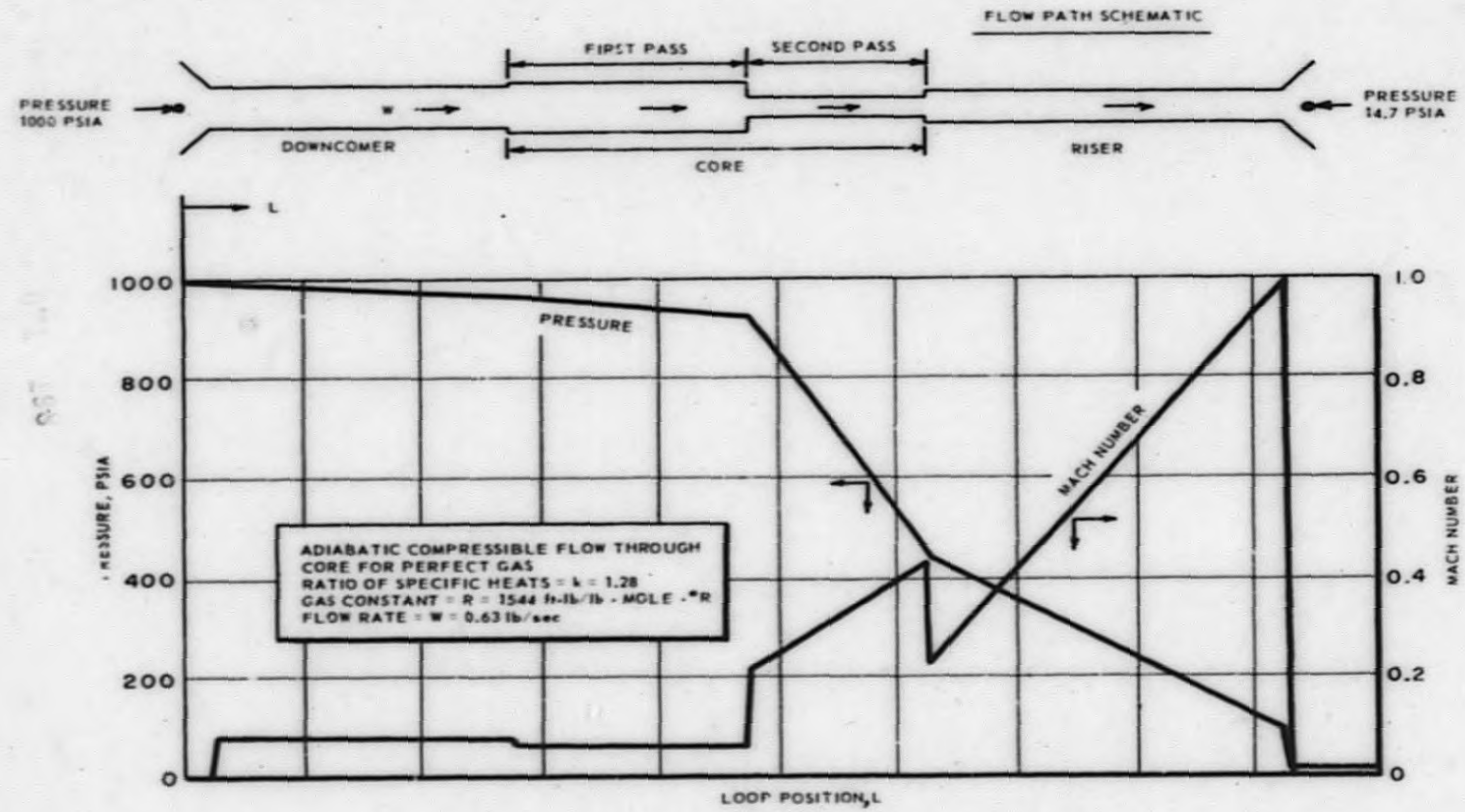


FIGURE IV.4
MAXIMUM PRESSURE DROPS FROM OUTLET STEAM LINE BREAK

FIG IV.4

8.9 In the event of the pipe break accident assumed in the paragraph above, the moderator would be slowly depressurized and the containment vessel slowly pressurized. As there would be no way to isolate the broken line, peak containment pressures would be significant but would not closely approach those attained in the "maximum credible accident" due to the longer time available for heat transfer for the relatively small break assumed. The reactor would be scrammed, the containment isolated, and emergency cooling actuated. Depressurization would be sufficiently slow that emergency cooling would be effective in avoiding excessive cladding temperatures initially. When emergency cooling became ineffective, cladding temperatures would rise but would not reach the melting point, so that substantial fission product release into the containment vessel would not be expected.

8.10 In the paragraphs above in this section, the conclusion that cladding melting will not occur if the fuel is cooled by steam flow for only a few seconds after scram and then the water level remains above the active core, is based on the characteristics of the Mark I fuel and the power densities indicated in this report. For other fuel types, particularly if they involve substantially higher power densities, the conclusion might well be that cladding melting would be expected if emergency cooling flow cannot be maintained for longer periods. In such cases, the fission product dispersion into the containment vessel would be increased, but it would not exceed that which is assumed to occur in the event of the "maximum credible accident" presented in Section IV. L.

H. SAFETY IN EVENT OF OPERATOR ERROR

1. Rod Withdrawal Error at Startup

1.1 The reactor will not be started up nuclearly until it has been preheated to the hot, pressurized condition, the process tubes unflooded, and a minimum steam flow established. There is no simple error that could be made that would result in the operator withdrawing rods and starting the reactor up before these steps have been accomplished. The principal non-procedural safeguards are the pressure and minimum flow requirements (interlocks) that must be satisfied in order to start up. The pressure interlock will be effective until the reactor temperature is sufficiently high that the desired reactivity coefficients of the core are obtained. The flow interlock gives assurance that the core has been unflooded at least sufficiently to pass a minimum flow, and precludes the possibility that the unflooding procedure could be forgotten entirely. The case of startup when the process tubes have not been completely unflooded is discussed in a later section.

1.2 Assuming that the reactor has been properly prepared for startup, the most severe error that the operator could make is to withdraw the control rods at the maximum rate and put the reactor on a reactivity transient. The safeguards provided to minimize the effects of such an error include the slow control rod withdrawal speeds provided to limit the maximum rate of reactivity addition to a safe value, the period scram function to shut the reactor down before it can attain a very short period, and the neutron flux scram trips at 125% of scale on any range. The latter safeguard, coupled with the on-scale rod withdrawal permissive in the intermediate and power ranges, assures that a flux scram setting is relatively near the existing flux at any time when rods are being withdrawn.

1.3 Section IV. J presents several analyses of severe reactor transients including a reactor startup accident. The analysis includes assumed failures besides the operator error indicated in the paragraph above. The effects on the reactor are shown to be safe even if some of the safeguards should fail.

2. Reactor Startup While Flooded

2.1 Section IV. H. 1, above, described some of the safeguards against starting the reactor up before it has been unflooded. In addition to the minimum flow interlock, which assures that some of the process tubes have been unflooded, another strong safeguard is provided. Each of the 12 outlet steam flow control bias valves is equipped with switches to cause rod run-in if the valves are not opened beyond a minimum position at startup (or are closed too far during operation). This provision requires that all 12 individual header valves have been opened and subjected to the unflooding operation before startup can occur. In combination with the many procedural safeguards associated with preparing the reactor for startup and establishing that all fuel bundles are receiving flow, it is very improbable that the reactor would be started up while flooded.

2.2 If it is assumed that the reactor is somehow started up while hot but flooded, the reactivity effect of ejecting the water out of the process tubes from nuclear heating would be zero or negative. The most serious possible effects of such operation would be if some of the water were not completely ejected from the downcomer or riser and ran back into the active fuel region. This could cause a reactivity addition, the magnitude of which would depend on the fraction of the core so affected. Chugging operation might occur until one of the flux spikes caused a reactor scram. This type of situation might conceivably occur if all the water were not completely blown out of the process tubes using the correct startup procedure. Even though the reactivity effects of some water running back into the process tubes would probably not be large enough to cause a severe power transient, special attention will be given to design and procedures to minimize the possibility that water could be held up in the risers and then run back into the active region of the core following reactor startup.

3. Errors in Handling Valves

3.1 The safety of the reactor in the event of valve malfunctions, discussed in Section IV. G. 2, is applicable in the event the indicated valve movements occurred as a result of operator errors in handling the valves or controls rather than due to equipment malfunction.

3.2 If an operator were to provide too rapid a flow of steam during preheating of the VSR in preparation for startup, it is possible that significant voids would be formed in the moderator water. Because the reactor has a positive moderator void coefficient in the cold flooded condition, a sudden increase in moderator voids within the core region could add reactivity. As long as the control rods are inserted (and there are many interlocks to assure this), the void formation would not cause the reactor to go critical. However, as an added safeguard, the maximum rate of preheating steam flow will be limited to a value that will not result in fast significant void formation in the moderator. Thus, an error in handling the preheating steam valve could not result in an accident.

3.3 If, during preheating of the reactor, the operator were to open all the steam flow control bias valves and then open the steam outlet main control valve, the process tubes could be unflooded too soon. Preheating would have had to progress above 212°F in order to pressurize the reactor sufficiently to cause unflooding when the valves are opened. Assuming that the reactor is pressurized to about 150 psig (366°F saturated), very conservative calculations indicate that it might take as little as 2 seconds to unflooded the process tubes. If the control rods were fully inserted (and there are many interlocks to assure this), the unflooding would cause a negative reactivity effect. If it is assumed that most of the control rods have somehow been withdrawn, the reactivity addition might be positive. If it were as positive as one dollar total addition (see Figure II. 13. a), then the rate of reactivity addition might be as large as 50 cents per second. If this reactivity addition occurred, the core would be protected adequately against damage by the period and neutron flux scram sensors.

4. Errors in Handling Pumps

4.1 The safety of the reactor in the event of pump malfunctions, discussed in Section IV. G. 1, is applicable in the event the indicated pump failures occurred as a result of operator errors in handling the pumps or controls rather than due to pump malfunctions.

4.2 The cold water accident discussed in Section IV. J represents the most severe error that could be made in handling the various pumps in the plant.

5. Rod Withdrawal Error at Power

5.1 The worst error of this type involves withdrawal of the strongest single control rod at the maximum shim speed. The effects of this error on the reactor are described in detail in Section IV. F. 5.

5.2 If the operator were to withdraw control rods in the wrong pattern for power operation, an undesirable distortion of the power distribution could occur without exceeding the rated power neutron flux limits. This could be done only slowly, however, and the bundle outlet steam temperature alarms would warn the operator that the power was too high in some bundles.

6. Refueling Errors

6.1 The reactor is designed to have a minimum shutdown reactivity margin of about 4%. It is therefore very improbable that a large enough calculational error could be made to result in overloading of the core. If the calculational error were made, the operators should notice the close approach to criticality and take corrective action. If sufficient fuel were added to exceed criticality, the reactor would be adequately protected by the period and flux scram circuitry which would scram the control rod which is kept "cocked" during refueling operations.

6.2 One way in which the reactor might achieve accidental criticality during fuel insertion would be for the operators to cock too many control rods. In the VSR design, however, an interlock is provided so that only one of the twelve control rods can be cocked during refueling. The control rod system is designed so that the reactor will remain shutdown by about 1% even if the strongest control rod is fully withdrawn. Additional protection is afforded during the refueling operation by an interlock which runs all control rods in if the flux amplifier range switches are not set down to about 10^{-3} rated power or less.

6.3 A detailed analysis of the "refueling accident" is presented in Section IV. J. The addition of reactivity at the maximum rate permitted by the refueling hoist is considered to be representative of the most severe kinds of reactivity accidents that could occur to the reactor due to errors in the refueling operation. The case of a fuel bundle being dropped into the core is hypothesized, without regard to credibility, and the results of the analysis presented also in Section IV. J, in order to assess the effect of the positive reactivity coefficients of the reactor in the event of an extremely fast reactivity transient.

7. Failure to Replenish Water in Emergency Cooling System

7.1 The emergency cooling system is designed to operate for an hour without any attention. If the operator should fail to add water to the secondary side of the emergency heat exchanger during the first hour of operation, automatic makeup is provided in order to maintain a minimum level on the secondary side of the exchanger. The minimum capacity which is thus maintained automatically is sufficient to maintain the reactor pressure below the safety valve settings. Thus, failure to attend to the emergency cooling system would not introduce a hazard to the plant.

7.2 If the plant is being started up and the emergency cooling heat exchanger level is not up to the normal value, it will not be possible to withdraw control rods until the situation is corrected. If the water level in the emergency cooling steam dump tank is too high or too low, or if there is high pressure in the dump tank, control rod withdrawal is prevented also. If the reactor is operating and any of these four conditions occurs in the emergency cooling system, the control rods will all be run in. Thus, the emergency cooling system is adequately protected against errors or malfunctions that would tend to compromise its ability to provide the required minimum steam flows if the reactor should suddenly be isolated from its normal heat sink.

8. Failure to Install Superheated Steam Couplings

8.1 In order to remove fuel bundles from the core, it is necessary to disconnect and remove the coupling between the fuel bundle riser outlet and the steam line which carries the steam out the individual vessel steam nozzles. These couplings must be reinstalled before again sealing the reactor vessel. This operation will be carefully checked to assure proper, resonably leak-tight, coupling and to minimize the possibility of leaving one of the couplings off.

8.2 If a coupling were to be left off, the principal effects would be to the uncoupled fuel bundle. When the water is blown out of the process tubes, it is unlikely that the uncoupled bundle would be blown out since moderator pressure would be present in both the downcomer and riser sections. When steam flow is established prior to nuclear heating, the bundle would receive no flow, but saturated steam would flow out the steam line. At this time, the operator might detect the problem if he observed that the bias valve setting did not match the normal flow through the bundle (or group of bundles with which it might be headered upstream of the bias valve). When the control rods are withdrawn and power brought up to about 5% of rated, the water left in the uncoupled bundle would begin to boil. The operator is required to check all bundle outlet temperatures before raising power above 5%. At this time it would be detected that there was no temperature rise in the steam line for the uncoupled bundle. Also, any bundles sharing the bias valve involved would probably be somewhat starved for flow, due to the abnormally high fraction of flow through the uncoupled steam line, and would indicate excessive superheating. The reactor would then be shut down and the situation corrected.

