U. S. Atomic Energy Commission
Docket No. 50-247
Exhibit B-7

Regulatory Suppl File Cy.

Received w/Ltr Dated 10-16-67

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

SEVENTH SUPPLEMENT TO:
PRELIMINARY SAFETY ANALYSIS REPORT

3443

INDIAN POINT UNIT NO. 2 - SEVENTH SUPPLEMENT

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1.1 INTRODUCTION

The design of the fluid system comprising the Safety Injection System for Indian Point Unit No. 2 is virtually completed. The purpose of this report is to inform interested parties of the nature of the final design. Figure 1-1 presents the flow diagram for the system. Figures 1-2, 1-3, 1-4, and 1-5, depict how this system concept was translated into final plant layout design. Additionally, the plot plan is presented as Figure 1-6 and the general plant arrangements are presented in Figures 1-7 through 1-16.

Analysis of the system's capability to fulfill emergency core cooling requirements was presented in the Sixth Supplement to the Preliminary Safety Analysis Report for Indian Point Nuclear Generating Unit No.2. The material covered there will not be covered here.

1.2 DESIGN BASIS

1.2.1 GENERAL BASIS

Adequate emergency core cooling is provided by the Safety Injection System whose redundant components operate in three modes. These modes are injection by the passive accumulator system, by the active safety injection system and by the recirculation system.

The primary purpose of the Safety Injection System is to deliver cooling water to the reactor core in the event of a loss-of-coolant accident to limit the fuel clad temperature and thereby ensure that the core will remain intact and in place, with its essential heat transfer geometry preserved. This protection is prescribed for all break sizes up to and including the hypothetical instantaneous double-ended rupture of the reactor coolant pipe, including the special cases of the failure of a control rod drive mechanism housing and the failure of a steam generator tube.

The basic criterion for assuring that the fuel element will remain in place and substantially intact during a loss-of-coolant accident is that the maximum calculated zircaloy clad temperature for the entire core will not exceed the zircaloy melting temperature, when the core is in its original heat transfer geometry. Also zircaloy-water reactions will be limited to an insignificant amount.

In the Sixth Supplement, the results of analyses were presented which demonstrated that the safety injection system meets the core cooling design objective for all break sizes up to and including the double-ended severance of a reactor coolant loop pipe. As the engineering parameters then used to perform the analyses were in some cases preliminary values, the analyses will be repeated using final parameters and the results presented in new curves as part of the final safety report. Sensitivity studies will be included which will show the approximate value of the peak clad temperature, apparently the result of a break between 0.5 square feet and 3.0 square feet (See Figure 2-8 of the Sixth Supplement).

The abcissa of the curve equivalent to Figure 2-8 will be extended to cover a greater time period to demonstrate that the uncovering of the core with small breaks at times beyond the present scale given does not constitute a problem. In the original study, information was obtained beyond 70 seconds but not shown. For the 0.5 square foot break, the temperature declined further, then rose to a peak value of about 1100°F.

The principal components of the Safety Injection System, which will provide emergency core cooling following a loss-of-coolant accident, consist of, (1) four Accumulators, (2) three high head Safety Injection Pumps, and (3) two Residual Heat Removal Pumps. The injection from these components provides the protection necessary to meet the core cooling objectives. Figure 1-17 is a bar chart which shows the range of core protection as a function of break diameter for various combinations of the above safety injection components as follows:

Bar A: two safety injection pumps
Bar B: two safety injection pumps and one residual heat removal pump
Bar C: one residual heat removal pump and three accumulators
Bar D: two safety injection pumps, one residual heat removal pump and three accumulators.

With minimum on-site emergency power available (two of three diesel generators), the emergency core cooling equipment available is represented by Bar D (two out of three safety injection pumps, one out of two residual heat pumps, and three accumulators). With these systems, the calculated maximum fuel cladding temperature is limited to a value below the cladding melting temperature for all break sizes up to and including the double-ended severance of the reactor coolant pipe.

The remaining three combinations (Bars A, B and C) represent degraded cases with operation of less than the minimum design emergency core cooling equipment. The operation of two safety injection pumps, (Bar A) provides core protection for break sizes up to an equivalent break diameter of 3 to 4 inches.

The operation of one residual heat removal pump with two safety injection pumps (Bar B) increases the range of core protection to a 14 inch equivalent break diameter (pressurizer surge line break).

The operation of one residual heat removal pump and the accumulators (Bar C) has been specifically analyzed for a range of equivalent break areas between the 14 inch pressurizer surge line break and the double-ended severance of the reactor coolant pipe. This analysis shows that the required core protection is provided by these systems for this range. The trend of the results moreover indicates that the range of core protection for this combination actually extends down to an equivalent break diameter of approximately 6 inches.

1.2.2 SAFETY INJECTION SYSTEMS RELIABILITY CRITERIA

To meet the general design basis, the following reliability criteria have been established:

a) Borated cooling water is supplied to the core through flow paths which are separated and redundant in the area of the containment inside of the missile barrier.

b) Loss of injection water through a severed reactor coolant loop or safety injection branch line is considered. (For the double-ended severance of a reactor coolant loop, loss of all safety injection water delivered to that loop is assumed. For rupture of an injection branch line between the loop and check valve, an allowance for spilling of injection water is determined according to the calculated back pressure of each injection line.)

c) Natural phenomena characteristic of the site were considered. (All associated components, piping, structures, power supplies, etc. are designed to Class I seismic criteria.)

d) Layout and structural design specifically protect the injection paths leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. (Single injection lines penetrate the missile barrier, with injection headers located in the missile-protected area. Individual injection lines are connected to the injection header, pass through the barrier and then connect to the loops. Separation of the individual injection lines is provided to the maximum extent practicable. Movement of the injection line associated with a rupture of a reactor coolant loop is accommodated by line flexibility and by the design of the pipe supports such that no damage beyond the missile barrier is possible.)

e) Initial response of the injection systems is automatic, with appropriate allowances for delays in actuation of circuitry and active components. Operation of the accumulators is self-initiated upon loss of Reactor Coolant System pressure. The active portions of the injection systems are automatically actuated by coincidence of low pressurizer water level and low pressurizer pressure signals. In addition, manual actuation of the entire injection system and individual components can be accomplished from the control room. In analysis of system performance, delays in reaching the programmed trip points and in actuation of components are conservatively established on the basis that only emergency on-site power will be available.

f) Redundancy of instrumentation, components, and systems is incorporated to assure that postulated malfunctions will not impair the ability of the systems to meet the design objectives. The system will be effective in the event of loss of normal station auxiliary power coincident with the loss of coolant, and will be tolerant of failures of any single component or instrument channel to respond actively in each system.

g) Provisions for periodic tests are provided to demonstrate, with due credit for manufacturer's performance tests and preoperational test results, the state of readiness and functioning capability of the injection systems. (The systems, including their power supplies, are designed to permit demonstration of readiness when the reactor is operating at power or at a hot shutdown. In addition, extensive shop performance testing of characteristics and preoperational functional testing will be carried out.)

1.3 SAFETY OBJECTIVES OF DESIGN

The following objectives have been adopted for the design of the Safety Injection System and for the evaluation of the adequacy of system performance under postulated accident conditions (Supplement 6).

1.3.1 TANK FUNCTIONS

ACCUMULATOR FUNCTION

The accumulators fulfill the short-term core cooling objective of providing passive safeguarding by rapidly reflooding the core following a large area rupture of the Reactor Coolant System. A large area rupture is defined as a rupture larger than the area exposed by a double ended rupture of the largest connecting pipe to the Reactor Coolant System, and includes the double ended break of a reactor coolant pipe. Operation of the system with three of the four accumulators delivering coolant to core (one accumulator spilling through the break) prevents fuel clad melting and limits metal-water reactions to an insignificant amount. The analysis supporting this result was presented in Supplement 6. Results of a preliminary analysis with only two of four accumulators delivering to the core show that fuel clad temperatures are kept below melting by a suitable margin and that the metal water reaction is negligible. A detailed analysis of the two-accumulator cases will be presented in the Final Safety Analysis Report. The results of the two accumulator analysis will demonstrate the margin provided in the accumulator design, and indicate that required protection can be extended with one accumulator isolated.

The accumulator operation is designed to be self-energized; that is, no external source of power or signal transmission is required in order to accomplish its design function.

The accumulator status relating to its readiness to perform as designed will be continuously monitored in the central control room. All functions requiring mechanical action, except the operation of the check valves at the loop connection are capable of testing under conditions of reactor power operation. The operation of the entire system including these check valves is possible with the reactor at cold shutdown.

BORON INJECTION TANK FUNCTION

This tank supplies 12% boric acid to the suction of the high head safety injection pumps. The purpose of this tank is to supply high boron concentration fluid to the Reactor Coolant System following an unexpected cooldown and depressurization.

1.3.2 PUMP FUNCTION

Pumps which are called upon to operate to fulfill core cooling objectives are designed to provide equal or greater protection against fuel damage than that defined above, for all break sizes up to and including the severance of the pressurizer surge line. In the evaluation of this performance it was assumed that the accumulators function when the system pressure falls below accumulator pressure. Account is taken for the flow of borated water into the system from the accumulators according to the pressure differential and pipe line resistance. However, it is assumed that only those components are operable which can be supplied by two of the three diesel generators in conjunction with the remaining engineered safeguard loads, Section 1.4.2. Further, the system can accommodate a failure at the time of the accident of any single active component to respond to the safety injection signal.

Pumps actuated by the safety injection signal supplement the accumulator function by making up for evaporation and spillage of coolant after accumulator discharge. This function generally does not govern the head or flow parameters for any pump, but influences the mode of operation under specific circumstances, as for example during the change-over from the injection to the recirculation cooling mode.

1.4 SYSTEM DESCRIPTION

1.4.1 GENERAL DESCRIPTION

The principal components of the safety injection system which provide emergency core cooling immediately following a loss of coolant are the four accumulator tanks, the three high head safety injection pumps, and the two low head residual heat removal pumps of the Auxiliary Coolant System. The high-head safety injection pumps and residual heat removal pumps are located in the auxiliary building. The residual heat removal pumps take suction directly from the refueling water storage tank located adjacent to the auxiliary building.

A boron injection tank located in the primary auxiliary building increases the system's capability to inject boron into the Reactor Coolant System. The tank is connected to the suction of the high head safety injection pumps. The tank contains boric acid at 20,000 ppm boron and is isolated from the safety injection suction line by two normally closed parallel valves. The valves open upon a safety injection signal. Flow to the pumps is taken preferentially from the boron injection tank. This is achieved by providing a gas overpressure in the tank in excess of the static head developed by the refueling water storage tank. When the boron solution in the tank is nearly expended, redundant valves automatically isolate the tank and pressure source on a low level signal, allowing flow to the safety injection pumps to come from the refueling water storage tank. Sufficient liquid remains in the tank to guarantee that the gas over the liquid cannot enter the line before the valves close.

The accumulator tanks and the residual heat removal pumps discharge into the cold legs of the reactor coolant piping, thus assuring core cooling by rapidly restoring the water level to a point above the top of the core for large breaks.

The safety injection pumps deliver borated water to two hot and two cold legs of the reactor coolant loops. These pumps augment the flow-pressure characteristics of the accumulator tanks and residual heat removal pumps, providing specifically for the makeup of coolant following a small break which does not immediately depressurize the Reactor Coolant System to the accumulator discharge pressure.

The design capacity of the accumulator tanks is based on one of the four tanks spilling and the remaining three containing sufficient water to fill the volume outside the core barrel below the nozzles, the bottom plenum and one half the core. The accumulator tanks are located inside the containment. The location of each tank is outside the crane wall and therefore each is protected against possible missiles which may be generated from the reactor coolant loop area.

The level of borated water in each accumulator tank can be adjusted remotely during normal plant operations. Refueling water is added using a high-head safety injection pump. Water level can be reduced by draining to the reactor coolant drain tank. Samples of the solution in the tanks can be taken in the sampling station for periodic checks of boron concentration.

The capacity of the refueling water storage tank is based on the requirements for filling the refueling canal and is approximately 350,000 gallons. This capacity provides borated water to assure:

a) A volume sufficient to refill the reactor vessel above the nozzles (21,000 gal.)

b) The volume of borated refueling water needed to increase the concentration of initially spilled water to a point that assures no return to criticality with the reactor at cold shutdown and all control rods, except the most reactive RCC assembly, inserted into the core (50,000 gal.).

c) A sufficient volume of water on the floor to permit the initiation of recirculation (125,000 gal.).

The Safety Injection System also contains the components necessary to assure long-term cooling of the core following delivery of the borated water in the refueling water storage tank. Either of the two recirculation pumps is capable of supplying the necessary long term flow of water for continued core cooling. The recirculation pumps take suction from a sump in the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers which are also located within the containment.

The water flows into the recirculation pump sump through a grating, under a baffle, upwards through a horizontal screen, over a second baffle, and downward to the pump suction. The water velocity through the sump is less than one foot per second.

In the event of a large rupture of the reactor coolant system, the recirculation flow path is within the containment. For the smaller breaks in the Reactor Coolant System where recirculated water must be injected against higher pressures, the system is arranged to deliver the water from the residual heat exchangers to the high-head safety injection pump suction and, by that route, to the reactor coolant loops. The system is also arranged to allow either of the residual heat removal pumps to take over the function of a recirculation pump. Water is delivered from the containment to the residual heat removal pumps from a separate sump inside the containment.

The recirculation pumps, the residual heat exchangers and piping and valves vital to the function of the recirculation loop are located in a missile-shielded space inside the polar crane support wall on the west side of the reactor primary shield.