8.3 If, for some reason, the uncoupled bundle were not discovered by the procedure designed to do so during startup, the bundle would boil dry in about half an hour. At 5% power with no steam flow, calculations indicate that the cladding would exceed normally acceptable temperatures but would not melt. Hence, no release of fission products would be expected unless power were increased above 5%. If power is increased, cladding melting will begin and some fission gases will be released. A very small natural convection head of a few inches of water will exist to provide a small steam flow through the fuel bundle. This will be sufficient to carry some of the released fission gases into the vessel steam space and into the main steam flow through the other bundles. The activity rise will be observed at the stack monitors or, if a significant release occurs, by the steam line activity monitors. The latter monitors would scram the reactor, initiate emergency cooling, and isolate the main steam line.

8.4 In the event a coupling is incorrectly installed, or develops a leak for some other reason, the detection methods would be similar to those indicated above. If the leak were not sufficient to permit detection, it is very improbable that it could result in cladding failure in the affected bundle.

J. ACCIDENT ANALYSES

1. Startup Accident

1.1 The reactor is started up in the hot, pressurized, unflooded condition. Under these conditions, the reactivity worth of the strongest control rod is about 3.2%. With a cosine squared reactivity worth versus rod position, a peak to average reactivity gradient of 1.57 results. For a stroke length of 48 inches and a withdrawal rate of 20 inches per minute, the maximum rate of reactivity addition would be less than 5¢/second.

1.2 The startup accident that was analyzed assumed that the reactor was initially hot and subcritical at a shutdown power level of about 10^{10} times below the high flux scram setting of 15.6 Mwt (125% of a rated power level of 12.5 Mwt). Through error, the operator withdraws control rods as fast as they can be withdrawn, and persists in this error until the reactor is shut down automatically. The maximum reactivity addition rate of 5¢/second is assumed to apply continuously throughout the course of the accident. Either a period scram or a flux scram while power is still low would terminate the accident without any appreciable power generation or fuel temperature rise. However, for this analysis, it was assumed that the reactor scram did not occur until the maximum high flux scram setting of 15.6 Mw was reached. This is a particularly pessimistic assumption since the design and interlocking of the neutron flux micromicroammeters is such as to require that in the intermediate and power ranges, the scram point never be further than a factor of about 20 above the existing flux level whenever the control rods are being withdrawn.

1.3 The maximum delay times and minimum scram speeds for the control rods are as tabulated below. These values were used in all the analyses presented in Section IV.J of this report.

Time, seconds	Inserted Rod Positions, % of Initially Withdrawn Distance
0	0 (scram signal)
0.2	0 (delay time until rod motion begins)
0.6	10
1.0	50
-	50 to 100 assumed to occur at normal rod run-in speeds

1.4 The startup accident analysis showed that even if the negative Doppler reactivity effect is neglected, shutdown by scramming the control rods from the high flux signal would result in the generation of less than 40 Mw-seconds of thermal energy. The maximum fuel temperature rise in the hottest part of the core would not exceed 400°F. The cladding temperatures would remain well below the rated power values.

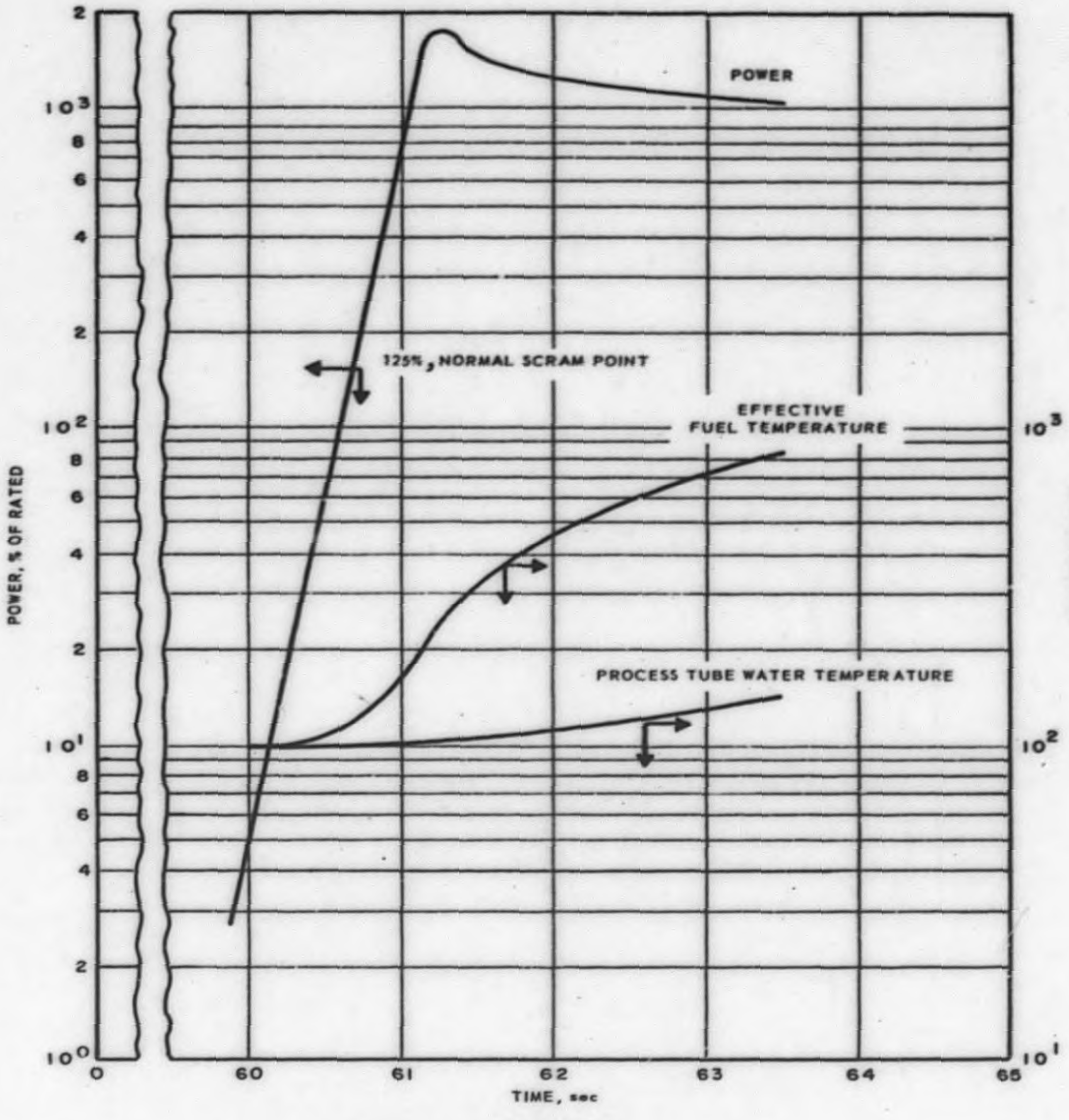


FIGURE IV.5
REFUELING ACCIDENT AT MAXIMUM HOIST RATE
SCRAM NOT SHOWN

2. **Refueling Accident**

2.1 The possibilities for refueling errors are discussed in Section IV. H. 6. The refueling accident summarized below is considered representative of the worst such accidents that could occur to the reactor due to refueling errors. The protection against occurrence of the accident is described elsewhere in Section IV and will not be repeated here.

2.2 For this analysis, it is assumed that the reactor is in the cold, flooded condition, that it is just critical at 10^{-11} rated power with a center fuel bundle removed and the 8 outer control rods withdrawn (near end of core life), and that the center fuel bundle is lowered into the core at the maximum rate permitted by the refueling hoist. It was assumed that the period scram failed and the reactor did not scram from high reactor flux until flux reached 125% of rated.

2.3 The maximum hoist rate is 20 inches per minute and the maximum fuel bundle reactivity worth is 2.8%. The effects of the negative Doppler coefficient, the positive moderator temperature and void coefficients, and the positive unflooding effect, were included in the analysis. The nuclear characteristics of the core with respect to temperature increases and unflooding, for the assumed initial core condition, may be seen on Figure II. 13. c.

2.4 The analysis was made with the aid of a digital computer. Figure IV. 5 shows the power trace and the increases in effective fuel temperature and process tube water temperature with time. The moderator water temperature rise was calculated also and the resulting positive moderator temperature reactivity increase included as the transient progressed. For simplicity, the computer calculation did not include the scram effects. The turnover in power shown on Figure IV. 5 is due to the Doppler effect, which would be effective before scram. The transient is slow enough that scram can be factored into the results. With scram, the peak effective fuel temperature would be about 700°F and the peak hot spot fuel temperature about 1500°F. The water within the process tubes would be heated only a few degrees by the time scram is effective, so there is no possibility of void formation or fast unflooding of the process tubes. The analysis shows that in the hypothesized accident the reactor would be shut down safely and without any fuel damage.

2.5 The "effective" temperature shown on Figure IV. 5 and mentioned in the paragraph above involves a concept which is used throughout the accident analyses in this report. In the calculation of fast power transients in which the Doppler coefficient plays an important role, the calculation is simplified by the use of an "effective" fuel temperature. The effective temperature of the core is that temperature which would produce a Doppler reactivity that is identical to what would be produced by integrating to account for local temperature (power) variations in the core volume. The effective temperature is obtained by integrating the power squared weighted temperature over the core volume. The correct Doppler coefficient to use in conjunction with the effective temperature is that shown on Figure II. 14. The effective temperature rise of the Mark I core is related to the peak temperature rise by the factor 2.3.

2.6 In the cold condition, the maximum control rod worth is approximately the same as the reactivity worth of the fuel bundle used in this refueling accident analysis. The hoist speed and rod withdrawal rates are about the same also. Therefore, if it were assumed that all the failures necessary to permit a startup accident in the cold condition could occur, the course of the accident would be essentially the same as calculated for the refueling accident.

3. Rod Withdrawal at Power

3.1 The analysis of the worst accident of this type is presented in detail in Section IV. F. 5.

4. Cold Unflooding Accidents

4.1 The refueling accident analysis presented in Section IV. J. 2, above, showed that none of the positive water-density-dependent reactivity effects of the reactor in the cold condition were detrimental to reactor safety for that type and degree of accident. To better assess the effect on safety of the cold reactivity effects of the design, two additional accidents were analyzed. As in the case of the refueling accident, and in accordance with the safeguard design criteria in Section IV. D. 1, these accidents also assume failure of the many interlocks designed to prevent the reactor from being made critical in the cold condition. Due to its potentially greater possibility for causing core damage, the positive unflooding reactivity effect is of particular interest in these analyses.

4.2 For the first case, it is assumed that the core is cold and flooded, is near the end of life, and has been brought critical and to 1% of rated power by withdrawal of the 8 peripheral control rods. The nuclear characteristics for this configuration are shown on Figure II. 13. c. The case with all control rods withdrawn, shown on Figure II. 13.a, would be more severe but is not a real possibility since even near the end of core life some control rods will be inserted when the reactor is cold and just critical. A reactivity step of 10¢ is inserted arbitrarily to start the reactor heating from an assumed starting point of 100°F.

4.3 A digital computer was used to calculate the transient response curves of reactor power, effective fuel temperature, and bulk water temperature shown on Figure IV. 6. The initial power transient is due to the 10¢ step. The remainder of the curve is influenced by the negative Doppler fuel temperature coefficient and the positive moderator temperature coefficient. The calculations were carried to the point where bulk moderator boiling would occur, which would control the transient due to the small but negative moderator void coefficient at 212°F. Figure IV. 6 shows that the transient is very slow. A high flux scram within a decade or so of the initial power would control the situation without any possibility of damage. The operators would have four minutes in which to observe the situation and shut the reactor down. From Figure IV. 6, it even appears that a flux scram at 125% of rated power would control the reactor safely, but this might not be the case. In this analysis, the temperature of the water within the process tube was not calculated separately from the moderator water. If it is assumed that all of the heat produced by the fuel heats the 15% of the water within the process tubes, rather than all of the water as shown on Figure IV. 6, then an additional reactivity addition might occur from fast unflooding of the water in the process tube before the 125% flux scram could become effective.