The eight switch sequence which accomplishes the change over from injection to recirculation is listed below. This switch over takes place when the level indicators in the refueling water storage tank indicate that the fluid has been injected. The level indicators in the containment sump will verify that the level is sufficient within the containment. The sequence is followed regardless of which power supply is available. The time required to complete the switch over is expected to be just the time for the switch gear to function.

1. Terminate safety injection signal in order that the control logic permits manipulation of the system. (At any time following completion of the auto start sequence.)

2. Close Switch One. (Remove and isolate unnecessary loads from the diesels)
   a) Trips one of three high head safety injection pump if all three are operating. (No action if two are operating)
   b) Trips both spray pumps
   c) Closes containment isolation valves at spray pump discharge and isolates the spray additive tank.

3. Close Switch Two (Establish cooling flow for residual heat exchangers)
   a) Starts one service water pump, conventional header
   b) Starts one component cooling pump

4. Close Switch Three (Remove and isolate unnecessary loads from the diesels)
   a) Trips both residual heat removal pumps
   b) Closes isolation valves at pump suction and discharge

5. Close Switch Four (Initiate internal recirculation flow)
   a) Opens valves on discharge of recirculation pumps
   b) Starts recirculation pump A (If A fails to start, use manual start on Pump B)
      (Pump B control switch is adjacent to switch four)

6. Close Switch Five (action nullified if only 2/3 diesels) (establish additional cooling capability if power permits)
   a) Starts second service water pump, conventional header
   b) If a) complete, starts second component cooling pump
   c) If b) complete, starts recirculation pump B

7. Check Reactor Coolant System Pressure
   A. If pressure greater than 150 psig, close Switch Six then go to Switch Eight (Provide recirculation at elevated system pressure)
      a) Aligns flow from residual heat exchanger to high head safety injection pumps
   B. If pressure is less than 150 psig, close Switch Seven then go to Switch Eight (Provides recirculation at low system pressure)
      a) Trips high head safety injection pumps

8. Close Switch Eight (Complete Safety Injection System isolation)
   a) Close safety injection pump minimum flow recirculation valves
   b) Complete the isolation of the flow path to the refueling water storage tank

Although the listed switches are manual, each automatically causes the operations listed. An indicating lamp is provided to show the operator when the operations of a given switch have been performed and when he should proceed with the next switching operation. In addition, lamps indicating completion of the individual functions for a given switch are provided. These lamps are adjacent to the switches. The time required to complete the switch over is expected to be just the time for the switchgear to operate. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from controls within the control room.

Assuming that three high head safety injection pumps, two residual heat removal pumps, and two containment spray pumps are running at their maximum capacity, the time sequence, from the time of the safety injection signal, for the change over from injection to recirculation is as follows:

a) In approximately ten minutes, sufficient water has been delivered to provide the required NPSH to start the recirculation pumps.

b) In approximately fifteen minutes, the first low level alarm on the refueling water storage tank sounds. The operator, at this point, takes appropriate action to assure that only two safety injection pumps, one residual heat removal pump, and two spray pumps are operating. Sufficient NPSH now exists for the operating pumps to run until the refueling water storage tank is empty. This alarm also serves to alert the operator to prepare for switch over to the recirculation mode.

c) The second low level alarm on the refueling water storage tank sounds at approximately 22 minutes. At this time, the operator must perform the switch over operation, if it has not been done previously.

Valves of the Safety Injection System which are under manual control (that is, valves which normally are in their ready position and do not receive a safety injection signal) have their positions indicated on a common portion of the control board. The indication is in the form of bright-dim white lights. At any time during operation when one of these valve's position is not in the ready condition for injection, its light will show dim in contrast with the bright lights of the remaining valves. In addition an audible annunciation will alert the operator to the condition and he will restore the valve to its normal ready position.

Testing

Routine periodic testing of the engineered safeguards components and all necessary support systems is possible. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions to be prescribed in the Technical Specifications. These conditions would include such matters as the period within which the component should be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

The operation of the remote stop valves in the accumulator tank discharge line may be tested by opening the remote test valves just downstream of the stop valve. Flow through the test line can be observed on instruments and the opening and closing of the discharge line stop valves may be sensed on this instrumentation. Test circuits are provided to periodically examine the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is raised.

Individual components of the Safety Injection System may be tested anytime the operator wishes, no inflow to the Reactor Coolant System will occur whenever the reactor coolant pressure is above 1500 psi. Valves may be manipulated from the control room as desired.

1.4.2 ELECTRICAL SUPPLY

Three diesel generators for Unit No. 2 provide reliable on-site emergency power for the engineered safeguards loads in the event of failure of station auxiliary power. Thus, even if a-c power to the station is lost concurrent with an accident, power is available for the engineered safeguards from on-site emergency power sources, capable of supplying the engineered safeguards load to assure protection of the public health and safety for any loss-of-coolant accident.

The safeguards loads which are powered from the normal or emergency power sources after the safety injection signal are:

		Starting Sequence and Number of components started	
		2 of 3 Diesels	Outside Power or 3 of 3 Diesels
a)	High Head Safety Injection Pumps	2	3
b)	Residual Heat Removal Pumps	1	2
c)	Containment Spray Pumps	1	2
d)	Service Water Pumps (Nuclear header)	2	2
e)	Containment Fan Cooler Units	3	5
f)	Auxiliary Feedwater Pumps	1	2
g)	Component Cooling Water Pumps	N.A.S.	N.A.S.
h)	Charging Pumps	N.A.S.	N.A.S.
i)	Auxiliary Building Fans	N.A.S.	N.A.S.

N.A.S. Not Automatically Started

All components necessary for proper operation of the safeguards system are operable from either local stations or from the control room. Pumps are operable at the switch gear locations under the control room and the valves can be manipulated from the motor control center in the auxiliary building.

1.5 COMPONENTS

1.5.1 TANKS

1. Accumulator Tanks

The tanks are stainless steel lined carbon steel and designed to ASME Section III, Class C. The charging gas is nitrogen and the borated water is driven by the expansion of the contained gas charge. Connections for periodically remotely venting or charging the gas space and for remotely draining or filling the fluid space, during normal plant operation, are provided. A relief valve is provided which will protect the tank as the operator charges the system with fluid.

The design parameters are given in Table 1-1.

TABLE 1-1

ACCUMULATOR TANKS' DESIGN PARAMETERS

Number	4
Type	Stainless steel lined carbon steel
Design Pressure	700 psig
Design Temperature	300°F
Operating Temperature	100-150°F
Normal Pressure	660 psig
Total Volume	1100 ft^3
Normal Water Volume	700 ft^3

2. Boron Injection Tank

The tank is stainless steel and is located iin the primary auxiliary building. The charging gas is nitrogen and the 12% boric acid is driven by the expansion of the gas contained in bottles adjacent to the tank.

Redundant tank heaters and line heat tracing are provided to insure that the solution will be stored at a temperature in excess of the solubility limit (130°F at a concentration of 20,000 ppm boron).

The heating elements are located near the bottom of the tank. The heated water rising from the elements provides natural circulation to keep the contents of the tank well mixed. The temperature set point on the controls is set high enough to ensure that no boric acid crystal precipitation will occur.

The equipment employed with the boron injection tank will be designed to the same quality standards and codes as the rest of the engineered safeguards equipment and as seismic Class I.

BORON INJECTION TANK
DESIGN PARAMETERS

Volume, solution	2150 gallons
Boron concentration (as boric acid)	20,000 ppm
Design pressure	100 psig
Design temperature	250°F
Operating pressure	70 psig
Operating temperature	150-180°F
Material	Stainless Steel

1.5.2 SAFETY INJECTION SYSTEM PUMPS

Three high-head safety injection pumps supply borated water to the Reactor Coolant System. The pumps are horizontal centrifugal type, driven by electric motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material.

Redundant tank heaters and line heat tracing are provided to insure that the solution will be stored at a temperature in excess of the solubility limit (130°F at a concentration of 20,000 ppm boron).

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1.5.2 SAFETY INJECTION SYSTEM PUMPS

Three high-head safety injection pumps supply borated water to the Reactor Coolant System. The pumps are horizontal centrifugal type, driven by electric motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material.

Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code. The acceptance standard for the liquid penetrant test is USA S.I. B 31.1, Code for Pressure Piping, Case N-10.

The pump design is reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include evaluation of the shaft seal and bearing design to determine that adequate allowances have been made for shaft deflection and clearances between stationary parts. The pumps and seals are reviewed to insure compatability of materials when handling radioactive material. Where welding of one pressure containing part to another is necessary, the fabricator is submitting welding procedures to Westinghouse for review and approval. This procedure must include evidence of qualification necessary for compliance with Section IX of the ASME Boiler and Pressure Vessel Code Welding Qualifications. This requirement applies to any repair welding performed on pressure containing parts.

The only pumps in the safety system requiring cooling under accident conditions are the safety injection pumps. Water cooling is required only by the pump bearings and is provided by boosters pumps driven by the safety injection pump motors. The pump seals are designed to operate at accident conditions without cooling water.

The pressure-containing parts of the pump are hydrostatically tested in accordance with Paragraph UG-99 of Section VIII of the ASME Code. Each pump is given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps are run at design flow and head, shut-off head and at additional points to verify performance characteristics. NPSH is established at design flow by means of adjusting suction pressure for a representative pump. This test is witnessed by qualified Westinghouse personnel.

1.5.3 VALVES

All parts of the safety injection valves in contact with borated water are austenitic stainless steel or equivalent corrosion resistant material. The motor operators on the injection isolation valves are capable of rapid operation, as defined later in this Section. All valves required for operation during safety injection or isolation of the system have remote full open or full close indication in the control room.

The check valves which isolate the Safety Injection System from the Reactor Coolant System are installed as close as practical to the reactor coolant piping to reduce the probability of a safety injection line rupture causing a loss-of-coolant accident.

A relief valve is installed on the safety injection pump discharge header to prevent overpressure caused by temperature changes or by in-leakage from the Reactor Coolant System. This valve is sized 3/4 x 1 inch and set at 1750 psig with a capacity of 5 gpm. The discharge is to the pressurizer relief tank.

The gas relief valve on the accumulator tanks protects them from pressures in excess of the design values.

Leakage across the valve disc will be specified and acceptance tested to the following values:

a. Conventional globe - 10 cc/hr/in. of nominal pipe size
b. Gate valves - 10 cc/hr/in. of nominal pipe size
c. Motor operated gate valves - 3 cc/hr/in. of nominal pipe size
d. Check valves - 10 cc/hr/in. of nominal pipe size
e. Accumulator check valves- 2 cc/hr/in nominal pipe diameter (seat leakage)

Relief valves will be totally enclosed.

Motor Operated Valves

The stainless steel valves employed in the Safety Injection System are designed and built in accordance with detailed Westinghouse specifications. The pressure containing parts (body, bonnet and discs) follow the design criteria established by the USA S.I. 16.5 or MSS SP66 specifications. The materials of construction for these parts are procured to conform with ASTM A182, F316 or F304 or A351 Gr CF8M or CF8. All materials in contact with the primary fluid, except the packing, are austenitic stainless steel or equivalent corrosion resisting material. The cast pressure containing components are subject to radiographic inspection as outlined in ASTM E-71 Class 1 or Class 2. The body, bonnet and discs are subject to a liquid penetrant inspection conducted in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Appendix VIII. The liquid penetrant acceptable standard is as outlined in USA S.I. B31.1 Case N-10.

The body-to-bonnet joint is designed to conform with ASME Boiler and Pressure Vessel Code Section VIII or USA S.I. B16.5 with a fully trapped, controlled compression, spiral wound asbestos gasket. The body-to-bonnet bolting and nut materials are procured to conform with ASTM A193 and A194, respectively.

The entire assembled unit is subject to a hydrostatic test as outlined in MSS SP-61 with the exception that the test is maintained from a minimum period of 30 minutes per inch of wall thickness. Any leakage is cause for rejection. Special attention is paid to the seating surface design.

The units are constructed principally with a venturi design. The venturi design reduces the seating area, hydraulic unbalance, and stroking distance. The seating design is of the parallel disc design or the flexible wedge design. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator has to work only against the frictional component of the hydraulic unbalance on the disc and the packing box friction. The discs are guided throughout the full disc travel to prevent chattering and provide easy gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The steam material is ASTM A276 condition B or precipitation hardened 17-4 stainless steel procured and heat treated to Westinghouse specification. These materials are selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. The packing material is John Crane 187-I or equal. The valve stuffing box is designed with a lantern ring leak-off connection with a minimum of a full set of packing above the lantern ring; a full set of packing is defined as a depth of packing equal to 1-1/2 times the stem diameter. The experience with this stuffing box design and the selection of packing and stem materials has been very favorable in both conventional and nuclear power plants.

The motor operator, "Limitorque", is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a "hammer blow" feature that allows the motor to impact the discs away from the fore or backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operational tested, cleaned and packaged in accordance with Westinghouse specifications. All manufacturing procedures employed by the valve supplier such as hard facing, welding, repair welding and testing are submitted to Westinghouse for approval.

For those valves which must function on the safety injection signal, 10 sec operation is specified. For all other valves, the operator will complete its function in 120 sec.

Valves which must function against system pressure are designed such that they will function with a pressure drop equal to full system pressure across the valve disc.

Manual Valves

The stainless steel manual globe, gate and check valves are designed and built in accordance with the requirements outlined in the motor operated valve description above.

The carbon steel valves are built to conform with USA S.I. B16.5 or MSS SP-66. The materials of construction of the body, bonnet and disc are procured to conform with ASTM A105 Grade II, A181 Grade II or A216 Grade WCB or WCC. The carbon steel valves pass only non-radioactive fluids and are subjected to hydrostatic test as outlined in MSS SP-61 except that the test pressure shall be maintained for at least 30 minutes per inch of wall thickness.

Accumulator Check Valves

The pressure containing parts of this valve assembly are designed in accordance with MSS SP-66. All parts in contact of the operating fluid are of austentic stainless steel or of equivalent corrosion resistant materials procured to applicable ASTM or WAPD specifications. The cast pressure-containing parts are radiographed in accordance with ASTM E-94 and the acceptance standard as outlined in ASTM E-71. The cast pressure-containing parts, machined weld end, furnished hard facings, and gasket bearing surfaces are liquid penetrant inspected per ASME B&PV Code Section VIII and the acceptance standard is as outlined in USA S.I. B31.1 Code Case N-10. The final valve is hydrotested per NSS SP-66 except that the test pressure is maintained for at least 30 minutes. The seat leakage test is conducted in accordance with the manner prescribed in MSS SP-61 except that the acceptable leakage is 2cc/hr/in, nominal pipe diameter.

The valve is designed with a low pressure drop configuration with all operating parts contained within the working parts within the body, which eliminates those problems associated with packing glands exposed to boric acid. The various working parts were selected for their corrosion resistant, tensile, and bearing properties. The clapper arm shaft is manufactured from 17-4 PH stainless steel heat treated to Rockwell C hardness of 32 to 37 after aging at 593C (1100°F) for 4 hours. Keyhole impact strength is 22 foot pounds minimum. The clapper arm shaft bushings are manufactured from Stellite No. 6 material.

The disc and seat rings are manufactured from a forging. The mating seals are hard faced with Stellite No. 6 to improve the valve seating life. The disc is permitted to rotate providing a new seating surface after each valve opening.

The valves are intended to be operated in the closed position with a normal differential pressure across the disc of approximately 1700 psi. The valves remain in this position except for testing and safety injection. Since the valve will not be required to normally operate in the open condition, hence be subjected to the abuse of flowing operation or subjected to impact loads caused by sudden flow reversal, it is expected that this equipment will not have difficulties performing its required functions.

When valve action is required, a differential pressure of less than 25 psig can shear any particles that may tend to prevent valve functioning. Although the working parts are exposed to the boric acid solution contained within the reactor coolant loop, a boric acid "freeze up" can not occur with this low a concentration.

The experience derived from the check valves employed in the Emergency Injection System of Carolina - Virginias Tube Reactor in a similar system indicates that the system is reliable and workable.

1.5.4 PIPING

All safety Injection System piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections at the safety injection pumps.

The piping beyond the accumulator tank stop valves is designed for reactor coolant system conditions (2485 psig, 650°F). All other piping connected to the accumulator tanks is designed for 700 psig and 300°F.

The safety injection pump suction piping (210 psig at 300°F) from the refueling water storage tank is designed for low pressure losses to meet NPSH (net positve suction head) requirements of the pumps.

The high pressure injection branch lines (1500 psig at 300°F) are designed for high pressure losses to limit the flow rate out of the branch line which may have ruptured at the connection to the reactor coolant loop. The branch lines are sized so that such a break will not result in a violation of the design criteria for the Safety Injection System.

The piping will be designed to meet the requirements set forth in (1) the USA S.I. B31.1 Code for Pressure Piping, (2) Nuclear Code Case N-7, (3) USA S.I. Standards B36.10 and B36.19, (4) ASTM Standards, and (5) supplementary plus additional quality control measures delineated in Westinghouse specification, defined on the next page.

Minimum wall thicknesses are determined by the USA S.I. Code formula found in the power piping Section 1 of the USA S.I. Code for Pressure Piping. This minimum thickness is increased to account for (1) the manufacturer's permissible tolerance of minus 12-1/2 per cent on the nominal wall, and (2) a 10 per cent allowance for wall thinning on the external radius during any pipe bending operations in the shop fabrication of the subassemblies.

Purchased pipe and fittings will have a specified nominal wall thickness that is no less than the sum of that required for pressure containment, mechanical strength, manufacturing tolerance, and an allowance for wall thinning associated with shop bending.

Thermal and seismic piping flexibility analyses are performed as required. Special attention was directed to the piping configuration at the pumps with the objective of minimizing pipe imposed loads at the suction and discharge nozzles.

Pipe and fitting materials are procured in conformance with all requirements of the latest ASTM and USA S.I. specifications. All high pressure materials will be verified for conformance to specification and documented by certification of compliance to ASTM material requirements. Specifications will impose additional quality control upon the suppliers of pipes and fittings as listed below.

1. Check analyses will be performed on both the purchased pipe and fittings.

2. Pipe branch lines between the reactor coolant pipes and the isolation stop valves will be purchased to ASTM A376 and the supplementary requirement S7 covering ultrasonic tests.

3. Fittings will be purchased to the specified requirements of ASTM A403 and quality control requirements of the fittings are equivalent to those for the pipe.

Shop fabrication of piping subassemblies is performed by reputable suppliers in accordance with specifications which define and govern material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging and shipment.

All welds in run sizes 2-1/2" and larger are of full penetration design. Reducing tees are used where the branch size exceeds 1/2 of the header size. Branch connections of sizes that are equal to or less than 1/2 of the header size are of a design that complies to rules for reinforcement set forth in the USA S.I. B31.1 Code for Pressure Piping. Bosses for branch connections are attached to the header by means of full penetration welds. All welding is performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code Section IX, Welding Qualifications. The shop fabricator is required to submit all welding procedures and evidence of qualifications for review and approval prior to release for fabrication. All welding materials used by the shop fabricator must have prior approval.

All high pressure piping butt welds containing radioactive fluid, at greater than 600 °F temperature and 600 psig pressure or equivalent, will be radiographed. The remaining piping butt welds will be randomly radiographed. The technique and acceptance standards are those outlined in UW-51 of the ASME B&PV Code Section VIII. In addition, butt welds are liquid penetrant examined in accordance with the procedure of ASME B&PV Code, Section VIII, Appendix VIII and the acceptance standard as defined in the USA S.I. Nuclear Code Case N-10. Finished branch welds are liquid penetrant examined on the outside and where size permits, on the inside root surfaces.

A post-bending solution anneal heat treatment is performed on hot-formed stainless steel pipe bends. Completed bends are then be completely cleaned of oxidation on all affected surfaces. The shop fabricator is required to submit the bending, heat treatment and clean-up procedures for review and approval prior to release for fabrication. Cleaning by acid pickling is not permitted.

General cleaning of completed piping subassemblies (inside and outside surfaces) is governed by basic ground rules set forth in the specifications. For example, these specifications prohibit the use of hydrochloric acid and limit the chloride content of service water and demineralized water.

Packaging of the piping subassemblies for the shipment is done so as to preclude damage during transit and storage. Openings are closed and sealed with tight-fitting covers to prevent entry of moisture and foreign material. Flange facings and weld end preparations are protected from damage by means of wooden cover plates securely fastened in position. The packing arrangement that the shop fabricator proposes is subject to approval.

1.5.5 COMPONENT SUPPORTS

For the hypothetical double-ended severance of a reactor coolant pipe, the functional integrity of the Safety Injection System connections to the remaining reactor coolant loops is not impaired. This integrity is established and maintained by the application of the following design.

(1) The reactor vessel, steam generators and pumps are supported and restrained to limit their movement under pipe break conditions (including a double-ended main pipe rupture) to a maximum amount will will assure the integrity of the steam and feedwater piping. The safety injection piping in the intact loops is designed to accommodate the limited movement of the loop components without failure. The coolant loop supports are designed to restrict the motion to about one-tenth of an inch, whereas calculations show that the attached safety injection piping can sustain a 3-inch displacement without exceeding the working stress range.

(2) The safety injection piping serving each loop are anchored at the missile barrier in each loop area to restrict potential accident damage to the portion of piping beyond this point. The anchorage is designed to withstand, without failure, the thrust force on the safety injection branch line severed from the reactor coolant pipe and discharging safety injection flow to atmosphere, and to withstand a bending moment equivalent to that which produces failure of the safety injection piping under the action of free end discharge to atmosphere or motion of the broken reactor coolant pipe to which the safety injection piping is connected. This prevents possible failure upstream from the support point where the branch line ties into the safety injection piping header.

All hangers, stops and anchors are designed in accordance with USA S.I. B31.1 Code for Pressure Piping and ACI 318 Building Code Requirements for Reinforced Concrete which provide minimum requirements on materials, design and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Specifically, these standards require the following:

(1) All materials used must be in accordance with ASTM specifications which establish quality levels for the manufacturing process, minimum strength properties, and for test requirements which assure compliance with the specifications.

(2) Qualification of welding processes and welders for each class of material welded and for types and positions of welds.

(3) Maximum allowable stress values are established which provide an ample safety margin on both yield strength and ultimate strength.

1.5.6 HEAT EXCHANGERS

The Residual Heat Exchanger and the Component Cooling Heat Exchangers must meet the requirements of the applicable sections of the ASME Boiler and Pressure Vessel Code. The tube side of the residual heat exchanger is a class C Nuclear Vessel to which paragraph N-2113 applies. The shell side of the Residual Heat Exchanger is a Section VIII welded vessel. The component cooling heat exchanger is also a Section VIII welded vessel. The Code has strict rules regarding the wall thicknesses of all pressure containing parts, material quality assurance provisions, weld joint design, radiographic and liquid penetrant examination of materials and joints, and hydrostatic testing of the unit as well as requiring final inspection of the vessel by a Code inspector.

The designs of the heat exchangers conform to the requirements of TEMA (Tubular Exchanger Manufacturers Association) for Class R Heat Exchangers. Class R is the most rugged class of TEMA heat exchanger and is used for units where safety and durability are required under severe service conditions. Items such as: tube spacing, flange design, nozzle location, baffle thickness and spacking, and impingement plate requirements are set forth by TEMA Standards.

Practice to assure that the heat exchanger designs meet the operating conditons without producing tubing vibration is to evaluate the design by means of an empirical formula. For example, the Residual Heat Exchanger has been evaluated by a formula developed and verified through many years of application by the Lummus Co., as follows:

$$L = 6.5 \left(\frac{Eb\,(d_o^{\,4} - d_i^{\,4})}{d_o G^2 B} \right)^{0.25}$$

where

L = pitch of baffles, inches

E = modulus of elasticity of tube materials

G = mass velocity, lb/ft^2-sec

V = specific volume of water on outside, ft^3/lb

d_o = outside diameter of tubes, inches

d_i = inside diameter of tubes, inches

b = free space between tubes, inches

This formula gives the limiting pitch of the baffles necessary to avoid possibility of vibration. By substituting the values applicable to this heat exchanger in the equation, L equals 39.26 inches whereas L is designed for 30 inches.

In addition to the above, additional design and inspection requirements are imposed by Westinghouse to assure rugged, high quality heat exchangers such as: confined-type gaskets, main flange studs with two nuts on each end instead of one to assure permanent leak tightness, general construction and mounting brackets sutiable for the plant seismic design requirements, tubes and tube sheet(s) capable of withstanding full secondary side pressure and temperature with atmospheric pressure on the primary side, ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Code of all steel tubes before bending, penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds, a hydrostatic test duration of not less than thirty minutes, the witnessing of all hydro and penetrant tests by a qualified inspector, a thorough final inspection of the unit for good workmanship and the absence of any gouge marks or other scars that could act as stress concentration points, and a review of the radiographs and certified chemical and physical test reports for all materials used in the units.

The Residual Heat Exchanger is a conventional shell and U-tube type unit. Tubes are seal welded to the tube sheet. Nozzle connections are flanged to facilitate tube bundle removal for inspection and cleaning. The shell is SA-106 Grade A or B carbon steel the tubes are SA-213 TP-304 or TP-316 stainless steel, and the channel and tube sheet are type 304 or 316 stainless steel.

Each Component Cooling Heat Exchanger is a fixed tube sheet type heat exchanger with removable flanged channel covers to permit rodding of the tubes. All piping connections are welded. The heat exchanger has a welded carbon steel shell and channels, and carbon steel channel covers. Tubes are Admiralty and the tube sheet is carbon steel. Sacrificial anodes are furnished in the channels to prevent galvanic corrosion.

1.5.7 MOTORS

Motors Located Outside of the Containment

Motor electrical insulation systems are supplied in accordance with USA S.I., IEEE and NEMA standards and are tested as required by such standards. Temperature rise design selection is made with some allowance for the occurence of accident conditions such that normal long life is achieved. Periodic electrical insulation tests made during the lifetime of the plant will detect deterioration, if any, of the insulation system.

Criteria for motors of the Safety Injection Systems require that under any anticipated mode of operation, the motor name plate rating is not exceeded. The motors have a 1.15 service factor for normal operation. Pump design and test criteria insure that motor loading does not exceed the application criteria.

Motors Inside the Containment

Insulation for motors on these applications is as normally supplied for 2300 volt systems even though it operates on only 460 volts. This high voltage insulation insures reliability under accident conditions.

Bearings are anti-friction, ball type, grease lubricated on which high temperature experience has been accumulated. Bearing loading and high temperature tests have been performed and the expected bearing lifes equal or exceed those specified by the American Federation of Bearing Manufacturers (AFBMA). Pump or fan motors which have a routine function in addition to a Safety Injection System function have bearing vibration detectors to continuously monitor for abnormal bearing conditions. Motor housing designs permit no air or vapor pressure differential across the bearings during or after the containment pressure rise associated with the postulated loss-of-coolant accident.

The motors are enclosed fan heat exchanger cooled. Motor bearings are cooled by the heat exchanger fan discharge air. Housings are designed to vent at accident pressures in order to prevent their collapse. Motor cooling circuits are sufficient to protect the motors from any additional entering atmosphere whether the motors are operating or not.