FIG. IV. 6

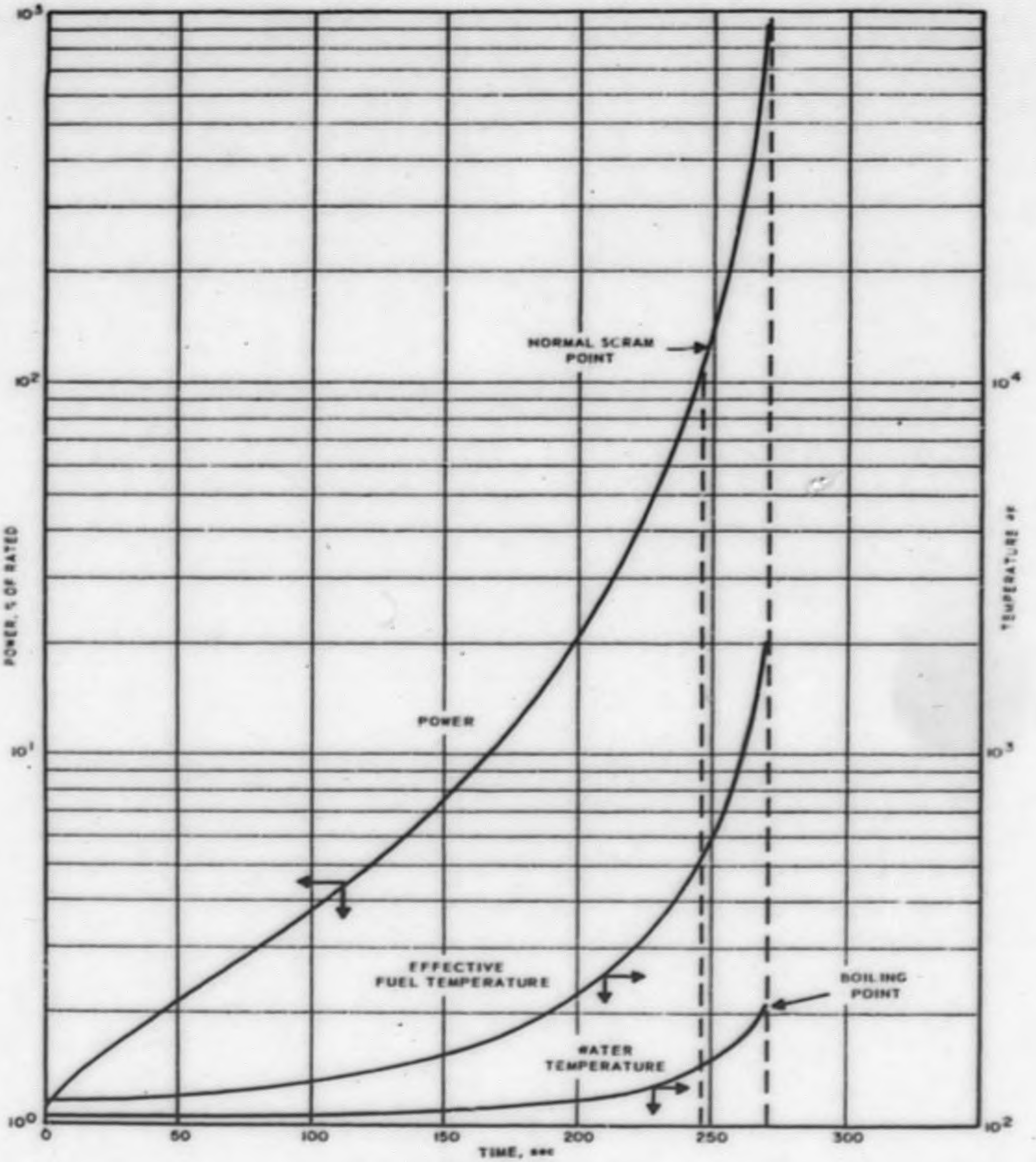


FIGURE IV.6
NUCLEAR HEATUP ACCIDENT
SCRAM NOT SHOWN

4.4 The total reactivity that could be added by unflooding all the process tubes, with the moderator water still at 100°F, would be about \$1.40. The rate of power rise is relatively slow in the postulated accident, and the water in different process tubes would reach the boiling point at different times. Thus, unflooding probably would be due to a combination of boiling and ejection of water, creating more of a ramp than a step reactivity addition. It appears plausible that an early flux spike might scram the reactor before the unflooding could progress significantly. However, if the \$1.40 were added essentially as a step function, calculations show that in the resulting short period excursion the Doppler coefficient would limit the rise in effective fuel temperature to about 2000°F. Peak UO_2 temperatures of 4500°F might be reached but no fuel would melt. Some cladding would be melted or damaged since the water would have been ejected and no steam flow would be available to cool the cladding. Some fission product gases and volatile solids would be released, but all these would be retained within the reactor vessel and would not create a radiological hazard. While the core damage would be extensive if the accident were carried to the extremes calculated here, it is judged that for nuclear heatup type accidents, the cold positive reactivity coefficients of the Mark I core do not detract significantly from the overall safety of the reactor.

4.5 The other accident which was analyzed, without regard to its credibility, was the case of a refueling accident in which the fuel bundle is dropped into the core to create an extremely fast reactivity and power transient. The starting assumptions were the same as for the refueling accident presented in Section IV.J.2, except that in this case it was assumed that when the fuel had been lowered just to the entrance of the fuel channel that it was dropped into the core. The analysis assumed minimum resistance during the bundle drop, permitting full insertion within 0.6 second. The analysis indicates that in the course of the ensuing transient, the prompt heating in the moderator might cause the further addition of about \$1.00 in reactivity from the positive moderator temperature coefficient. However, even without this addition, the fuel bundle worth of 2.8% is too large to be controlled by the Doppler coefficient. Since the high flux scram could not be effective soon enough, the shutdown mechanism would be destruction of a portion of the core. Because of the delay times associated with ejection of water in this fast transient, the core would be shut down before any further reactivity could be added by fast unflooding of the process tubes. The conclusion reached from the analysis of this very pessimistically conceived accident is that the positive reactivity coefficients of the core would not significantly alter the course or effects of such an accident. The results would be essentially the same if all reactivity coefficients were negative under the assumed conditions of the accident.

5. Rod Drop-Out Accidents

5.1 The control rod worths have been minimized by designing the core for low burnup and using stainless steel control blades without supplemental boron. This has made it possible to limit the average control rod worth to about 1.3%. However, in the most adverse rod pattern, in the hot condition, the worth of the strongest rod is about 3.2%. As indicated by Section IV.J.1, the rod speeds are limited so that the reactor will be safe even if the strongest rod is withdrawn at its maximum rate when the reactor is critical. Section IV.G.4 discussed the safety of the reactor in the event of stuck control rods and in the event of stuck control rods separated from their drive shafts inside the reactor. Because of the substantial failures that would have to occur, and the procedural safeguards to minimize individual rod worths and promote prompt detection of stuck control blades, the occurrence of a severe rod drop-out accident is considered very improbable. The remainder of this section describes the course and effects of the worst potential rod drop-out accidents without particular regard to the plausibility of the sequences of failures and errors that would be required for the accidents to occur.

5.2 Three cases of control rod drop-out in the hot condition were analyzed. For each case it was assumed that the reactor was initially hot and the core unflooded. Case A assumed that the reactor was operating at 100% power and 100% steam flow, with scram set at 125% rated flux. Case B assumed the reactor was at 5% power and 25% or greater steam flow, with the scram set at 125% rated flux. Case C assumed that the reactor was just critical at 10^{-11} rated power, with 25% or greater steam flow, and scram set at 125% rated flux (period and intermediate flux scrams assumed unavailable). For each case, it was assumed that the control pattern was such as to maximize the worth of a fully inserted rod that was stuck in the core and separated from its fully withdrawn drive shaft. This maximum worth rod (3.2% reactivity worth) then became unstuck and fell out of the core under essentially free-fall conditions (about 0.56 second to fall out completely). The analysis was made with the aid of a digital computer. Scram effects, not calculated by the computer code, were included by hand calculation and are factored into some of the results presented.

5.3 Figure IV.7 shows the power traces and effective fuel temperatures for each of the three cases analyzed. The effects of scram are not shown, but would have no effect on the power peaks attained. The power transient is controlled by the Doppler coefficient and the moderator void coefficient. The void coefficient is negative in the hot condition and is effective in the fast excursion principally due to prompt neutron and gamma heating directly in the moderator. Figure IV.8 shows, for each case, the percentage of the core which exceeds given temperatures. Figure IV.8 includes the effect of scram at 125% of rated flux. Case A, which started at 100% power, is the least severe case. Even though the effective temperature for Case A, shown on Figure IV.7, is higher than for the other cases (due to initial power level) the effective temperature rise is less, the peak temperature rise is less, and the peak temperatures attained are substantially less than for the other cases. For Case A, the peak UO_2 temperature is about 4200°F. Thus, no fuel melting will occur and no cladding melting or fission product release should occur. The steam flow available protects the cladding from damage.