Motors which operate only during or after the postulated accident will be designed as if used in continuous service. Periodic operation of the motors and tests of the insulation will ensure that the motors remain in a reliable operating condition.

Some of the motors are provided only to drive safeguard equipment. Although these motors are normally run only for test, the design loading and temperature rise limits are based on accident ambient conditions. Normal design margins are specified for these motors to make sure the expected lifetime includes allowance for the occurrence of accident conditions.

1.5.8 INSPECTION AND INSTALLATION OF EQUIPMENT IN THE FIELD

Equipment and materials are delivered by either railroad, barge or motor vehicle. As material is received, it is checked for cleanliness, damage, and checked against the bill of material specification for completeness. If trouble is found, an immediate request is made for rectification. Special inspection procedures and specifications for proper repairs are prepared in the event of damage.

After the receiving check, the materials and equipment are placed in storage if erection cannot proceed immediately. Small items are placed in a warehouse and segregated according to material classification or designated ultimate use. The very large items of equipment are stored outdoors, off the ground, and covered. All openings remain sealed until erection except when further inspection or pre-erection work may be required; afterwards they are resealed until installation.

Components and materials destined for use in the nuclear and engineered safeguards systems receive special consideration during storage and installation to assure the assembled systems do not contain foreign material that could create operating problems.

Special attention is given to maintaining the equipment in a clean and uncontaminated condition during erection. Requirements for the highest grade commercial cleanliness are specified. Specifications are prepared which dictate special handling instruction and precautions to be observed when opening and inspecting any internal workings where foreign material could cause damage or impair the proper operation of the equipment.

Dirt and debris which could contaminate the atmosphere and the equipment in the immediate area of the building are continuously removed. All equipment is protected from physical harm, and kept sealed when not open for inspection.

Should it be necessary to open up a piece of equipment in an area where the atmopshere is contaminated, as a result of grinding or concrete finishing for example, the cause of the air pollution is terminated and/or a protective tent is erected to minimize chances of contamination.

Dessicants are used and periodically monitored in components which are susceptible to damage by moisture. Heaters installed in equipment for moisture control are kept energized when required. Special precautions are taken to assure that the dessicant has been removed prior to system operation.

As part of the final cleaning procedures, a visual inspection of each system is performed following a solvent wash. All systems are flushed using demineralized water during which time temporary screens are installed in the pump suction lines as required.

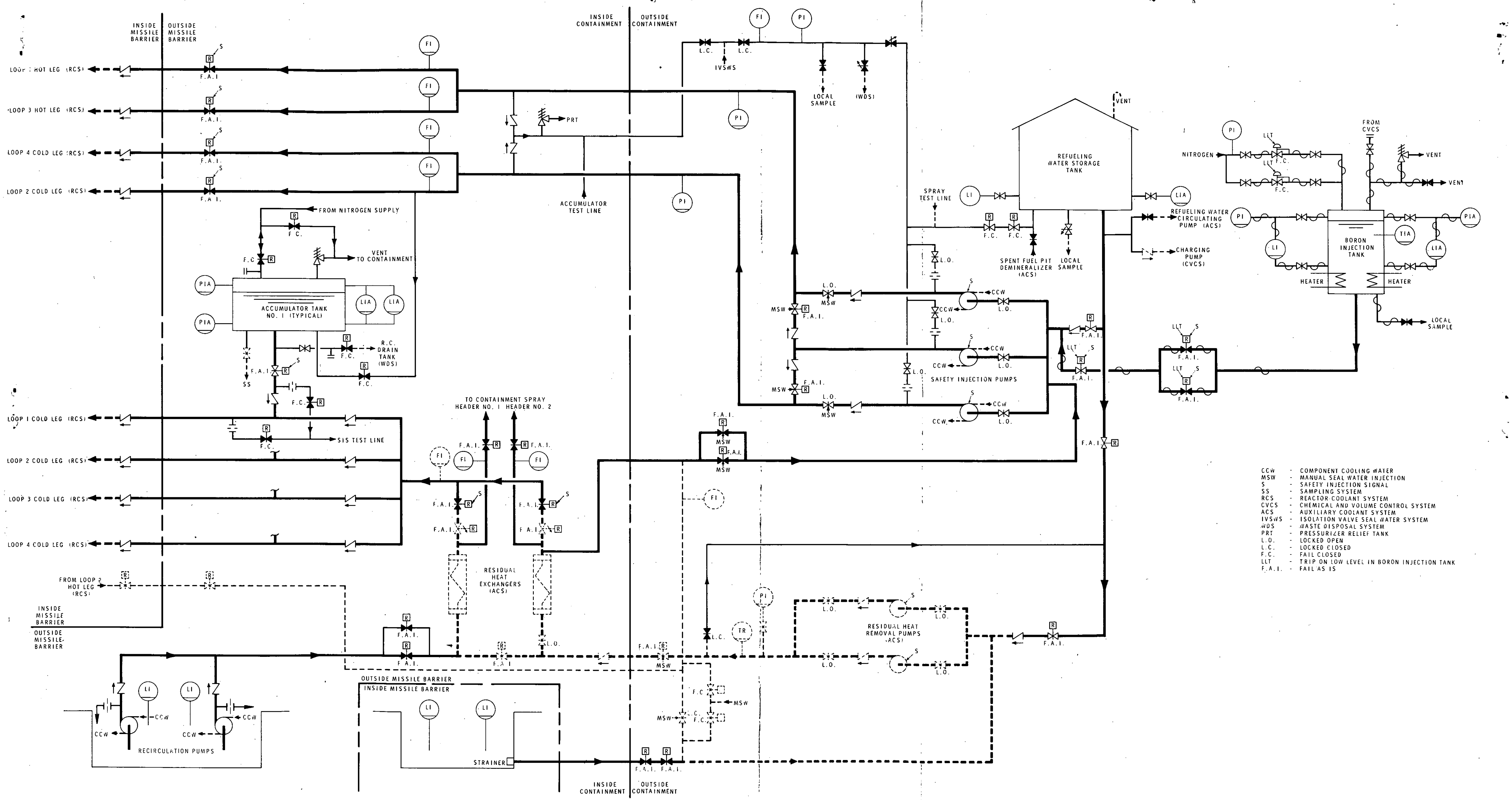
Erection procedures are issued to assure that the equipment and materials are installed correctly and according to design. These procedures include such items as sequences of installation, when necessary, and specifications for welding. Particular attention is paid to methods which are not standard to the construction industry. Included in the welding specifications are non-destructive tests required such as dye penetrant, radiography, and ultrasonic.

Qualified resident supervisory service engineers with training in electrical, civil, mechanical instrumentation, and welding categories, experienced in installation and checkout of piping and equipment at other nuclear sites, review the specifications and procedures. Special attention is given to systems incorporated in the engineered safeguards. These supervisory engineers continually monitor the installation and checkout of the equipment for conformance to the procedures and specifications. During critical phases of the work, they provide personal guidance and record data for future reference. Specified tests are witnessed and accepted or rejected according to the results. If repairs are required, the work is supervised and the subsequent retesting witnessed.

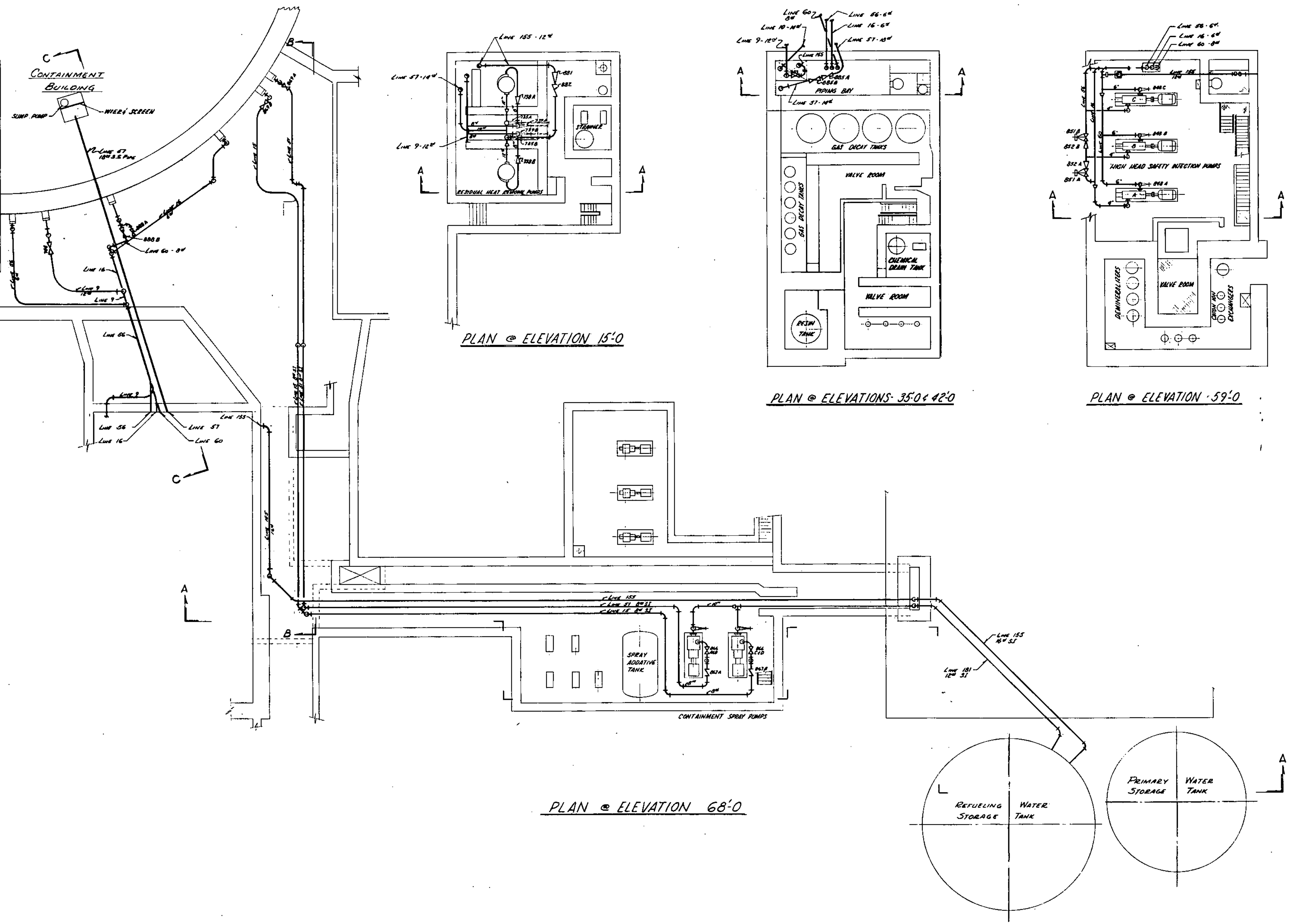
Do

Not

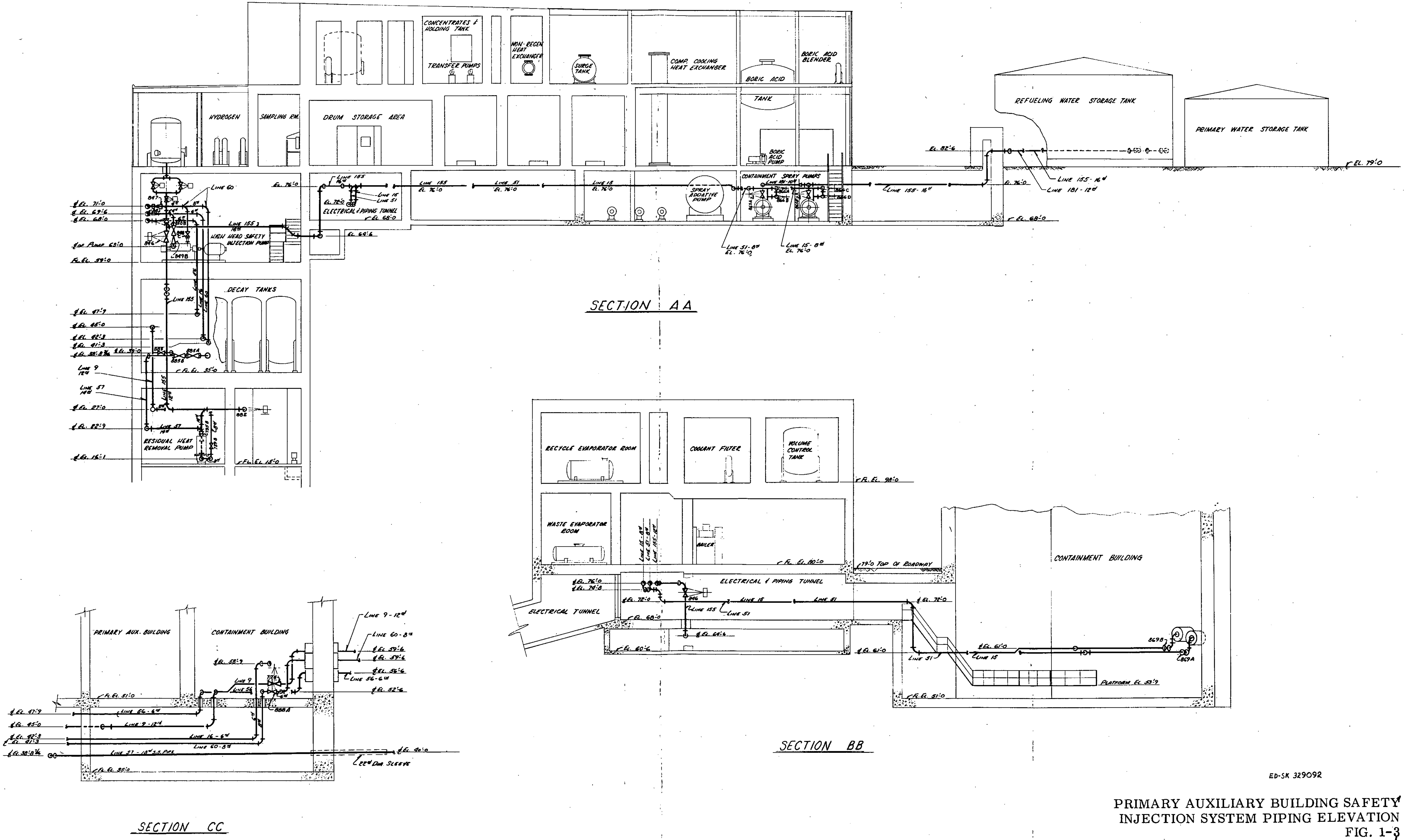
Film



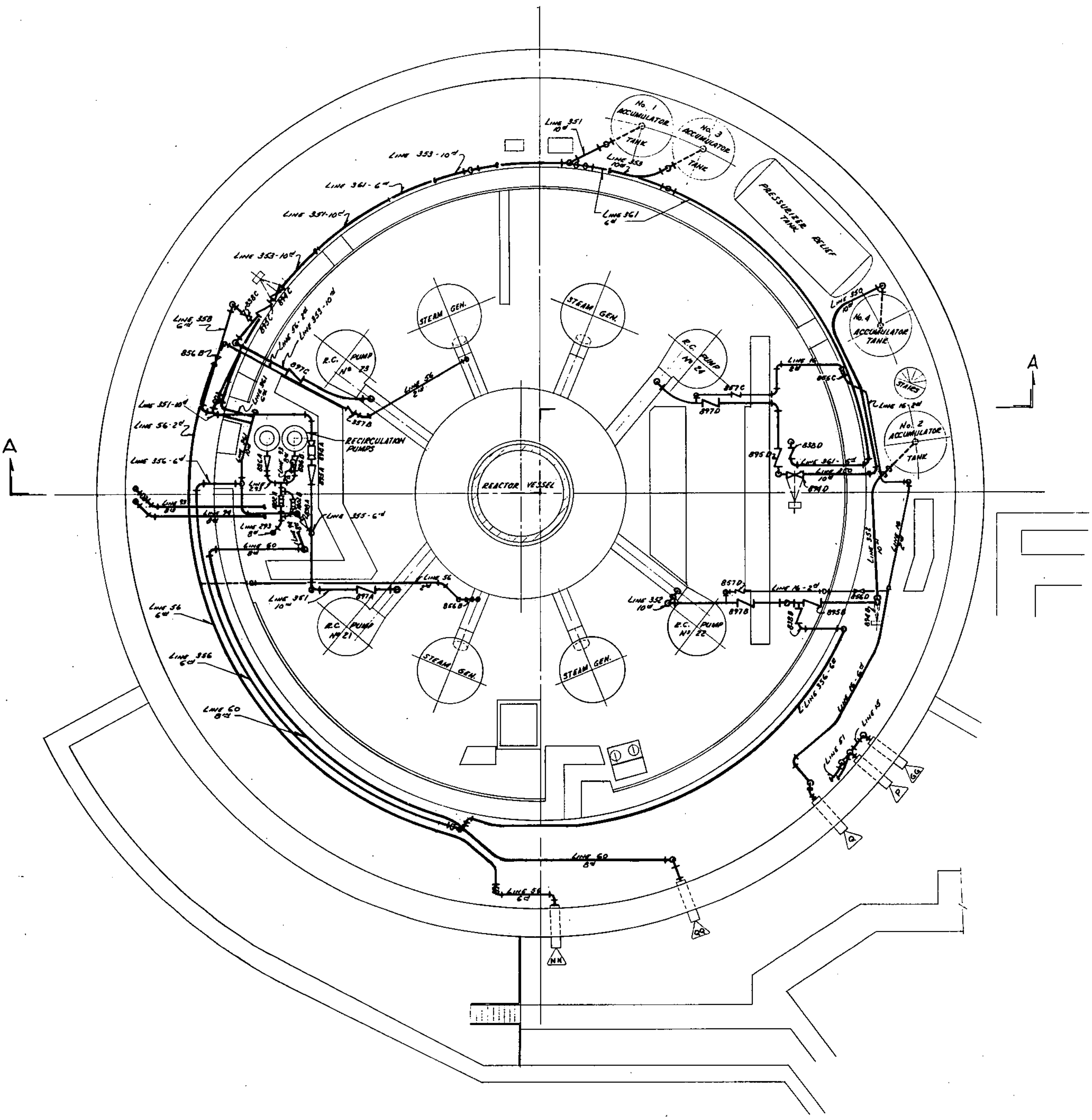
SAFETY INJECTION SYSTEM FLOW DIAGRAM
FIG. 1-1