FIG. IV. 7

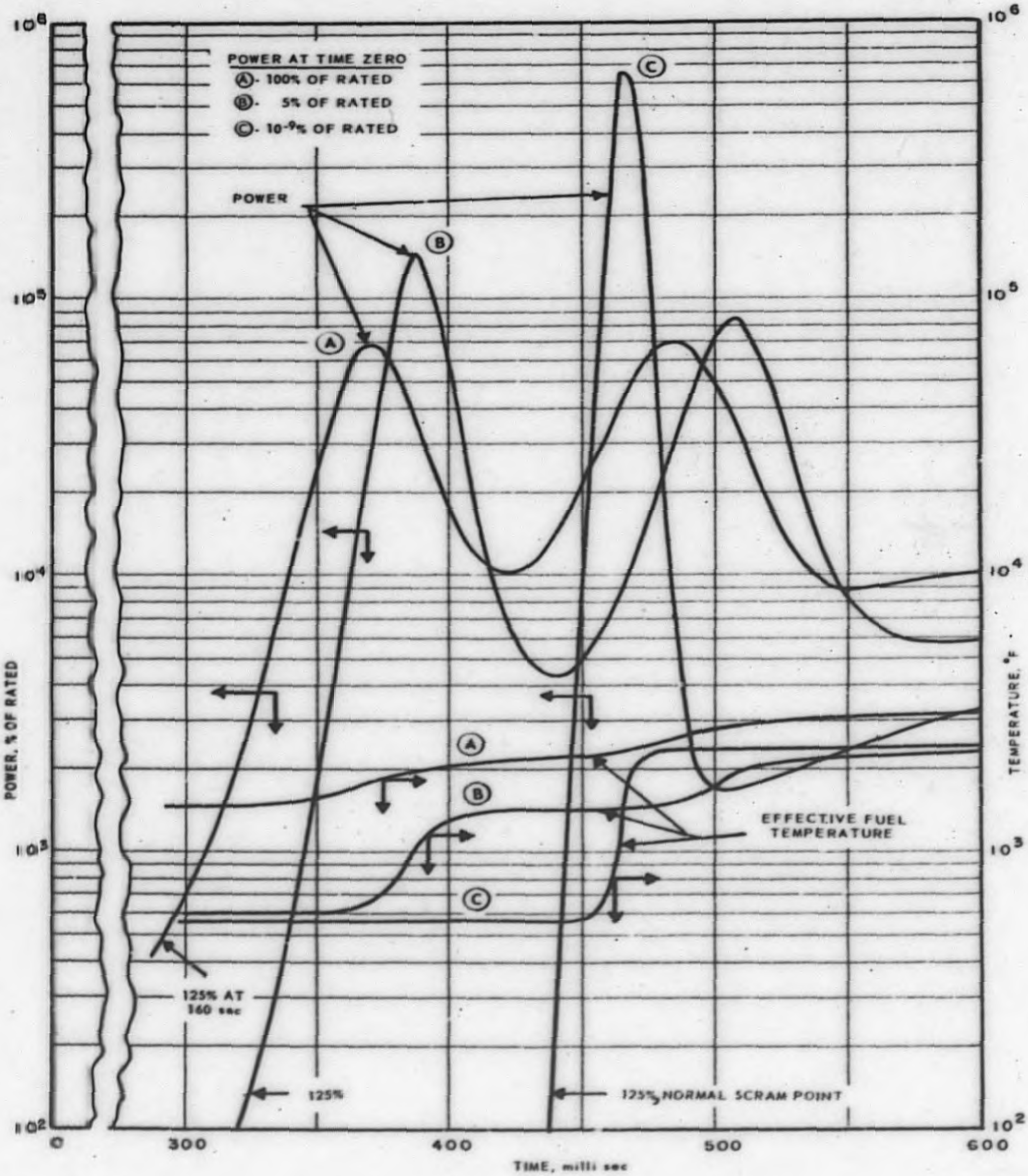


FIGURE IV.7
ROD DROPOUT ACCIDENT POWER TRACES
SCRAM NOT SHOWN

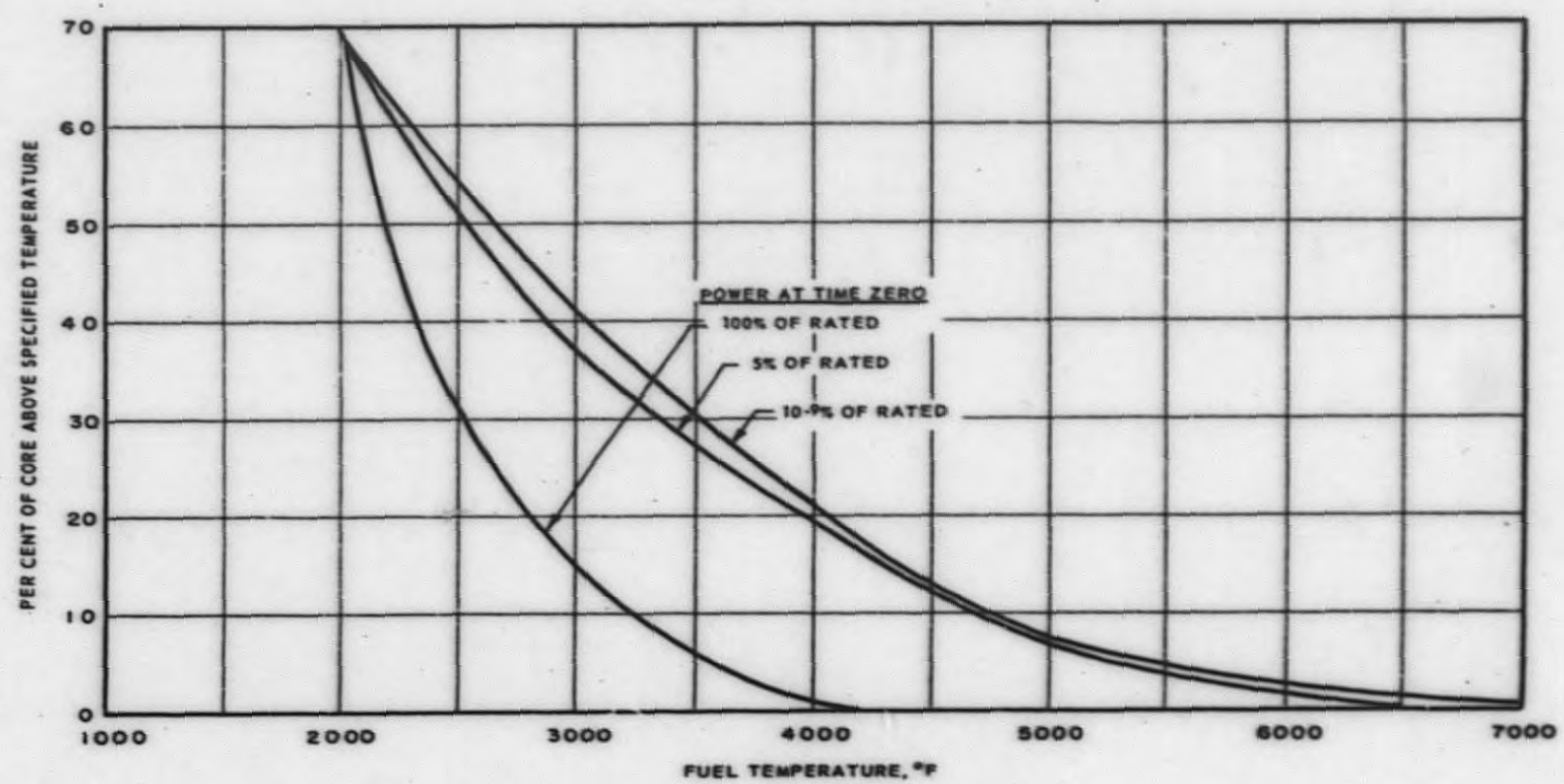


FIGURE IV-8
ROD DROPOUT ACCIDENT FUEL DAMAGE

FIG. IV-8

5.4 Cases B and C, starting at 5% and 10^{-11} rated power respectively, show essentially the same degree of core damage. Figure IV.8 shows that fuel temperatures in excess of the melting point of 5000°F would be exceeded in about 7% of the core volume. Smaller fractions of the core exceed calculated temperatures in the 6000 - 7000°F range, which indicates that fuel melting would occur (the calculational model does not recognize the change of state or energy required for the change of state) in some of the fuel but that fuel vaporization would not occur. Some cladding rupture and fission product release might occur from internal gas pressure in a small fraction of the fuel. The degree of cladding melting or damage will be dependent on the steam flow available. This phase of the accident, and the amount of fission product release, will be evaluated in more detail as the design progresses. It is apparent, however, that the radiological effects of the release would be less severe than those associated with the "maximum credible accident" to the reactor. Any fission products released would be retained automatically within the reactor and steam systems.

5.5 If it is assumed that a rod drop-out accident occurs when the reactor is in the cold, flooded condition, the results would be essentially the same as indicated for the drop-in of a fuel bundle described in Section IV.J.4.5.

6. Cold Water Accident

6.1 The only source of cold water to the reactor is the feedwater flow, which enters through a circular sparger above the core to make up for moderator loss from boiling. The feedwater is normally at a temperature of about 130°F and flows at 13,500 pounds per hour when the reactor is operated at 12.5 Mwt. The feedwater mixes with the recirculating moderator downflow outside of the core shroud and flows to the bottom of the vessel before entering the egg-crate structure and then the flow channels at the bottom of the core. The makeup feedwater flow rate is controlled by the reactor level controller and may range up to a maximum flow of about 30,000 pounds per hour.

6.2 To prevent a sudden injection of cold water into the reactor, and consequent collapse of moderator steam voids and addition of reactivity, an interlock is provided to prevent an operator from starting the feedwater pump unless the feedwater valve to the reactor is closed. In addition, the reactor is shut down automatically by control rod run-in whenever the feedwater pump is tripped, i.e., the reactor cannot be operated unless the feedwater pump is running. Thus, the worst cold water accident which might occur would be due to opening of the feedwater valve at the maximum rate.

6.3 The accident analyzed assumed that the reactor was operating at rated power with a low feedwater flow rate of 3,000 gpm, a moderator recirculation rate of 1×10^6 pounds per hour, feedwater at 100°F, average void fraction of 5% by volume, and a void coefficient of -1.5×10^{-3} $\Delta k/k/\%$ increase in average void volume. These assumptions are expected to produce a substantially more severe situation than the normal rated values of these items presented elsewhere in this report. The incident is assumed to be initiated by failure of the level controller and opening of the feedwater valve at its maximum rate of one minute for full stroke, which changes feedwater flow by about 1000 pounds per hour per second. Each 1000 pound per hour increase in feedwater rate reduces subcooling by about 0.5 Btu/pound, reduces the void volume by about 1.5%, and increases reactivity by about 0.2% (30¢)

6.4 Considering the accident only as a reactivity transient at 30¢ per second, calculations indicate that the high flux scram at 125% of rated power will limit the maximum fuel temperature rise to about 200°F and the maximum cladding temperature rise to less than 100°F. No fuel damage will occur from this accident.

7. <u>Core Flooding Accidents</u>

7.1 In the hot operating condition, the positive reactivity effect associated with flooding the superheat steam process tubes makes it advisable to evaluate plausible causes of flooding to assure that these positive reactivity effects do not detract significantly from the overall reactor safety. Three cases have been investigated.

7.2 One possible cause of process tube flooding is failure of a process tube due to corrosion or cracking. The moderator pressure will be slightly higher than the pressure inside the process tubes, so any break below the moderator level will result in water being forced into the crack and out with the steam flow. If the crack were large enough, the drop in steam bundle outlet temperature would be observed or alarmed. Since the core contains 288 process tubes, and the maximum reactivity associated with flooding all of them is 0.7% ($1.10), the reactivity addition from complete instantaneous flooding of one tube would have negligible effects on the reactor.

7.3 Flooding of all of the process tubes takes place by raising the moderator level until it covers the saturated steam inlet to the fuel bundles. This operation is carried out normally in preparation for refueling the reactor. Interlocks are provided to shut the reactor down by automatic control rod run-in if a high moderator level is reached (22.5-foot level). The same signal closes the feedwater make-up valve, and the make-up valve cannot be opened again unless moderator pressure is down to at least 500 psia (the interlocks serve another purpose also, as indicated in Section IV.D.7) At this low pressure, control rods will be held in also by a scram signal. Also, at the maximum feedwater rate of about 30,000 pounds per hour, the level rises only about three inches per minute. Since it is a minimum of about two feet from the 22.5-foot level to the level at which flooding begins, the operator has several minutes to respond to the alarms and the gradual rise in level before any flooding could occur. These many design features provide very good assurance that the reactor will never be flooded when it is in the hot operating condition. In accordance with the safeguard criteria in Section IV.D.1.4, an analysis was performed assuming that these interlocks all failed and the core was flooded during full power operation from a rise in moderator level.

7.4 To maximize the positive flooding reactivity, the control rods were assumed to be in a "ganged" pattern. This is a very unlikely pattern for power operation and would be avoided procedurally. The maximum flooding reactivity for this pattern is about +$1.10 as indicated on Table II.8. At the maximum rate of moderator level rise, it takes about one second to supply enough water to flood a fuel bundle. It has not been determined whether the flooding levels of the bundles will be at different heights or not, but the worst case from the reactivity addition standpoint would be for the center bundle to be flooded first. Having all bundles flood at the same time would be a less severe case since it would take 32 seconds to supply enough water to flood all of them, and the reactivity ramp would be reduced to the effect of flooding the "average" bundle. For the case of flooding the center fuel bundle in one second, and using a maximum-to-average reactivity weighting of 2 for this bundle, the maximum rate of reactivity addition would be about 7¢/second. From the "rod withdrawal at power" analysis of 5¢/second ramp (Section IV.J.3 and IV.F.5) and the "cold water accident" analysis of a 30¢/second ramp (Section IV.J.6), the reactivity effects of a 7¢/second addition may be seen to be quite safe if the reactor scrams on a high flux signal at 125% of rated power.

7.5 Thermal shock to the fuel cladding is the more severe aspect of inadvertent core flooding. Fortunately, however, the steam quality in the reactor vessel varies from about 100% at three to four feet above the steam-water interface to about 2% at the interface. Therefore, as the moderator level rises, the steam entering the process tube will gradually become wetter and wetter. This will cause a gradual reduction in outlet steam temperatures and cladding temperatures. The low steam temperature alarms will provide another warning to the operator. The reduced cladding temperature will decrease the thermal shock to the cladding if flooding does occur. It is conceivable that the thermal shock could cause small fissures to form in the cladding, as described in Section IV.G.5, but no significant release of fission product radioactivity would be expected to result from inadvertent flooding.

7.6 In the foregoing discussion and analysis of core flooding resulting from an inadvertent increase in moderator level, some rather significant factors which would tend to mitigate the effects of the accident have been neglected. These factors will exist, but are difficult to apply quantitatively in the analysis. In the case where all steam inlets may be at the same height, and about a half minute is required to supply enough water to fill all the process tube, if water would readily enter the tubes, much of it would also readily pass through the fuel and out the steam line. Much of the water would evaporate also, from both flashing due to the pressure drop and boiling upon contact with the hot fuel cladding. It would necessarily take longer than a half minute to flood the core. Furthermore, there is the inherent difficulty of making large amounts of water enter the hot annuli at the entrance to the active core. Calculations indicate that due to the heat stored in the fuel alone (neglecting continued or increased heat production), a "slug" of water one foot long in the outer annulus would completely evaporate in about 0.3 second with nucleate boiling on the surface. A volume expansion of 20 to 1 would occur. This would tend to accelerate the flow in the forward direction and, in addition, would expel steam backwards against further incoming flow. The ensuing flow pulsations would significantly delay the actual flooding, requiring at least 2 minutes to dissipate just the stored heat in the fuel. From the reactivity addition viewpoint, the inherent tendency of the hot or operating reactor to eject water from the process tubes (negative reactivity if flooding causes positive reactivity) provides an inherent safety mechanism to slow and reverse the reactivity effects of core flooding.

7.7 The last hot flooding case considered was the sudden collapse of process tubes against the outer fuel cladding, to essentially "flood" the former steam space with moderator water. The maximum reactivity effect of collapsing a single process tube would be negligible (less than +1¢), as in the case of flooding a single process tube described in paragraph 7.2 of this section. The only way to cause a significant sudden reactivity addition would be to collapse essentially all of the process tubes instantaneously. This could be done only by imposing a large pressure differential across the process tubes. There is no way to increase the moderator and steam dome pressure fast enough to cause any significant additional pressure across the process tubes, e.g., the incoming steam rate is limited, the large steam space provides a cushion against sudden pressure changes, and the fuel cannot provide a sudden increase because of the low thermal conductivity of the UO_2 and the insulating effect of the steam space and process tube between the fuel and moderator. The other way to impose a pressure differential is to decrease the pressure downstream of the fuel bundle. The worst possible case would be breaking the bundle steam line just outside the reactor vessel. The results of a study of this line break, discussed in Section IV. G. 8. 8, show that the process tubes have a safety factor of at least 2 against collapse under the worst conditions. In order to collapse many process tubes, it would be necessary to break the main steam outlet line. This break would impose a much smaller pressure differential across any process tube wall, so that the safety factor against collapse is even larger when many process tubes are potentially subject to collapse. Since no conceivable way is known in which a major fraction of the process tubes could be collapsed simultaneously, it is concluded that the reactor is adequately protected against positive reactivity additions from this cause.

8. <u>Loss of Steam Flow</u>

8.1 The safety of the reactor in the event of loss of inlet steam flow due to inlet valve closures or failures, or scram of VBWR, has been discussed in Section IV. F and IV. G. Loss of outlet steam flow has been discussed similarly. The most severe case of loss of steam flow to the reactor would be due to closure of the main steam outlet control valve when all steam flow is going to the VSR condenser. The control valve is a faster moving valve than the outlet isolation valves (2-second full stroke cf. about 10-second full stroke for isolation valve) and is not equipped with a reactor shutdown signal on valve closure as are the isolation valves. The case analyzed assumed that the control valve was initially about 60% open (corresponding to steam flow for 12.5 Mwt operation) and closed, due to a sudden failure, in about 1.2 seconds. The steam flow through the valve was assumed to decrease linearly with time.

8.2 In order to maximize the vessel pressure rise and positive pressure reactivity effects, and to delay the low inlet flow signal, it was assumed that the inlet control valve did not respond to the vessel pressure rise signal to throttle inlet flow. The analysis indicates that as the control valve starts to close and the outlet flow starts to drop, the reactor pressure and reactivity will increase. The pressure rises to a peak about 30 psi above the initial values and, at worst, all moderator voids will collapse and add 80¢ in reactivity. The neutron flux will rise and cause scram at 125% rated flux, and will peak at about 10 times rated flux. Emergency cooling is initiated after about 1.1 seconds, when inlet steam flow has decayed sufficiently to trip the flux/flow ratio signal (the flux signal has a built-in time constant or delay so that the fast rise in flux will not trip the flux/flow ratio emergency cooling signal). The peak fuel cladding temperature is calculated to be about 1750°F and the hot spot UO_2 temperature about 3000°F in the course of the accident. No UO_2 melting would occur and no serious cladding damage would be expected. However, it is considered desirable to limit peak cladding temperatures to about 1500°F until further fuel development has better defined the tolerable limits. Therefore, consideration will be given to improving the response of the emergency cooling trip for this situation. The principal delay is caused by the lag of about 1 second between a change in outlet steam flow and the reflection of this change in inlet steam flow. Using an outlet steam flow signal, the reactor would scram from high flux/flow ratio about 0.25 second after the transient started, the peak pressure rise would be 15 psi, and the peak cladding temperature would be 1450°F.

9. <u>Sizing Reactor Safety Valves</u>

9.1 The reactor safety valve minimum capacities are selected on the basis of the hypothetical accidents indicated in the safeguard design criteria (Section IV.D.5). There is no plausible sequence of events which could cause these accidents, but they are chosen since it is then apparent that the safety valves will be adequate for any plausible accident. Since the purpose of the safety valves is primarily to protect the reactor vessel from damage, the effects of the hypothesized accidents on the core has not been analyzed in detail.