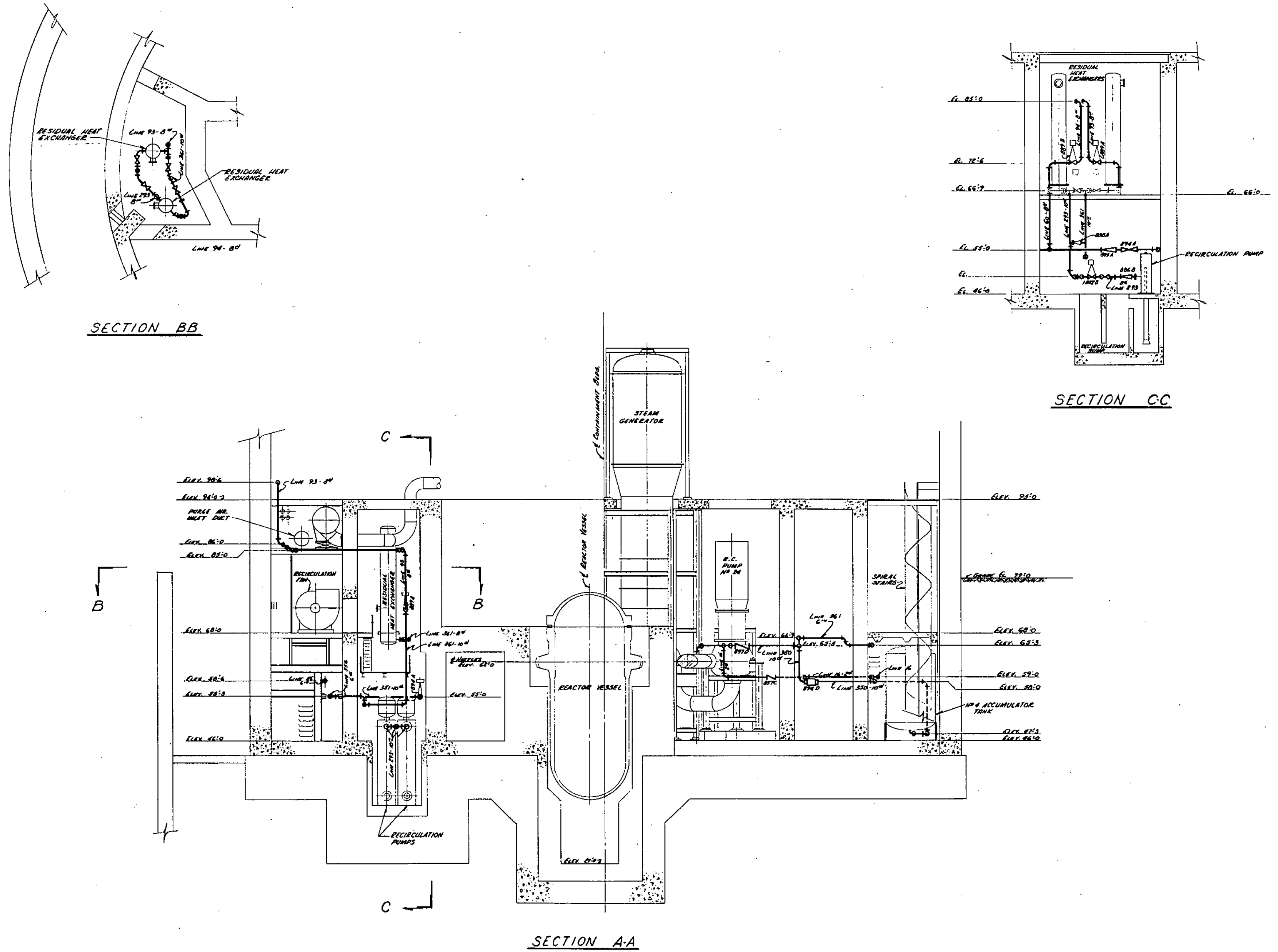
PRIMARY AUXILIARY BUILDING SAFETY INJECTION SYSTEM PIPING PLAN
FIG. 1-2



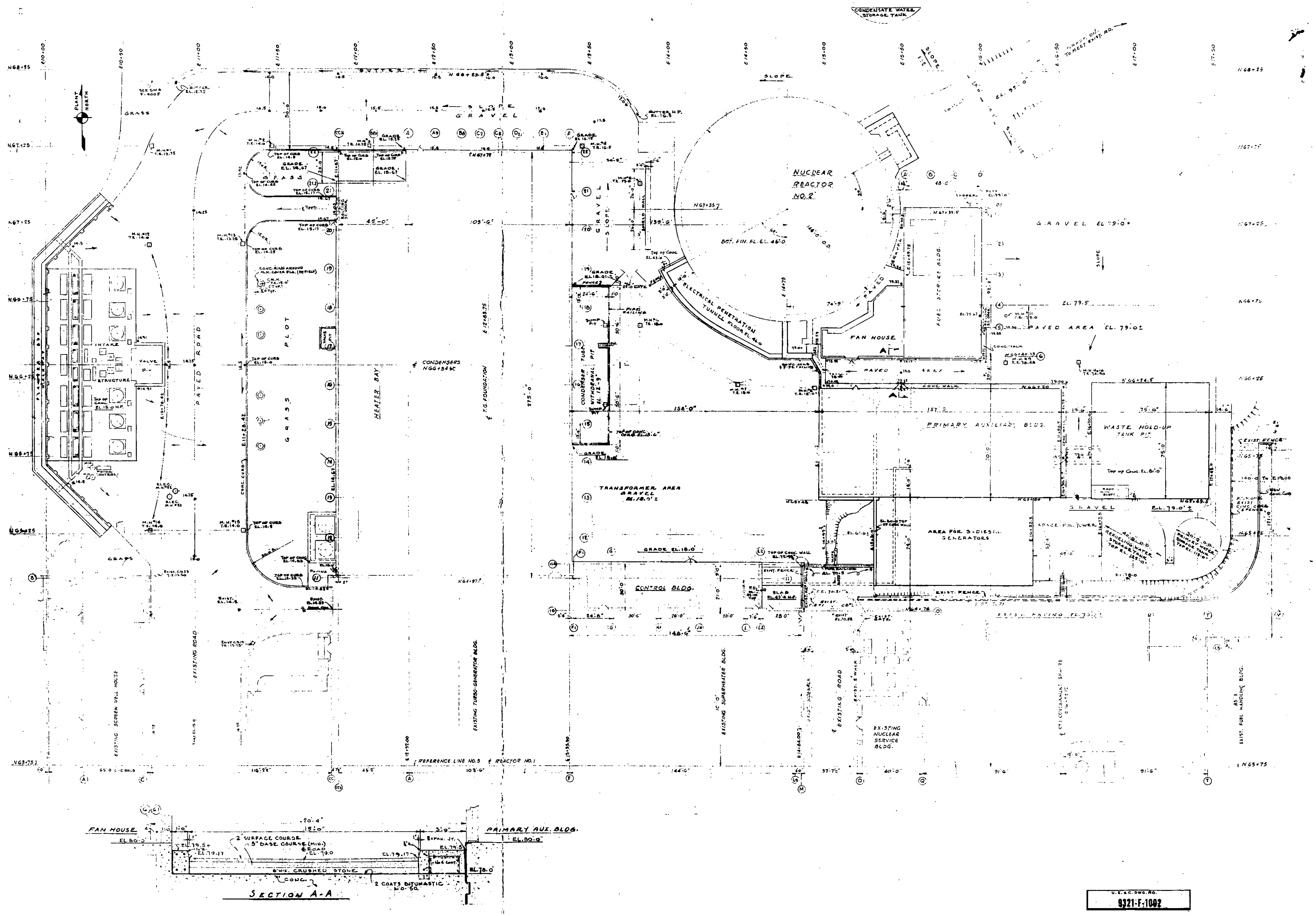
PRIMARY AUXILIARY BUILDING SAFETY INJECTION SYSTEM PIPING ELEVATION
FIG. 1-3



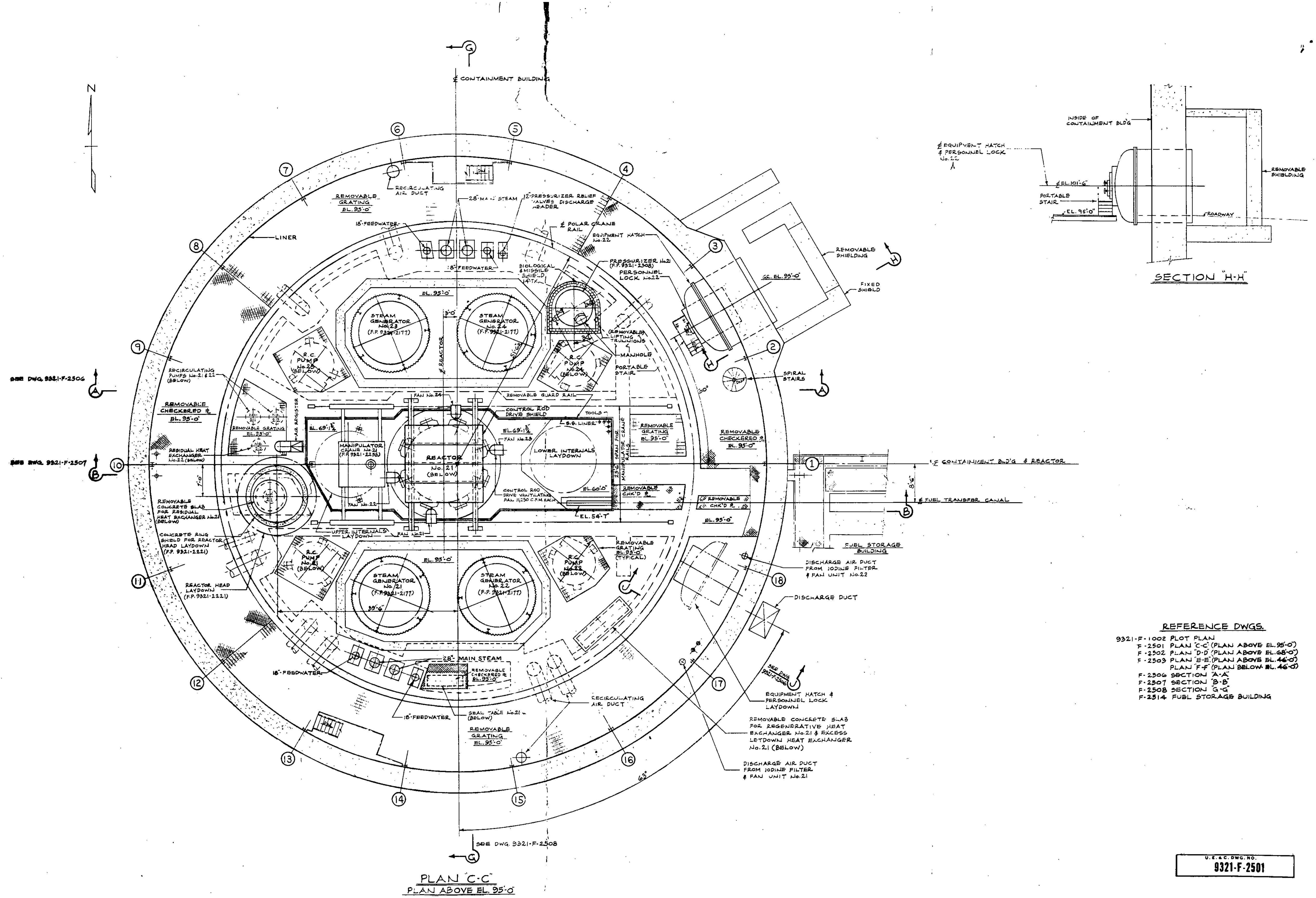
CONTAINMENT BUILDING SAFETY
INJECTION SYSTEM PLAN
FIG. 1-4



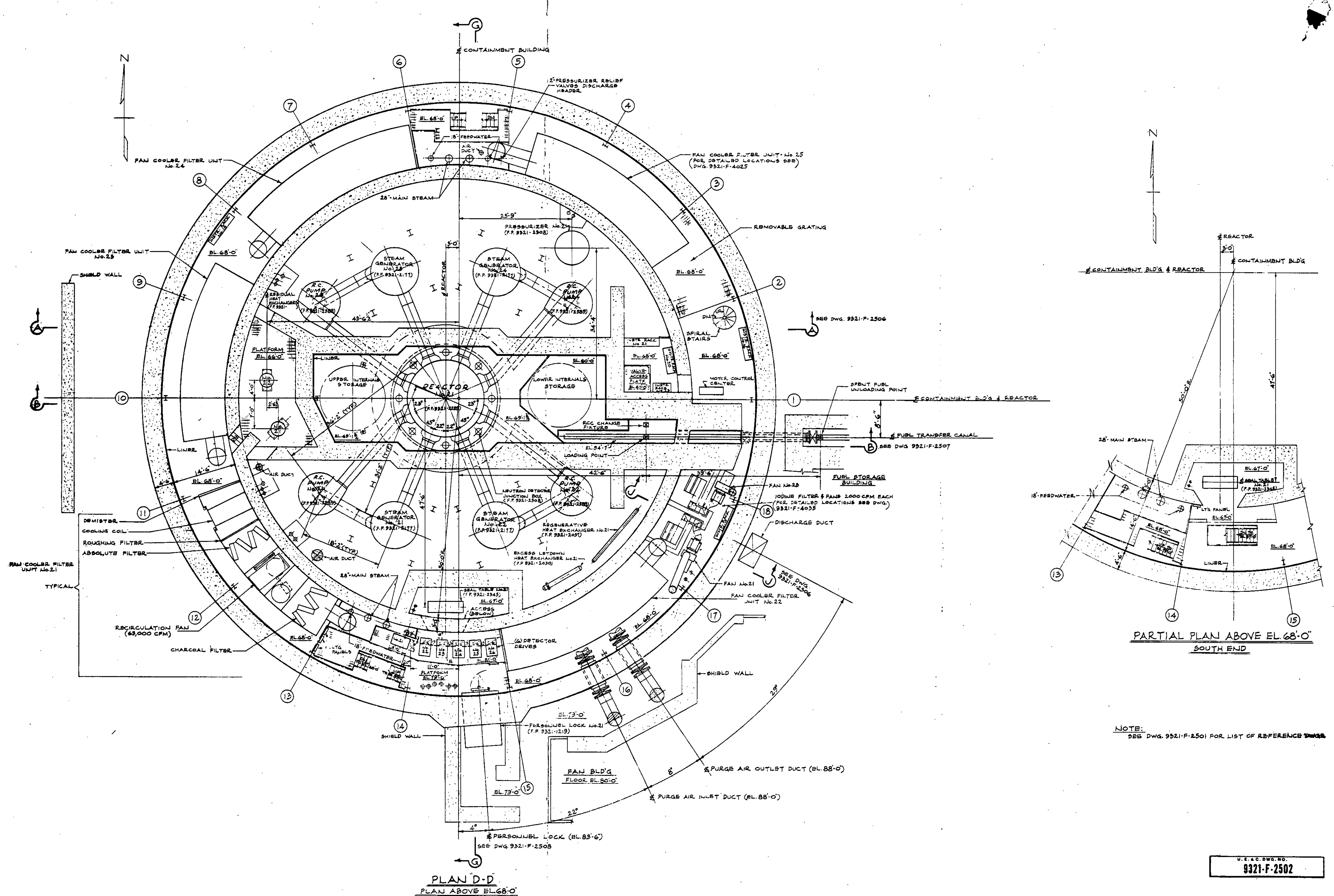
CONTAINMENT BUILDING SAFETY
INJECTION SYSTEM PLAN ELEVATION
FIG. 1-5



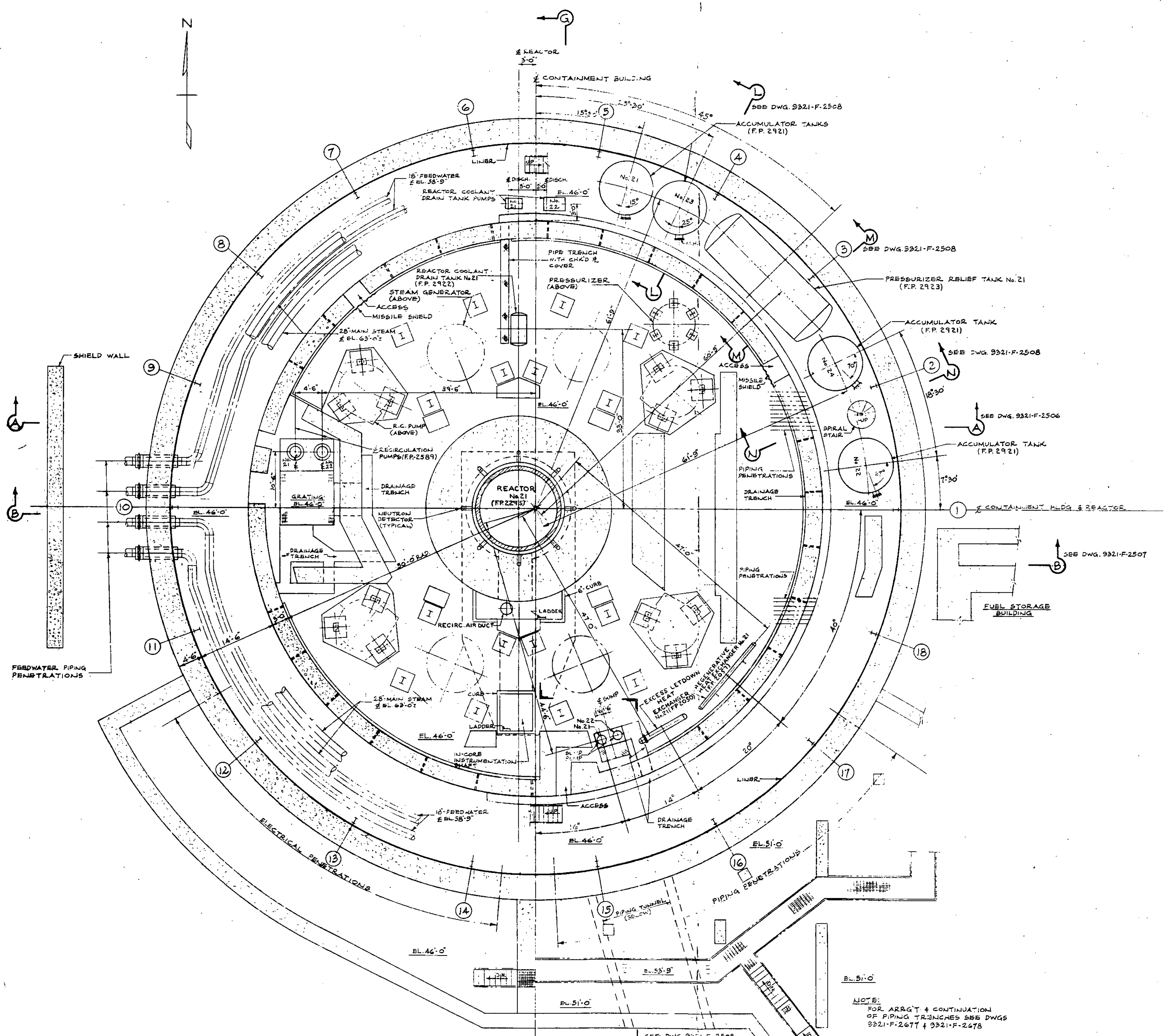
SITE PLOT PLAN
FIG. 1-6



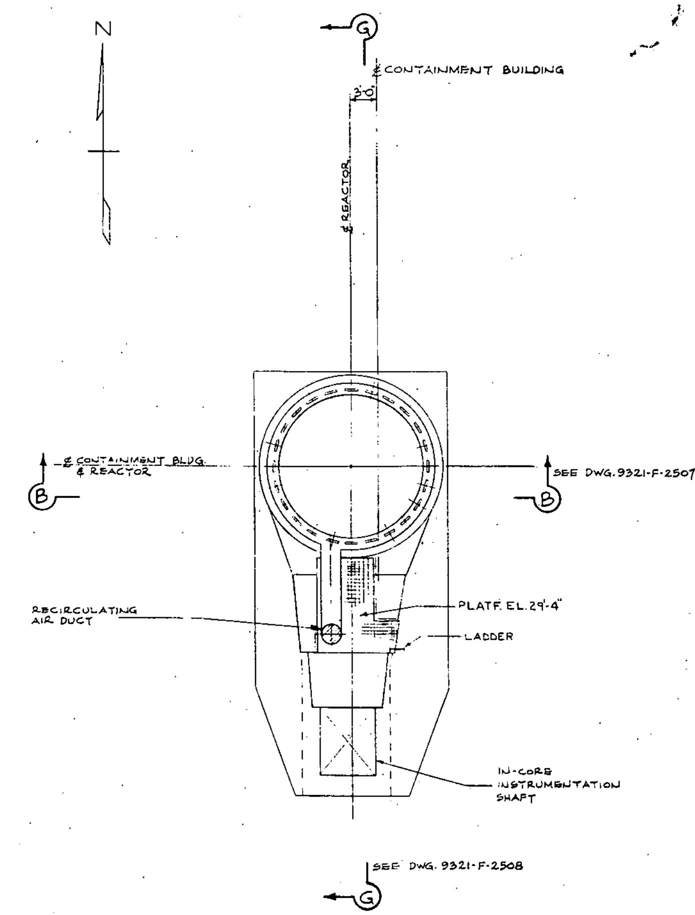
CONTAINMENT BUILDING
GENERAL ARRANGEMENT PLANS SHEET 1
FIG. 1-7



CONTAINMENT BUILDING
GENERAL ARRANGEMENT PLANS SHEET 2
FIG. 1-8



PLAN "E-E"
PLAN ABOVE EL. 46'-0"