9.2 For sizing the main safety valves, it is assumed that the reactor inlet and outlet steam lines are isolated, that the feedwater has stopped, and that the reactor is operating at the relief valve pressure with all heat produced going into the formation of saturated steam. Assuming the reactor is operating initially at a rated power of about 23 Mwt, and that this is increased to a steady-state power of 32 Mwt (140% of 23 Mwt) when steam is discharging through the relief valves, the maximum steam production at 1250 psig would be 180,000 pounds per hour if the moderator has lost all subcooling. The safety valves have been sized for a minimum of 200,000 pounds per hour, as indicated in Section II.F.2. The 140% power margin is considered more than adequate to compensate for the effects of the positive pressure coefficient since the reactivity in voids at 140% power and 1250 psig is greater than under rated conditions even if subcooling is not lost at the higher power. Also, operating the fuel at 140% of rated would result in at least a 25¢ reduction in reactivity due to the negative fuel temperature Doppler coefficient.

9.3 The additional safety valve located on the superheat steam outlet piping is of the electromatic type, and is sized on the basis of passing all the steam which could be produced by normal moderator boiling at 140% of a rated power of 23 Mwt. Reactor pressure is assumed to be at the relief pressure of about 1200 psig. The 140% power margin is based on the same considerations as in the paragraph above. The heat which does not go into moderator boiling must be stored in the fuel or go into superheating the moderator steam produced. Assuming the same ratio of moderator power to total power as for the Mark I core, and assuming feedwater flow and subcooling are maintained, the steam flow produced by the moderator would be about 40,000 pounds per hour. The safety valves have been sized for a capacity in excess of 55,000 pounds per hour as indicated in Section II. F. 2.

10. <u>Reactivity Step</u>

10.1 Although there is no mechanism to cause a large instantaneous reactivity addition to the reactor when it is operating at rated power, the reactor response to such a case has been analyzed and found to be quite safe for additions as large as 92¢. The calculated effects of this hypothetical accident cover any number of lesser accidents which might by hypothesized. Such an analysis is often useful in setting limits on the reactivity worth of reactor experiments which might add reactivity if they failed during operation.

10.2 The analysis of a 92¢ step reactivity addition, with the reactor operating initially at 100% power and 100% steam flow, showed that the Doppler effect would respond before the scram initiated at 125% rated flux, to limit the power peak to about 12 times rated. In less than a second, power would be back down to 100%. The peak temperature rise of the inner cladding would be less than 200°F, and the peak UO_2 temperature rise would be less than 500°F. Thus, no UO_2 melting would occur and the peak cladding temperature would be within the permissible limits.

K. SAFETY AGAINST FIRES AND OTHER GENERAL HAZARDS

1. Fire

1.1 In the event of a fire in the containment building, damage would be restricted to insulating materials and paint. In no event could damage to wiring result in a reactivity increase, and in almost all cases the reactor would be shutdown. A fire in the control building would affect the reactor similarly. A fire in the gas-fired boiler, VBWR, or the dump condenser building could effect the reactor in the same way as equipment failures at those locations. The fire protection system is described in Section II. K. 4.

2. Flood

2.1 The possibility of a dangerous flood does not exist at the site of this reactor because of its location on a broad hillside.

3. Earthquake

3.1 The reactor containment vessel has been designed to withstand earthquake forces of the magnitude appropriate to this area. In addition, seismographic detectors have been provided to scram the reactor in the event of a disturbance.

4. Weather

4.1 The reactor containment building has been designed to withstand any anticipated wind loads or storms. Since over half of the containment shell and almost all of the interior weight is below grade level, tipping of the vessel is inconceivable.

5. Strike or Riot

5.1 Precautions against strike or riot damage include a fence around the reactor area and a perimeter fence around the Laboratory site. Guards are provided at the main gate in the perimeter fencing. Other gates in the perimeter fencing are kept locked and the keys are subject to administrative control. Severing of electrical cables to the plant will result in shutdown of the reactor. A keylock is provided to prevent unauthorized operation of the control rods.

L. THE "MAXIMUM CREDIBLE ACCIDENT"

1. General

1.1 The term "maximum credible accident" is commonly used to depict the reactor accident which, if it occurred, would cause the most severe environmental effects of any accident which is considered "credible" for the reactor. The safety analyses in Part IV of this report show the high degree of assurance provided by the plant design against any accident that might cause the release of hazardous amounts of radioactive materials. In the few cases where some fuel damage might be conceived to occur from power excursions or flow loss during power operation, any fission product materials released would be retained within the reactor and steam systens, and would not constitute an environmental hazard. The only cases in which a large fraction of the core fission products might be released from the reactor system involve ruptures in the reactor primary system. Since none of the nuclear excursion accidents analyzed could cause a reactor rupture, the most severe case of interest is considered to result from a mechanical failure (Section IV. G. 8). In any case of a system rupture involving a significant release of radioactive materials, the released materials would be contained within the containment building which houses the reactor. Since this building, or vessel, is designed with excellent assurance of retaining substantially all of the radioactive materials which might be released in the event of a system rupture, the environmental hazard is limited to direct radiation from the contained materials plus the radiological effects of leakage of a small fraction of the contained materials from minor imperfections which are assumed may exist in the containment vessel shell.

1.2 Later portions of this section of the report provide detailed information on the important considerations related to the "maximum credible accident", i.e., the containment pressure considerations, the release of fission product materials into the containment vessel, the direct radiation effects, and the effects from fission product leakage into the environment. The general description of the "maximum credible accident" which follows is intended to provide perspective to the later more detailed discussions.

2. General Description of "Maximum Credible Accident"

2.1 The "maximum credible accident" is initiated by a near-instantaneous, complete severance of one of the 14-inch-diameter pipe nozzles near the bottom of the reactor vessel. The reactor is operating at 1000 psia pressure, but is quickly shut down due to void formation or scram from signals of low level, low pressure, or high containment pressure. The safety system signals also initiate closure of all isolation valves. The moderator water is all lost from the reactor vessel within a few seconds, but steam flow will continue from the break for several seconds until the inlet steam flow to the reactor can be stopped. As the moderator water is released from the break, part of it will flash to steam and increase the pressure within the containment vessel. The air initially inside the containment vessel will be heated and will increase in pressure also. Even during the fast release of moderator, some of the steam formed will condense on the enclosure shell. The peak containment pressure will be reached in about 15 seconds and then will begin to drop rapidly as the cool masses inside the containment vessel are heated up.

2.2 The emergency cooling system will be actuated by the safety system signals shortly after the break occurs. Steam will continue to flow through the core for a few seconds, perhaps until the inlet steam flow is stopped. Decay heat will then begin to increase the temperature of the UO_2 fuel and the cladding. If reactor feedwater flow continues, it will run out the bottom break. Without any moderator water to keep the process tubes cool, radiation from the fuel to the process tubes will not prevent the fuel from overheating. Within a few minutes, the cladding will begin to melt, releasing a small percentage of gaseous fission products. When the first fuel material reaches 4000°F, a larger fraction of the gaseous and volatile fission products will be released. When melting of the UO_2 occurs, at 5000°F, some of the less volatile fission product materials will be dispersed also. By the end of an hour, about 40% of the fuel will have melted, and by 24 hours about 90% will be molten.

2.3 As the fuel slowly evolves its fission products, many of the released materials will condense on cooler surfaces within the reactor vessel and in the lower portions of the containment vessel. Those which become dispersed into the containment atmosphere will be subject to washdown by "rain" which occurs as the pressure in the containment is gradually reduced. When the containment spray system is actuated, it will provide additional washdown and pressure reduction effects. The competing mechanisms of fission product dispersion and washdown will result in a net inventory of fission product materials in the containment atmosphere at any time. This amount will increase at first and then will gradually diminish as the rate of dispersion decreases. The inventory with time will determine the direct radiation which personnel might receive while remaining at various distances from the visible portion of the containment vessel for given periods of time.

2.4 The containment pressure at any time will determine the leakage rate. Together with the fission product inventory with time, the fission product leakage rate with time is determined. Since both the fission product dispersion and washdown are selective with respect to the various types of fission product materials, the fission product leakage rate will vary not only in total amount but in the fractions of various types of materials of interest. The amount of each type of material that leaks with time, and the meteorological conditions, determine the radiation exposures which personnel might receive, at various distances and for various periods of exposure, from direct radiation from the passing "cloud", from fallout of materials, or from inhalation of materials which irradiates the critical organs (lungs, thyroid, bone).

3. Containment Vessel Pressure

3.1 The "maximum credible accident" provides the basis for design of the containment vessel with respect to design pressure and leakage rate. The design pressure must equal or exceed the peak pressure which could accompany the "maximum credible accident" (Section IV. D. 9). The containment vessel must be built with the maximum practical degree of leak-tightness and must have a demonstrated leakage rate of not more than is assumed to occur in the event of the "maximum credible accident". The remainder of this section deals with the detailed basis for calculation of the peak accident pressure, the containment vessel design pressure, and the post-accident pressure reduction with time.

3.2 As described in Section L. 2, above, the rise in containment pressure is due primarily to a rapid loss of the reactor moderator water from a nozzle failure near the bottom of the reactor vessel. The steam within the reactor system and the saturated steam which continues to flow until the isolation valves are closed contribute an additional pressure rise. The calculation of the peak accident pressure includes the following assumptions:

3.2.1 At the time of the system rupture, the reactor vessel, core, process tubes, and risers are assumed to be filled with saturated water at 1000 psia up to the 23-foot level. This is above the normal moderator level. above the 22. 5 foot level at which scram and moderator makeup are stopped, and at the level at which the steam inlet isolation valves are closed. No credit is taken for voids. Under the assumed conditions, the reactor system possesses the maximum energy content. The water volume will be about 800 cubic feet, and the mass of steam in the vessel and piping about 650 pounds.

3.2.2 It is assumed that 2500 pounds of saturated steam flows in from the boiler and VBWR. The inlet steam isolation valves will be designed to close rapidly enough to limit the flow to 2500 pounds assuming that the steam line is broken just inside the containment vessel.

3.2.3 An allowance is made for 30 cubic feet of saturated water in recirculation lines which might be added at a later date if forced moderator recirculation were desired.

3.2.4 The containment vessel is assumed initially to be at 100°F, 14. 7 psia, and 100% relative humidity. The free volume is calculated to be at least 142, 000 cubic feet. At the time the peak containment pressure is reached, it is assumed that thermal equilibrium exists between all water phases and the containment air.

3.2.5 No credit is taken for heat losses in calculating the peak pressure. No second order contributions to pressure build-up are included. No energy from a nuclear excursion is included since none of the excursions analyzed could cause a reactor rupture. No energy from a metal-water chemical reaction is included in the peak pressure calculation since the peak pressure will occur and then diminish prior to initial fuel cladding melting.

3.3 Based on the above assumptions, the calculated peak containment vessel pressure is about 52.8 psig and the corresponding temperature about 277°F. The design pressure of the containment vessel has been set about 10% higher, at 58 psig, and the vessel will be tested at 115% of the design pressure.

3.4 The 10% margin in the design pressure is provided principally to permit flexibility in the detailed design of the reactor system and future modifications to the plant. For the purposes of this discussion, the 10% margin may be regarded also as additional protection against phenomena which may be new or different with superheat reactors, and which may not now be as well understood as for better known and more fully developed reactor types. The increase in pressure from 52.8 psig to 58 psig represents an arbitrary addition of about 1.5×10^6 Btu of energy. The margin between the calculated peak accident pressure and the test pressure of the containment vessel represents about 4×10^6 Btu. The heat transfer to the containment shell and concrete during the 15 seconds required to reach the peak pressure provides an additional margin of at least 10^6 Btu. The following paragraphs indicate the degree of additional protection provided by these margins against conceivable sources of energy addition not included in the calculation of peak pressure.

3.5 If all the stainless steel in the fuel cladding and velocity booster tubes were to undergo a metal-steam reaction, the energy released would be about 230,000 Btu. If all the hydrogen produced in this reaction were to burn also about 2×10^6 Btu would be produced. However, as indicated in Section IV.G.5.5, it is doubtful that stainless steel will undergo a rapid or violent reaction with steam. Also, to obtain any significant degree of reaction it would be necessary for the cladding to be molten and very finely divided. As the cladding will not begin melting until after the peak pressure is reached, and as even then there is no apparent mechanism to provide good dispersion of the molten material, it is very unlikely that a metal-steam reaction could influence the peak pressure at all. Even if a metal-steam reaction should occur, it is unlikely that much, if any, of the hydrogen would burn. When the hydrogen is first liberated it would be expected to be mixed with steam, and by the time it was mixed with the air in the containment vessel the average concentration of hydrogen would not exceed the lower burning limit of 4% by volume. Thus, hydrogen release would not be expected to influence the peak pressure and probably would not affect the long-term pressure reduction either.

3.6 Second order heat inputs that might conceivably occur during the 15 seconds required to reach the peak containment pressure are estimated to amount to about 10^6 Btu. Included is transfer of heat stored in the fuel and cladding, heat from the reactor internals, heat from the reactor vessel shell, fuel decay heat, and heat from electrical and miscellaneous items. These second order heat inputs are quite small compared with the internal energy of the containment vessel contents of about 27×10^6 Btu at 58 psig, and would be compensated by the minimum calculated heat transfer to the cooler masses during the first 15 seconds (see paragraph 3.4, above).

3.7 The accident analyses have indicated that a system rupture is not likely to be accompanied by a nuclear excursion. However, if an excursion did occur simultaneously with the rupture, it would not contribute significantly to the peak pressure in the containment vessel. For example, about 5×10^5 Btu would be generated in the hypothetical case of a control rod drop-out when the reactor is operating at 100% power (Section IV. J. 5).

3.8 The pressure response with time following attainment of the peak containment vessel pressure is of interest to assure that the containment pressure never again reaches the peak pressure, and for use in calculating the leakage rate of the containment vessel with time. The assumptions used in calculating the pressure versus time relationship presented on Figure IV. 9 include the following:

3.8.1 The peak containment pressure is increased from 52.8 psig to 58 psig by the arbitrary addition of 1.5×10^6 Btu. This is done in order to evaluate the containment vessel for its full design pressure.

3.8.2 Decay heat from the fuel is added directly to the contents of the containment vessel. Decay heat is based on long-term operation at a steady-state power level of 25 Mwt.

3.8.3 The steam, water, and air in the containment vessel are assumed to be at the same temperatures throughout the period of calculation.

3.8.4 Heat stored in, and transferred to and from, the containment wall occurs only over the area not covered by the concrete liner. Ambient air temperature is 100°F.

3.9 As shown on Figure IV. 9, the containment pressure will drop to about 50 psig in 100 seconds and to 42 psig in an hour without the need for artificial cooling. The pressure will drop to about 5 psig in 24 hours. Thus, the use of the containment spray system is not required to attain the desired pressure reduction, but it could be used to facilitate fission product washdown and to speed pressure reduction if desired.

4. Summary of Radiological Effects

4.1 As indicated previously, the radiological effects of the "maximum credible accident" of interest at off-site locations have two forms; first, the direct radiation from the fission products contained in the free volume space of the reactor enclosure (containment vessel); and second, possible leakage of a small fraction of these fission products from the enclosure, which could cause radiological effects from direct radiation from the passing cloud, direct radiation from radioactive materials deposited on the ground, internal exposure due to inhalation of radioactive materials as the leakage passes the point of exposure, and might require some control of the use of contaminated off-plant land.

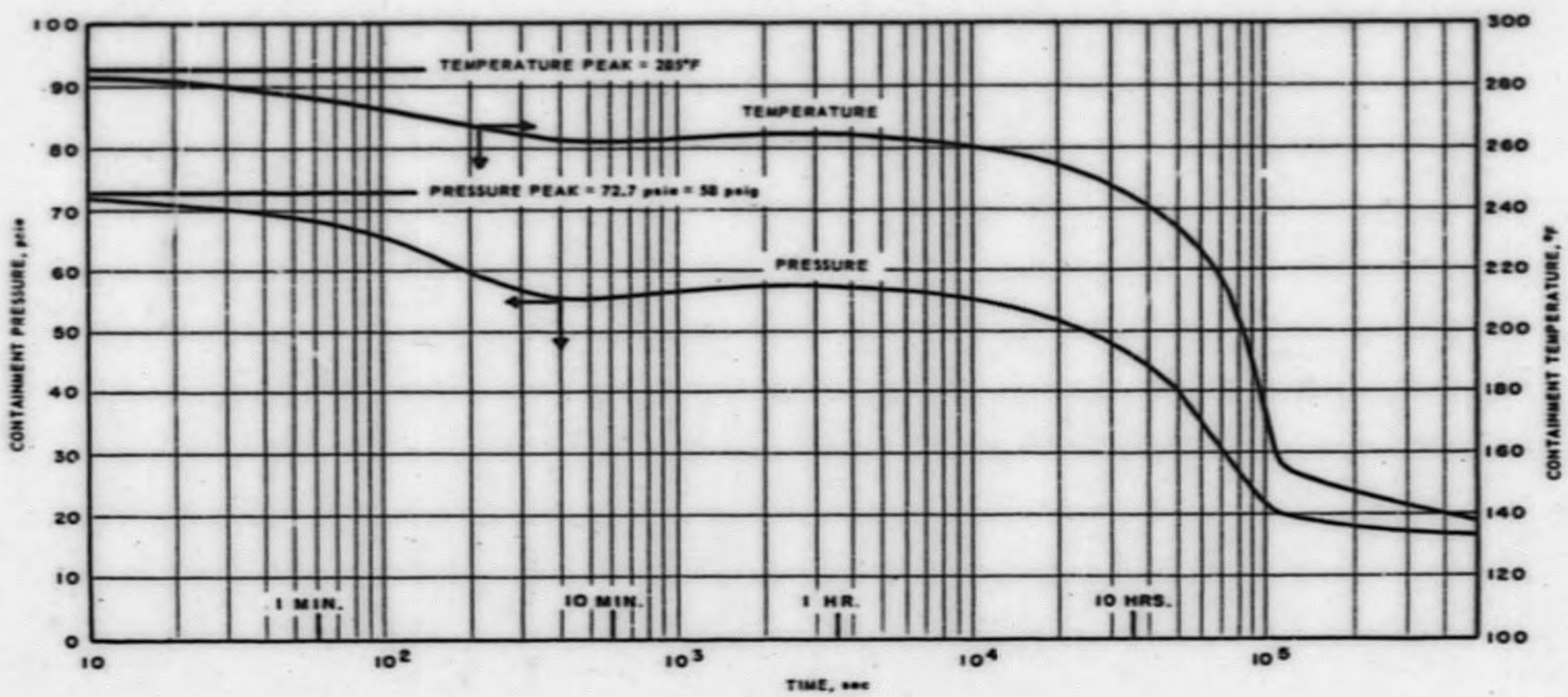


FIGURE IV.9
POST-ACCIDENT CONTAINMENT PRESSURE

FIG. IV. 9

4.2 The off-plant radiological effects of the postulated accident and leakage may be summarized as follows:

4.2.1 No one at an off-plant location (2000 feet or greater from the reactor) would receive a dose of direct external radiation from the enclosure in excess of about 0.8 roentgen.

4.2.2 No one beyond the site boundary would receive an external radiation dose from the passing leakage cloud in excess of about 40 mrads in the first two hours, or in excess of 0.3 rad for the entire course of the accident.

4.2.3 No one beyond the plant boundary would receive a dose of more than about 170 mr from material deposited on the ground, even if occupancy were continued on the cloud centerline for a period of 60 days.

4.2.4 No one beyond the plant boundary, even if exposed on the cloud centerline for the entire course of the accident, would receive internal doses from inhalation of more than about:

25 rems to the thyroid,
12 rems to the lungs, and
$<$ 0.1 rem to bone

4.2.5 Required control of off-plant land would be minimal. No evacuation of residents would appear necessary due to the relatively low doses resulting from ground deposition. Some control of agricultural products may be desirable. The most significant item appears to be control of milk products, which in the least favorable atmospheric diffusion case, may involve about one square mile to a distance of about 6 miles in some direction from the plant, for a month or two.

4.3 It is important to recognize that the predicted effects due to leakage of fission products from the enclosure are the maximum possible for the postulated accident, as no wind direction diversity is considered during the course of the leakage. In all probability, the effects indicated in conclusions 4.2.2 through 4.2.5 will be of greatly reduced significance due to wind direction diversity which will occur during the approximately 8-hour period of significant leakage.

5. Radioactive Materials Available for Environs Effects

5.1 Fission products in the reactor fuel may become available for release to the enclosure free volume, at a rate dependent on the achievement of high temperature in the core due to afterheat as described in Section IV. L. 2. Certain fractions of the various fission products will move from the fuel to the enclosure free volume depending on their physical characteristics. The evaluation was made on the basis of transport from the fuel to an airborne condition in the enclosure free volume in accordance with the following percentages:

100% of the noble gases (Kr and Xe).
25% of the halogens (I and Br).
15% of the volatile solids (Te, Se, Ru, Cs).
0.3% of all other solid fission products.

5.2 The inventory of these fission products which remains in an airborne condition in the enclosure free volume is a function of the degree of "washout" which will occur in the enclosure free volume as a result of condensation of steam. This evaluation considers reduction of free volume fission product inventory by this "washout" as removal, per second, of 9×10^{-5} of the halogens, and 3×10^{-4} of the volatile and other solids. This is based on data from "Meteorology and Atomic Energy", Page 95, Figure 7.5.

5.3 The fission product inventory in the enclosure free volume also is reduced with time due to radioactive decay, which has been factored into the evaluation.

5.4 With regard to direct radiation exposure outside of the enclosure building, only a certain fraction of the fission products remaining in the enclosure free volume will be "visible" from points outside the enclosure. This is due to the fact that portions of the free volume are below ground, and below and within concrete walls and floors of substantial shielding effect. For this plant, a geometric evaluation indicates that not more than 65% of the enclosure free volume is "visible" from a point at the distances evaluated. This evaluation does not consider additional shielding available in some directions due to other plant structures near the enclosure building.

5.5 With regard to possible leakage of fission products to atmosphere through enclosure imperfections, important reduction in possible leakage is provided by the natural cooling of the enclosure free volume which reduces residual enclosure pressure and possible leakage rate. The possible radiological effects of such leakage are based on the assumption that the enclosure leakage may be at a rate as high as 1% per day at the enclosure design pressure, and at reduced leakage rates as enclosure residual pressure is reduced.

5.6 The fission products in the enclosure vapor space at various times after the accident are shown in Table IV.1. For this preliminary evaluation of the "maximum credible accident", the fission product inventory with time has been determined by conservative application of detailed calculations made for a 60 Mwt core of about the same size. The inventory shown on Table IV.1 has been corrected for an assumed 25 Mwt power level for the VSR.

6. <u>Direct External Gamma Radiation from Enclosure</u>

6.1 The quantities of fission products available to contribute to the direct external gamma radiation from the enclosure were based on considerations given above. This direct radiation is a sensitive function of the gamma energy levels of the radioisotopes present, because of the

TABLE IV.1

FISSION PRODUCT INVENTORY IN ENCLOSURE VAPOR SPACE

Time After Accident	Inventory, megacuries		
	Noble Gases	Halogens	"Solids"
5 minutes	0.18	0.04	<0.01
30 minutes	1.64	0.40	0.18
2 hours	2.09	0.22	0.03
6 hours	1.89	0.06	<0.01
16 hours	1.46	<0.01	-
2 days	1.03	-	
5 days	0.68		
10 days	0.36		

variable shielding effect for different gamma energies of the large thickness of air available between the enclosure and the site boundary. Therefore, this evaluation was made by calculating the exposure contribution from each gamma radiation level from each isotope in the noble gas, halogen and volatile solid fission product categories, together with their appropriate daughter fission products, and by considering shielding and build-up factors for both air and the steel enclosure wall.

6.2 The results of these evaluations at distances ranging from one-quarter mile to one mile are shown on Table IV. 2. At 2000 feet, the dosage rate is insignificant for the first 2-1/2 minutes after the accident, it rises relatively rapidly to a plateau of about 100-200 mr/hr in the period from about one-half hour to two hours after the accident, and then reduces again to an insignificant level about one day after the accident. The integrated dose to a person continuously exposed for the entire course of the accident at 2000 feet would not exceed about 0.8 roentgens. In the first two hours, the integrated dose is estimated to be about one-quarter roentgen. Similarly, at one mile exposure is insignificant for the first few minutes after the accident, and reaches a peak of about one mr/hr about 1-1/2 hours after the accident. Integrated doses of about one mr in the first two hours and about four mr for the entire course of the accident are estimated. Due to the geometry of the site boundaries, the doses at one-half and one mile present a more realistic upper limit for off-site locations than those at 2000 feet.

7. Leakage from Enclosure

7.1 The leakage rate of the various fission product groups was determined based on enclosure free volume fission product inventory as outlined above, and the leakage rate variability due to enclosure residual pressure reduction. The leakage rates and integrated leakage at various times after the accident are shown for the fission product groups on Table IV. 3. All leakage rates are insignificant for the first few minutes and then rise rapidly in the period of one-half to one hour. The leakage of noble gases is the most significant due to the larger fraction escaping from fuel, and the fact that the noble gas leakage rate is reduced only by the enclosure volume leakage rate reduction and by radioactive decay. The other fission product classes show leakage rates which reduce to insignificant levels much earlier in time after the accident due to enclosure free volume inventory reduction by "washout".

7.2 The radiological effects of these leakages were evaluated at the following four selected points in the atmospheric diffusion spectrum, which are believed to encompass the conditions which might be encountered at the reactor site.

POINTS OF EVALUATION

Diffusion	Wind Speed	C_y	C_z
Strong Inversion	1 m/s	----	----
Strong Inversion	5 m/s	----	----
Neutral	5 m/s	0.21	0.14
Unstable	5 m/s	0.3	0.3

TABLE IV. 2

DIRECT RADIATION FROM ENCLOSURE

DOSE RATES AFTER ACCIDENT, milliroentgens per hour

Time	1/4 mile	2000 ft.	1/2 mile	1 mile
10 minutes	110	20	5	<0. 1
1 hour	990	170	42	0. 8
2 hours	670	120	28	0. 7
4 hours	360	60	14	0. 4
8 hours	150	24	5	0. 15
1 day	13	1.6	0. 24	<0. 1
2 days	10	1. 1	0. 17	-
5 days	7	0. 7	0. 11	-
10 days	4	0. 4	<0. 1	-

INTEGRATED DOSE FOLLOWING ACCIDENT, milliroentgens

Exposure Period Following Accident	1/4 mile	2000 ft.	1/2 mile	1 mile
10 minutes	10	2	0. 4	<0. 1
1 hour	600	110	30	0. 4
2 hours	1400	260	65	1. 2
4 hours	2400	430	110	2. 3
8 hours	3300	580	140	3. 3
1 day	4100	650	160	3. 5
2 days	4400	680	170	
5 days	4900	740	180	
10 days	5500	800	190	

TABLE IV. 3

FISSION PRODUCT LEAKAGE FROM ENCLOSURE

FISSION PRODUCT LEAKAGE RATES, millicuries per second FROM ENCLOSURE VAPOR SPACE TO ATMOSPHERE

Time After Accident	Noble Gases	Halogens	"Solids"
5 minutes	20	5	0
30 minutes	180	43	20
2 hours	220	23	3
6 hours	200	6	<1
16 hours	140	<1	-
2 days	59	-	
5 days	34		
10 days	16		

INTEGRATED FISSION PRODUCT LEAKAGE, curies FROM ENCLOSURE VAPOR SPACE TO ATMOSPHERE

Time After Accident	Noble Gases	Halogens	"Solids"
First 10 minutes	17	4	1
First hour	550	120	48
First 4 hours	3000	350	75
First 8 hours	5800	440	75
First day	14, 000	420	75
First 2 days	21, 000	420	75
First 5 days	34, 000	420	75
First 10 days	43, 000	420	75

8. Meteorological Diffusion Evaluation Methods

8.1 The atmospheric diffusion methods of Sutton were used for the neutral and unstable cases. Due to the empirically indicated inadequacies of the Sutton method for inversion conditions, calculation methods based on Hanford diffusion results, as outlined in Report HW-54128*, were used for the inversion cases.