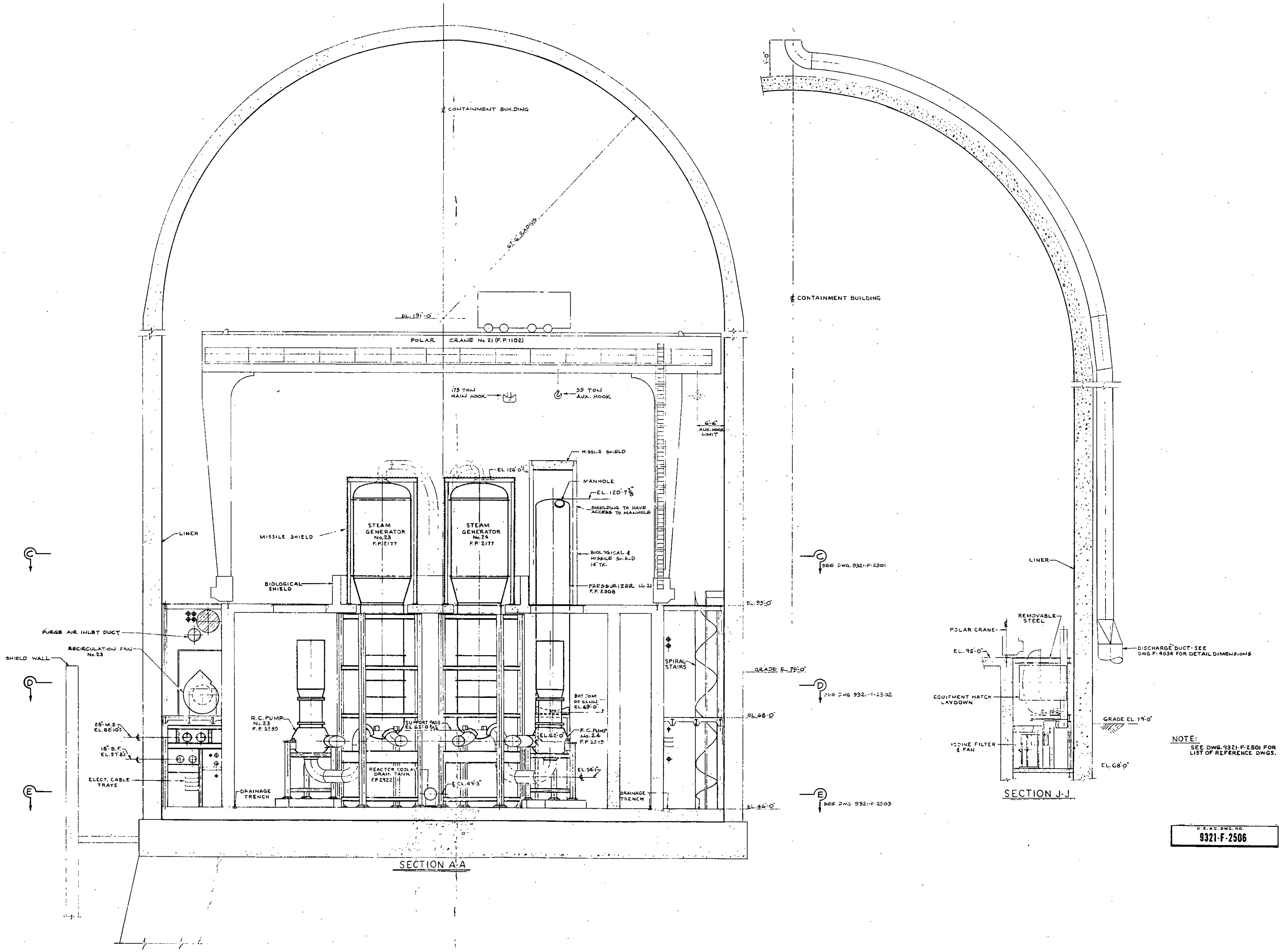


PLAN "F-F"
PLAN BELOW EL. 46'-0"

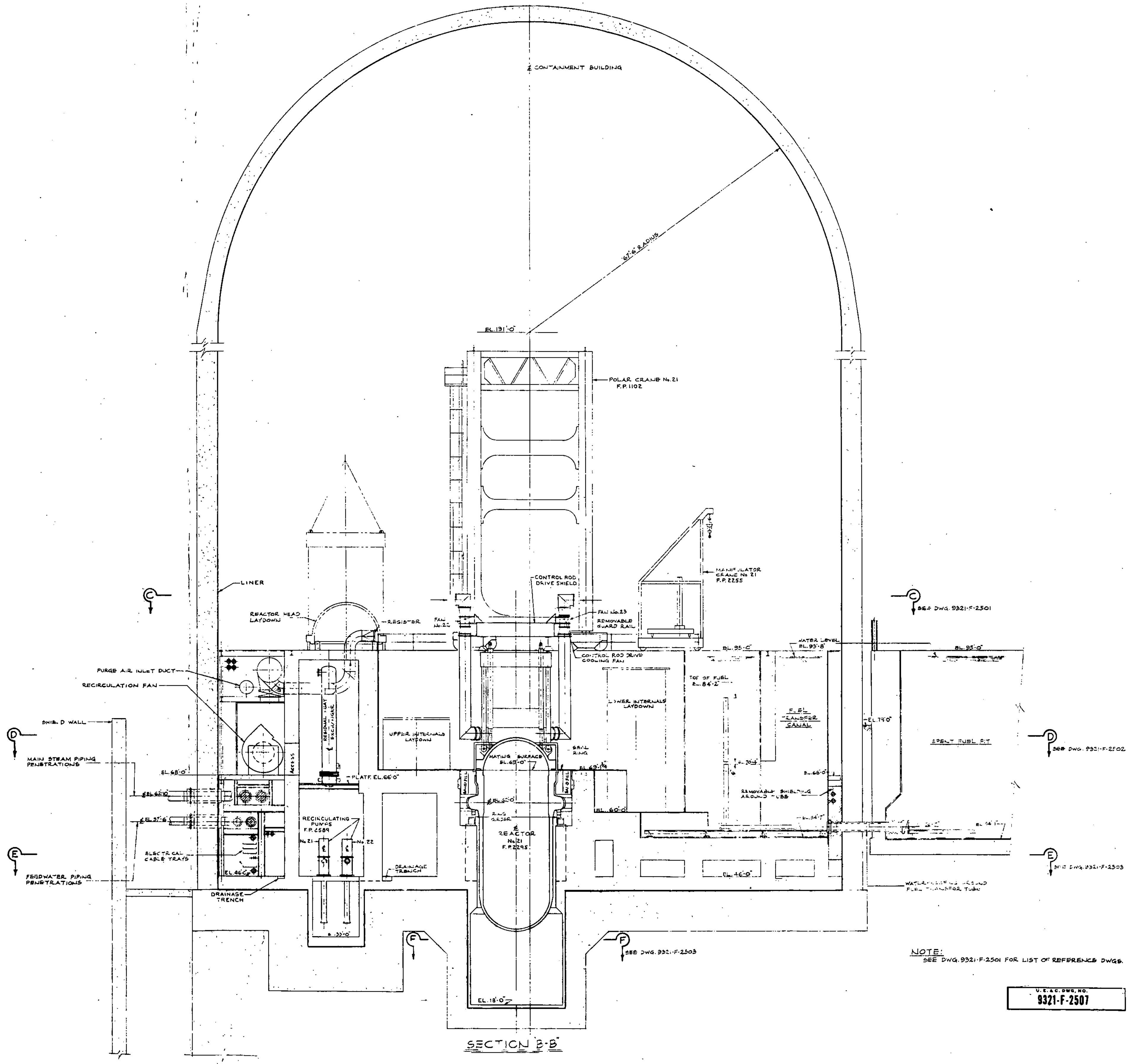
NOTE:
SEE DWG 9321-F-2501 FOR LIST OF REFERENCE DWGS.