8.2 Weather Conditions: This evaluation assumed that the weather conditions involved no precipitation and that the incident occurred during hot summer weather. Precipitation would deposit more contamination close to the plant than this evaluation indicates, thus reducing contamination levels further away. If the incident occurred in cooler weather, the fission product leakage from the enclosure would be less than indicated due to more favorable heat transfer and consequent more rapid reduction in the enclosure post-accident pressure.

8.3 Elevation of Release: Leakage from the enclosure is considered to occur near the ground level. This appears reasonable as most enclosure penetrations are near grade. If the postulated leakage occurred at some significantly different height, such as by emission from the stack, the off-plant consequences of passing cloud dose, ground deposition, and possible inhalation would occur at greater distances than this evaluation shows, but their magnitude would be vastly reduced.

8.4 Initial Dilution by Building Wake: This evaluation recognizes that initial immediate dilution of the leakage will occur due to the turbulent wake of the enclosure structure produced by the passing wind. It is estimated that the effective wake cross-section is of the order of one-half of the vertical cross-section of the enclosure structure. No additional immediate dilution by other nearby structures is considered. This effective wake has been equated to a semi-circle of equivalent area centered at ground level. Centering the initially diluted leakage at some greater height would reduce the off-plant effects of leakage from those evaluated. It is noted that the radius of the equivalent semi-circle is about the same as the enclosure radius. To obtain an estimate of this initial dilution of the leakage, the radius of the equivalent semi-circle was estimated to represent about 1-1/2 standard deviations of the cloud width. From these considerations, virtual source points were calculated at various upward distances dependent upon the diffusion conditions, and are:

150	meters	for strong inversion -	1	m/s	wind	speed
170	"	" " "	5	"	"	"
100	"	" neutral conditions	5	"	"	"
50	"	" unstable "	5	"	"	"

---

* HW-54128, "Calculations on Environmental Consequences of Reactor Accidents", Interim Report, by J. W. Healy, December 11, 1957.

These estimates of the virtual source distances agree generally with the methods of Holland for the neutral and unstable cases and are more conservative for the strong inversion case.

8.5 Wind Direction Diversity: To illustrate the worst possible effects of the postulated leakage, this evaluation assumes no variation in wind direction during the one-day period of significant leakage following the accident. The empirical results from Hanford indicate that a reduction factor of about 1.6 may actually apply for an emission period in the range of 30 to 60 minutes, and a reduction factor of 2 to 3 may apply for a several-hour emission period. Initial evaluation of wind direction variability over a 1-1/2 year period at the site of the Dresden Nuclear Power Station indicates that the average wind direction diversity for any hour, when wind velocities exceed one-half meter per second, is in the range of 100° on the compass. Thus a wind diversity factor possibly as much as 10 may actually apply to a release where the period of emission is an hour or more. These factors should be given consideration when appraising the off-plant radiological effects due to leakage which are presented.

8.6 Travel Time: This evaluation includes the effect of wind travel time from the enclosure to off-plant distances, which has the effect of important reduction in the off-plant radiological effects of the leakage in the 0 to 10 minute period for all cases evaluated, and in the 0 to 60 minute period when the wind speed was taken to be one meter per second.

9. External Radiation Dose from Passing Cloud

9.1 Evaluation of effects of passing cloud air concentrations downwind were estimated using the Sutton and Hanford methods as outlined above. Particular emphasis was taken in this evaluation in the conversion from air concentration to integrated dose for the passing cloud effect. Due to the radioactive decay of the equilibrium fission product mixture which occurs during the post-accident period, the conversion from concentration to dose becomes more favorable in reducing dose as the decay period available increases. For the noble gas, halogen, and solid fission product groups, the concentration required in an infinite cloud to produce a certain dose was evaluated for the radioactive decay periods of interest in the post-accident period. Selected values of the air concentration in an infinite cloud, in units of microcuries per cubic centimeter, which will produce a dose of one mrad per hour with hemispherical geometry are:

AIR CONCENTRATIONS (μc/cc) GIVING
ONE MRAD PER HOUR DOSE RATE

Decay Time	Noble Gases	Halogens	Solids
1 Hours	1.6×10^{-6}	0.78×10^{-6}	1.4×10^{-6}
4 Hours	2.3×10^{-6}	0.79×10^{-6}	2.5×10^{-6}
8 Hours	3.1×10^{-6}	0.88×10^{-6}	2.5×10^{-6}
16 Hours	4.2×10^{-6}	1.0×10^{-6}	2.6×10^{-6}

9.2 The dose from the passing cloud based on uniform concentration and infinite cloud considerations was then corrected for the finite cloud size and Gaussian distribution of cloud concentration. For the various diffusions evaluated, and for cloud sizes calculated at the one-quarter to one mile distances, the ratio of finite cloud dose to infinite cloud dose was found to range from 0.06 to 0.54. The reduction of cloud concentration at the distances evaluated, because of prior deposition on the ground of halogens and solids, was factored into the dose from the passing cloud. This correction is of small magnitude since most of the passing cloud dose is due to noble gases.

9.3 The results of these evaluations are shown on Table IV.4 for various distances. At 2000 feet, considering no wind direction diversity, the passing cloud dose, for inversion conditions with low wind speed, is approximately 40 mrads in the first 2 hours and less than 400 mrads for the entire course of the accident. For other conditions of higher wind speed, the integrated dose is lower by a factor of 5 to 35, depending upon the diffusion condition. At the one-mile distance, a two-hour dose of about 20 mrads, with about 200 mrads for the entire course of the accident, is indicated in the least favorable wind speed and diffusion condition, with similar reductions for the more probable higher wind speeds.

9.4 These calculated doses assume that the receptor is on the center of the cloud path continuously for the period evaluated and that no incidental shielding, such as that provided by housing, is available. Due to the geometry of the site boundaries and the actual current population distribution, the evaluations at a one-half and one-mile distance provide a more realistic upper limit to the probable off-plant dose than that indicated at the 2000 foot distance.

10. External Radiation Dose from Ground Deposition

10.1 The fall-out concentrations of radioactive materials were determined on the basis of particle settling by eddy diffusion only, since settling by gravity is expected to be negligible in this case. It is expected that the particulate radioactive material which might leak from the enclosure will be only a few microns in diameter. If the material were of a significantly larger diameter, it would be washed out at a much faster rate within the containment vessel and thus would not be available for leakage. Also, if the particles were larger, they might not be able to escape from the enclosure since the leakage that may occur is expected to be restricted to that which could pass through minute imperfections in the wall or through penetration seals.

10.2 The extent of halogen and solid fission product deposition on the ground is a function of the apparent deposition velocity. The deposition velocity is considered to be a function of the diffusion condition and wind speed. Deposition velocities used in this evaluation were based on British results cited in HW54128, and are summarized in the following table.

TABLE IV. 4

PASSING CLOUD RADIATION DOSE*

Exposure Period Following Accident	Distance	Dose, millirads: Strong Inversion Wind Velocity 1 m/sec	Strong Inversion Wind Velocity 5 m/sec	Neutral Wind Velocity 5 m/sec	Unstable Wind Velocity 5 m/sec
First 10 minutes	1/4 mile	0.003	0.10	0.04	0.02
	2000 feet	0	0.08	0.02	0.01
	1/2 mile	0	0.02	0.008	0.002
	1 mile	0	0.008	0.001	0.0006
First Hour	1/4 mile	17	4.2	1.6	0.77
	2000 feet	14	3.5	1.1	0.47
	1/2 mile	11	2.9	0.86	0.35
	1 mile	4.2	2.2	0.38	0.14
First 4 Hours	1/4 mile	76	17	6.7	3.1
	2000 feet	71	14	4.9	2.3
	1/2 mile	59	12	3.6	1.4
	1 mile	40	9.7	1.8	0.65
First 8 Hours	1/4 mile	120	27	11	4.9
	2000 feet	110	23	7.6	3.4
	1/2 mile	96	20	5.8	2.3
	1 mile	65	16	2.8	1.0
First Day	1/4 mile	200	43	18	7.9
	2000 feet	180	38	14	5.4
	1/2 mile	150	33	9.6	3.8
	1 mile	110	26	4.7	1.6
First 2 Days	1/4 mile	250	53	22	9.9
	2000 feet	220	47	17	6.5
	1/2 mile	180	41	12	4.7
	1 mile	130	32	5.8	2.1
First 5 Days	1/4 mile	330	70	29	10
	2000 feet	290	61	22	8.5
	1/2 mile	250	54	15	6.1
	1 mile	190	42	7.6	2.7
First 10 Days (Essentially Infinite)	1/4 mile	370	80	34	12
	2000 feet	340	71	25	9.9
	1/2 mile	290	61	17	7.1
	1 mile	220	49	8.8	3.1

*NOTE: No wind direction diversity during release. Dose taken along cloud centerline.

DEPOSITION VELOCITY CHARACTERISTICS

Diffusion	Wind Velocity	Ratio of Deposition Velocity to Wind Velocity		Deposition Velocity, cm/sec	
		Particles	Halogens	Particles	Halogens
Strong Inversion	1 m/s	1.5×10^{-4}	2.4×10^{-3}	0.015	0.24
Strong Inversion	5 m/s	1.5×10^{-4}	2.4×10^{-3}	0.075	1.2
Neutral	5 m/s	3×10^{-4}	4.6×10^{-3}	0.15	2.3
Unstable	5 m/s	6×10^{-4}	8×10^{-3}	0.3	4

10.3 This evaluation provides for correction due to radioactive decay after the material is deposited on the ground. As the amount of deposition is a function of air concentration, and as the air concentration is depleted by prior deposition at locations closer to the source, correction for this depletion has been made for deposition at the distances illustrated. In addition, the dose rate from the deposited material has been corrected for the finite size of the deposited source. This correction is a function of the standard deviation of cloud width, and at one-half and one mile distances varies within the range of 0.30 to 0.54.

10.4 Dosage rates from material deposited on the ground were calculated considering that a deposition of one curie per square meter will produce a dosage rate of ten roentgens per hour, one meter above the ground surface.

10.5 The possible integrated doses from material deposited on the ground, assuming no wind direction diversity during the period of release, are shown on Table IV. 5. The results indicate that even under unfavorable diffusion conditions, the dose possible in the first two hours is less than 10 mr, and of the order of 25 mr for the first day. The integrated dose was carried to an essentially "infinite" 60-day period, during which period the integrated dose would not exceed the order of 150 mr.

10.6 This evaluation assumes no reduction of deposited material during the 60-day period except that due to radioactive decay. In all probability, actual integrated doses would be reduced by the effects of rain and other weathering action. The doses illustrated are the maxima applying at the center of the cloud path with any incidental shielding, such as by housing, not considered.

11. Ground Deposition of Iodine-131

11.1 Due to the radioisotopic composition of the postulated leakage, the most significant individual effect from contamination of off-plant land would be the necessary control of milk production from grazing cattle because

TABLE IV. 5

RADIATION DOSE FROM GROUND DEPOSITION*

		Dose, milliroentgen			
		Strong Inversion		Neutral	Unstable
Exposure Period Following Accident	Distance	Wind Velocity 1 m/sec	Wind Velocity 5 m/sec	Wind Velocity 5 m/sec	Wind Velocity 5 m/sec
First Hour	1/4 mile	< 1	<1	<1	<1
	2000 feet	< 1	<1	<1	<1
	1/2 mile	< 1	<1	<1	<1
	1 mile	< 1	<1	<1	<1
First 4 Hours	1/4 mile	4	6	2	1
	2000 feet	2	4	1	<1
	1/2 mile	1	2	<1	<1
	1 mile	< 1	1	<1	<1
First Day	1/4 mile	31	38	16	11
	2000 feet	22	28	9	6
	1/2 mile	19	23	6	3
	1 mile	10	12	2	1
First 10 Days	1/4 mile	120	130	56	41
	2000 feet	92	100	34	22
	1/2 mile	76	85	21	15
	1 mile	40	48	9	4
First 60 Days (Essentially Infinite)	1/4 mile	200	220	74	56
	2000 feet	150	170	44	29
	1/2 mile	120	130	28	19
	1 mile	60	76	16	8

*NOTE: No wind direction diversity during release.
Dose taken along cloud centerline.

of the iodine-131 food change relationship from vegetation to cattle to milk to man. Following the Windscale incident, the British used a limit of 20 rads to the thyroid as a criterion for control of milk. They indicated that such a dose would result from milk containing 0.1 microcuries per liter, and that this concentration resulted from cattle grazing on pasture containing one microcurie of iodine-131 deposited per square meter.

11.2 Assuming an unvarying wind direction during the postulated leakage period, the maximum deposition is found to occur during strong inversion conditions with a wind speed of 5 meters per second. On the deposition pattern centerline, the maximum deposition at one mile is about 60 microcuries of iodine-131 per square meter. On the cloud centerline, it is found that the deposition is reduced to 1 microcurie per square meter at a distance of about 7 miles. Considering the cloud widths required to reduce to this same limit, an off-plant area about six miles long, with a maximum width of about one-quarter mile, and containing about one square mile, would be contaminated above the proposed control limit. After about one month of radioactive decay, the area above the proposed limit would be small, as the deposition at one mile would be only about 4 microcuries per square meter. These postulations are for the case of no wind direction variation during the approximately 8-hour period of significant iodine leakage. If favorable wind diversity actually occurred during this release period, the significance of the deposition concentrations would be much less. Considering an over-simplified case where the actual wind diversity distributed the material within a 90° angle, the average iodine-131 deposition at one mile would be only about 2 microcuries per square meter.

12. Internal Dose to Thyroid

12.1 Internal exposure to the thyroid gland from inhalation of the fission product mixture in the passing cloud is primarily due to iodine radioisotopes. This exposure was evaluated considering the dose from thyroid deposition of iodine-131, 133, and 135. Other iodine radioisotopes of half lives of 2.3 hours or less were not included, considering their low rem per microcurie ratio for lifetime dosage considerations, and because of the estimated 3 to 6 hour thyroid uptake time after the material is inhaled. The lifetime thyroid dose was evaluated for the three isotopes considering a breathing rate of 20 liters per minute, and a thyroid deposition of 15% of that which was inhaled.

12.2 The total lifetime thyroid dose for inhalation during various periods after the accident, at various distances and atmospheric diffusion conditions, is shown on Table IV.6 At 2000 feet, the lifetime dose from inhalation on the cloud centerline is less than 15 rems for exposure in the first 2 hours, and about 25 rems for exposure during the entire course of the leakage under the least favorable diffusion conditions. For the other more probable diffusion conditions, the dose is less by a factor of from 5 to 100. The similar lifetime thyroid doses for inhalation at a distance of one mile show a maximum of about 4 rems for exposure in the first two hours, and about 9 rems for exposure to the entire course of the leakage.

TABLE IV.6

LIFETIME THYROID DOSE FROM INHALATION OF PASSING CLOUD*

Exposure Period Following Accident	Distance	Dose, rems: Strong Inversion, Wind Velocity 1 m/sec	Strong Inversion, Wind Velocity 5 m/sec	Neutral, Wind Velocity 5 m/sec	Unstable, Wind Velocity 5 m/sec
First Hour	1/4 mile	5.8	1.6	0.2	<0.1
	2000 feet	4.1	1.1	0.1	<0.1
	1/2 mile	3.0	0.8	<0.1	<0.1
	1 mile	0.7	0.4	<0.1	<0.1
First 4 Hours	1/4 mile	26	5.9	0.8	0.3
	2000 feet	18	4.1	0.5	0.1
	1/2 mile	13	3.3	0.2	<0.1
	1 mile	6.2	1.6	<0.1	<0.1
First Day (Essentially Infinite)	1/4 mile	36	7.8	1.2	0.4
	2000 feet	25	5.6	0.7	0.2
	1/2 mile	19	4.5	0.4	0.1
	1 mile	9	2.2	0.1	<0.1

*NOTE: No wind direction diversity during release.
Dose taken along cloud centerline.

12.3 This evaluation assumes no reduction of quantities inhaled due to wind direction diversity during the course of the exposure, or to exposure to reduced air concentrations as would be the case if the receptor were within a structure.

13. Internal Dose to Lung

13.1 Dose to the lungs was evaluated considering that all volatile and other solid fission products inhaled were insoluble, and by use of conventional standard man metabolic factors.

13.2 Considering the composition of the postulated leakage, the analysis indicates that essentially all of the lifetime dose to the lungs is due to the longer-lived radioisotopes of cesium, ruthenium, and tellurium. For exposure off-plant for the entire course of the accident, the maximum calculated lifetime lung dose is about 12 rems. A summary for various distances, times and diffusion is tabluated below.

LIFETIME DOSE TO LUNGS, rems

Exposure Time	Strong Inversion		Neutral	Unstable
At 2000 feet	1 m/s	5 m/s	5 m/s	5 m/s
1st two hours	5	1	<1	<1
Continuous	12	2	<1	<1
At one mile				
1st two hours	2	<1	<1	<1
Continuous	6	1	<1	<1

13.3 This evaluation assumes no reduction of quantities inhaled due to wind direction diversity during the course of the exposure, or to exposure to reduced air concentrations as would be the case if the receptor were within a structure.

14. Internal Dose to Bone

14.1 The analysis of bone dose indicates that essentially all of the contribution is due to the longer-lived radioisotopes of strontium, yttrium, zirconium, barium, and cerium, together with their appropriate daughter fission products. Due to the low fractions of these fission products in the postulated leakage, the analysis indicates that dose to bone at any off-plant exposure, and for any exposure time or diffusion, is always less than 0.1 rem for a lifetime.

DOCKET NO. 50-183
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LICENSE APPLICATION

FOR

VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR

GENERAL ELECTRIC COMPANY
ATOMIC POWER EQUIPMENT DEPARTMENT
2151 South First Street
San Jose, California

LICENSE APPLICATION
FOR
VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR

General Electric hereby applies for a construction permit and operating license for the Vallecitos Experimental Superheat Reactor (hereinafter "VESR") which General Electric plans to construct at the Vallecitos Atomic Laboratory, Alameda County, California.

The VESR will be a research and development facility within the meaning of Section 104(b) of the Atomic Energy Act of 1954, as amended. The VESR will be employed principally as a test bed for superheat reactor fuel, as more fully described in the Preliminary Hazards Summary Report (hereinafter "GEAP 3643") which is attached hereto and made a part hereof.

General Electric plans ultimate operation of VESR using steam produced by the Vallecitos Boiling Water Reactor (hereinafter "VBWR") as well as steam produced by a gas-fired boiler. Accordingly, General Electric requests authority under License No. DPR-1, as amended, to operate the VESR in conjunction with the VBWR. As more fully discussed in Section II-G of GEAP 3643, General Electric does not intend to utilize VBWR-produced steam in operation of the VESR prior to accumulation of operating experience in which steam produced by a gas-fired boiler is the only source of cooling steam. GEAP 3643, consequently, places primary emphasis on operation with gas-fired boiler produced steam and the information contained therein has been primarily predicated on a thermal power limit of 12.5 MW. Future reports will fully support General Electric's request to operate the VESR in conjunction with the VBWR and will justify an increase in thermal power limit.

As stated in GEAP 3643, steam lines extended from the VESR to the VBWR and necessary nozzles added to the VBWR system will be blanked off until operation of the VESR on VBWR steam is approved. General Electric does not believe that the addition of such nozzles to the VBWR constitutes a material alteration, within the meaning of 10-CFR-50.91, which would require the issuance of a construction permit prior to effectuation of that addition.

Design of the VESR has progressed through the conceptual and scoping design stages into the detailed design phases. General Electric respectfully submits that the design information contained in this application, including the safety analyses carried out as a part thereof, provides the reasonable assurance, required by Section 50.35, that the VESR can be constructed and operated without undue risk to the health and safety of the public.

Subsequent to issuance of the Construction Permit requested herein, and prior to issuance of an Operating License, this application will be supplemented by sufficient additional information, as required by Sections 50.35 and 50.36, to assure that the final design will provide reasonable assurance that the health and safety of the public will not be endangered and to permit the conversion of the Construction Permit into an Operating License.

It is anticipated that construction of the VESR will be completed between April, 1962 and February, 1963. Special nuclear material for the fabrication of fuel elements will be required by August, 1961, and a request for allocation of that material is included herein.

This application combines an application for a Construction Permit and Operating License pursuant to 10-CFR-Part 50, an application for License to receive, possess, and use special nuclear material pursuant to 10-CFR-Part 70, for use in connection with the operation of the reactor, and an application for a byproduct material license pursuant to 10-CFR-Part 30 to receive possession of and title to, and to transfer to those authorized to receive the byproduct material which results from the operation of the reactor. If subsequent to issuance of a Construction Permit and after examination of the additional information which will be submitted by General Electric in support of proposed operation of the VESR in conjunction with the VBWR, the Commission concludes that public hearing is required in Docket No. 50-18, as well as in the instant docket before such operation may be authorized, General Electric requests that the applications for operating authority in both dockets be consolidated for review in a single hearing.

I. APPLICATION FOR CONSTRUCTION PERMIT AND LICENSE PURSUANT TO TITLE 10, CODE OF FEDERAL REGULATIONS, PART 50

A. Information Required by Section 50.33

1. Corporate and financial information regarding the General Electric Company is contained in Section I-A-1 of Amendment No. 41 to License Application for Vallecitos Boiling Water Reactor (Docket 50-18) which by reference is made a part hereof. Copies of General Electric's latest Annual Report were submitted to the Commission by letters dated March 30 and November 1, 1960.

2. This application is for a construction permit and operating license for the VESR to be licensed under Section 104(b) of the Atomic Energy Act of 1954.

3. General Electric requests the license be issued for a period of ten (10) years from date of issue and reserves the right to request an extension of this period.

B. Information Required by Section 50.34

Information required by Section 50.34 is contained in GEAP 3643.

C. Agreement Limiting Access to Restricted Data in accordance with Section 50.37

General Electric will not permit any individual to have access to Restricted Data until the Civil Service Commission shall have made an investigation and report to the Atomic Energy Commission on the character associations and loyalty of such individual, and the Atomic Energy Commission shall have determined that permitting such person to have access to restricted data will not endanger the common defense and security.

II. APPLICATION FOR LICENSE TO RECEIVE, POSSESS, AND USE SPECIAL NUCLEAR MATERIAL PURSUANT TO TITLE 10, CODE OF FEDERAL REGULATIONS, PART 70.

A. Information Required by Section 70.22

1. Corporate and financial information regarding the General Electric Company is contained in Section I-A-1 of Amendment No. 41 to License Application for Vallecitos Boiling Water Reactor.

2. Special Nuclear Material will be used in connection with the operation of the VESR which will be constructed at the Vallecitos Atomic Laboratory, Alameda County, California. The general plan for carrying out the research and development activities utilizing this material is described in GEAP 3643.

3. General Electric requests the license be issued for a period of ten (10) years from date of issue and reserves the right to request an extension of this period.

4. The initial Special Nuclear Material required will be Uranium enriched to 3.6% in the isotope U-235. Various types of fuel will be used in connection with the operation and to complete the objectives of the VESR.

5. An allocation of 90.1 kilograms of U-235, required for the initial fuel fabrication, is requested and receipt will be required during August, 1961. The following table 6 shows the estimated Plutonium production, U-235 consumption and operating losses, as well as a schedule of receipt and transfer of special nuclear material by years:

Year	Total Re-ceipts Kg. U-235	Pu Produc-tion Kg.	U-235 Con-sumed in Reactor Kg.	Fuel Fabri-cation Losses Kg.U-235	Scrap Returned For Recovery Kg.U-235	Spent Fuel Returned For Recovery Kg.U-235	Total Inventory Including Reactor Load Kg. U-235
1961	90.1	0	0	0.43	7.3	0	82.37
1962	1.5	.4	1.5	0	0	0	80.72
1963	23.52	1.2	4.5	0.12	1.9	17.0	79.07
1964	21.90	1.2	4.5	0.11	1.79	15.5	79.07
1965	21.90	1.2	4.5	0.11	1.79	15.5	79.07
1966	21.90	1.2	4.5	0.11	1.79	15.5	79.07
1967	21.90	1.2	4.5	0.11	1.79	15.5	79.07
1968	21.90	1.2	4.5	0.11	1.79	15.5	79.07
1969	21.90	1.2	4.5	0.11	1.79	15.5	79.07
1970	21.90	1.2	4.5	0.11	1.79	15.5	79.07

6. General Electric Company has more than 20 years of experience in the field of atomic energy. Uranium-235 was isolated in a General Electric Laboratory in 1940. General Electric was active in the work of the Manhattan District Project during World War II. Since 1946, General Electric has operated Hanford for the AEC. General Electric also operates the AEC's Knolls Atomic Laboratory and the Aircraft Nuclear Propulsion Project in Evendale.

The Atomic Power Equipment Department, which is responsible for the work to be performed as described in this application, was organized in the spring of 1955 and employs approximately 1600 people, representing a wide range of technical skills. George White is the Department's General Manager.

The Department has designed and constructed several power, test, and research reactors, including a 180 electrical megawatt nuclear power plant for the Commonwealth Edison Company of Illinois, a 62 thermal megawatt Reactor for the Allgemeine Electricitats Gesellschaft of Germany, and the VBWR and GETR at the Vallecitos Atomic Laboratory.

Managerial personnel who will direct the operation of the VESR are:

E. W. O'Rorke, Manager Vallecitos Irradiations & Services Operations	Fourteen (14) years experience at Hanford in fuel element fabrication operation. Three (3) years at APED in managerial positions involving Fuel Development activities, Reactor Operations, and Supporting activities.
R. C. Thorburn, Manager Nuclear Safety	Six (6) years experience at Hanford in Radiological Science. Two years (2) at California Research and Development Company as Supervisor of Health Physics. Five (5) years at APED in Nuclear and Reactor Safety Work.
L. Kornblith, Jr., Manager Reactor Technical Operation	Nine (9) years nuclear engineering experience at Enrico Fermi Institute of Nuclear Physics, as Chief Engineer in charge of cyclotron operation. Five (5) years at APED in Reactor work.

The Manager of the VESR will be appointed at a later time, and will possess a high degree of managerial ability and reactor operation experience.

General Electric reserves the right to replace the above individuals with others of similar experience and competence without amendment of this application.

7. A description of the equipment and facilities to be used to protect health and minimize danger to life or property is included in GEAP 3643.

8. A preliminary description of the proposed procedures to protect health and minimize danger to life and property is included in GEAP 3643.

To the best of my knowledge and belief, the information contained herein is accurate.

GENERAL ELECTRIC COMPANY
ATOMIC POWER EQUIPMENT DEPARTMENT

BY /S/ Geo White
George White
General Manager

ATTEST:

/S/ Charles W. Wilder
Attesting Secretary

Subscribed and sworn to before me this
1st day of February, 1961.

/s/ Violet R. Burtt
Notary Public in and for the County
of Santa Clara, State of California.

My Commission expires July, 1964.

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