U.E.&C. DWG. NO.
9321-F-2503

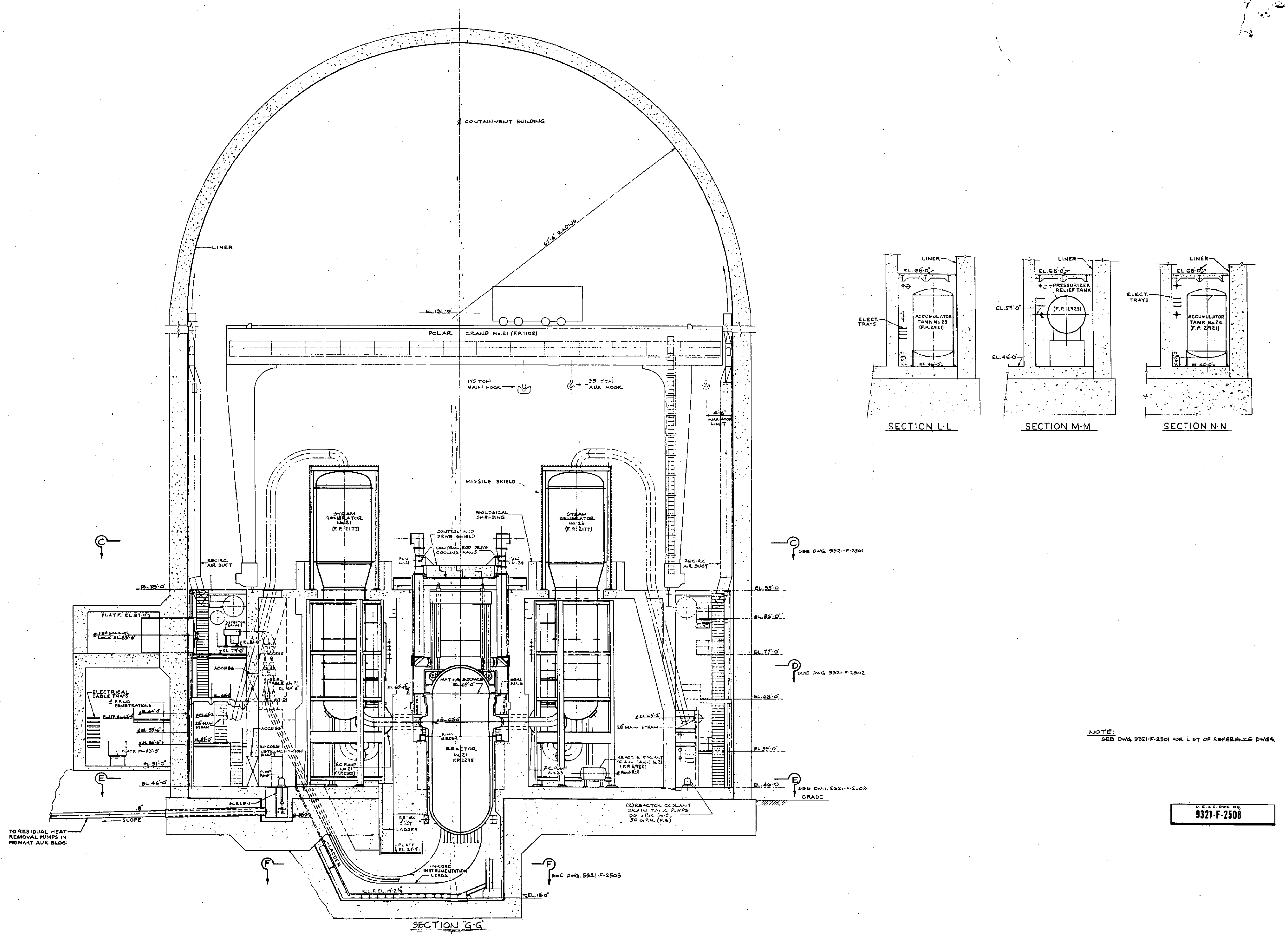
CONTAINMENT BUILDING
GENERAL ARRANGEMENT PLANS SHEET 3
FIG. 1-9



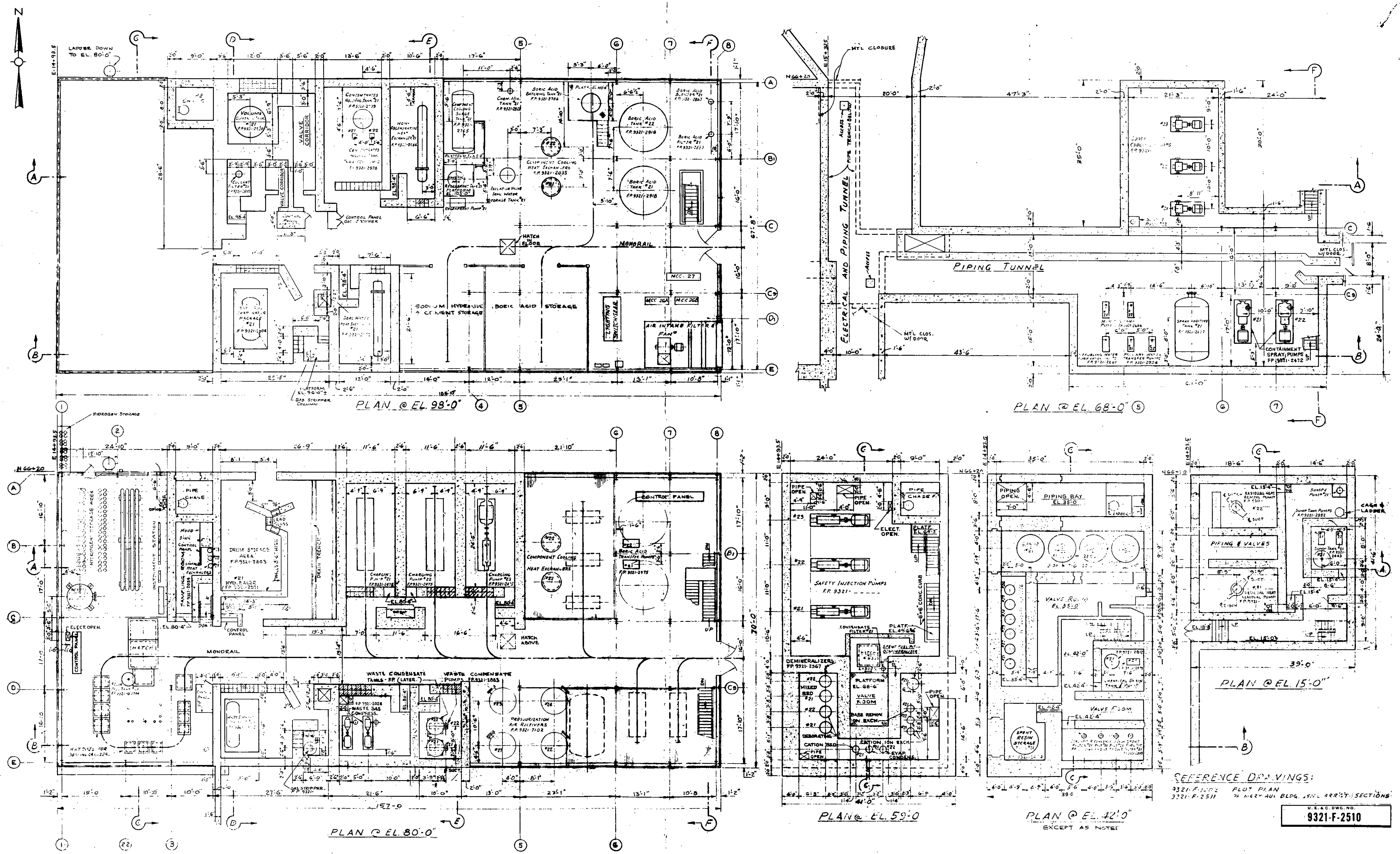
CONTAINMENT BUILDING
GENERAL ARRANGEMENT ELEVATION SHEET 1
FIG. 1-10



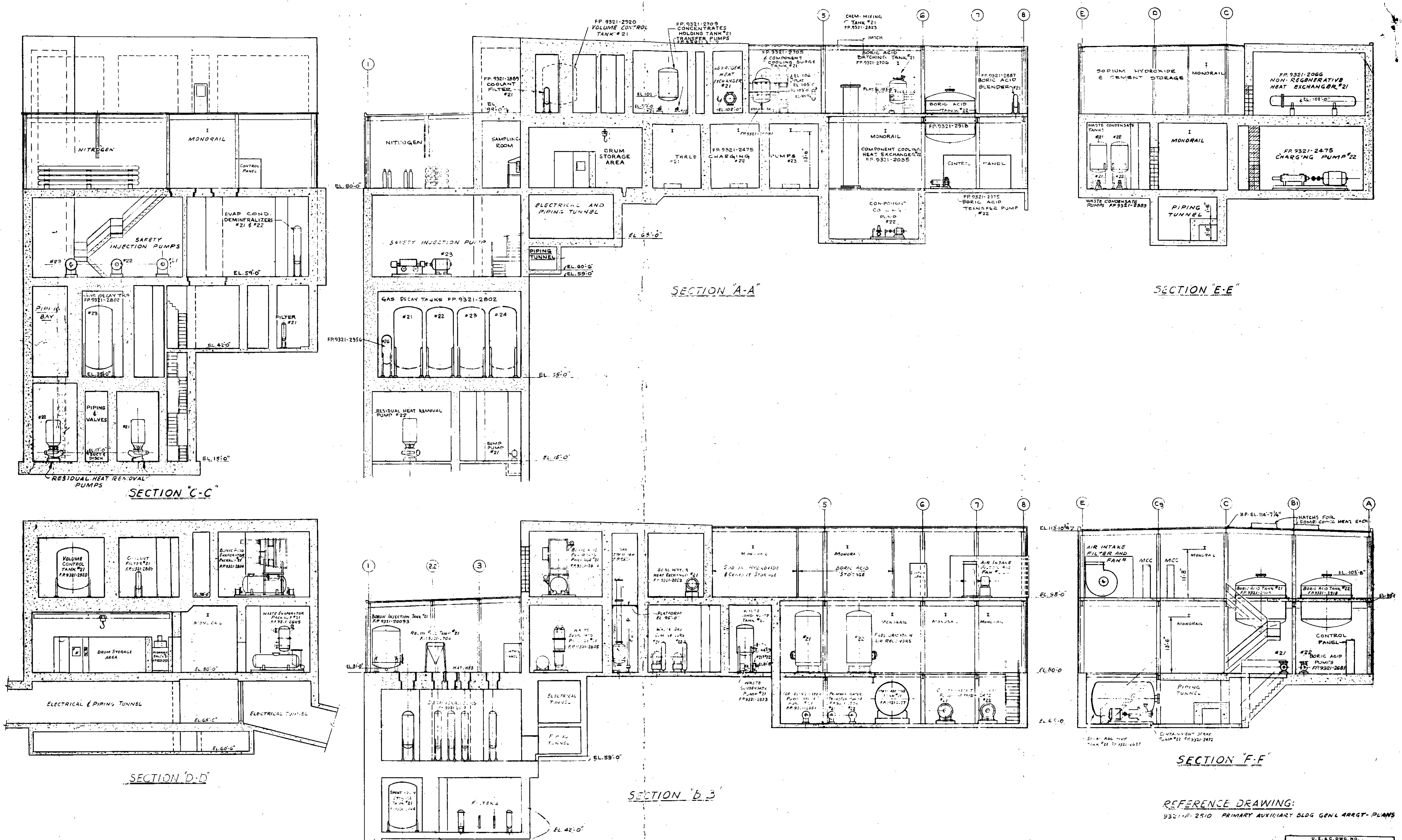
CONTAINMENT BUILDING
GENERAL ARRANGEMENT ELEVATION SHEET 2
FIG. 1-11



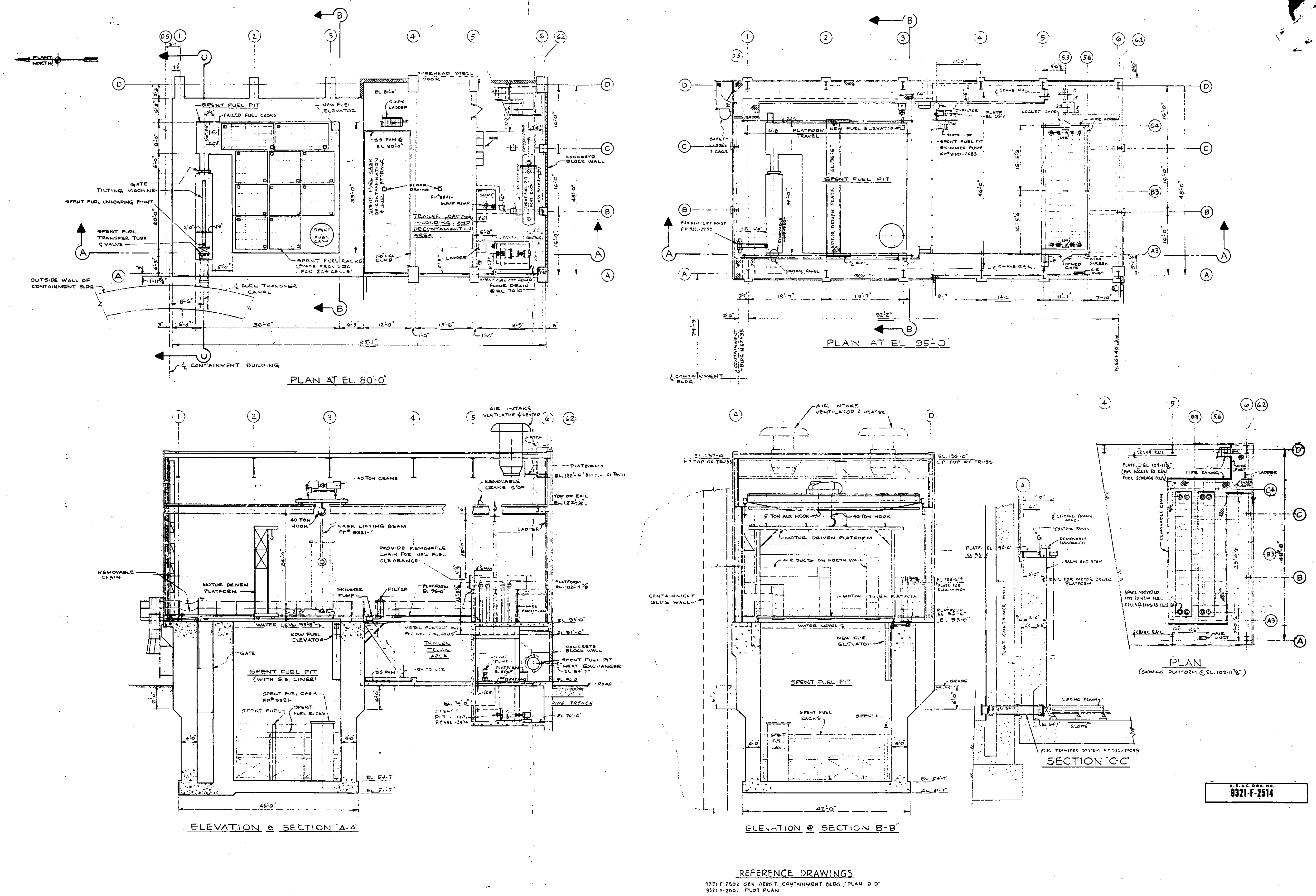
CONTAINMENT BUILDING
GENERAL ARRANGEMENT ELEVATION SHEET 3
FIG. 1-12



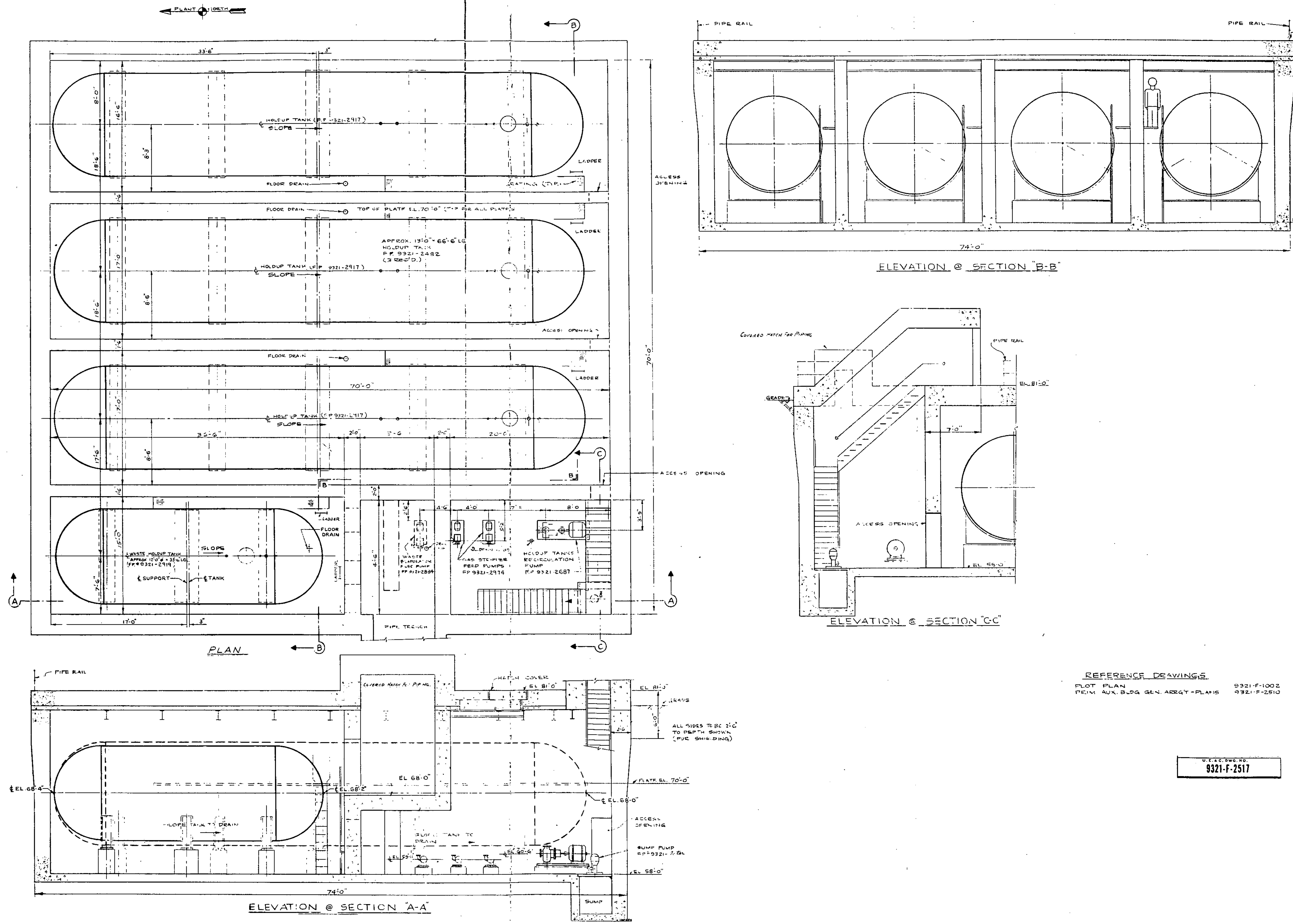
PRIMARY AUXILIARY BUILDING
GENERAL ARRANGEMENT PLANS
FIG. 1-13



PRIMARY AUXILIARY BUILDING
GENERAL ARRANGEMENT ELEVATION
FIG. 1-14



SPENT FUEL PIT BUILDING GENERAL ARRANGEMENT
FIG. 1-15



HOLDUP TANK BUILDING GENERAL ARRANGEMEN
FIG. 1-16

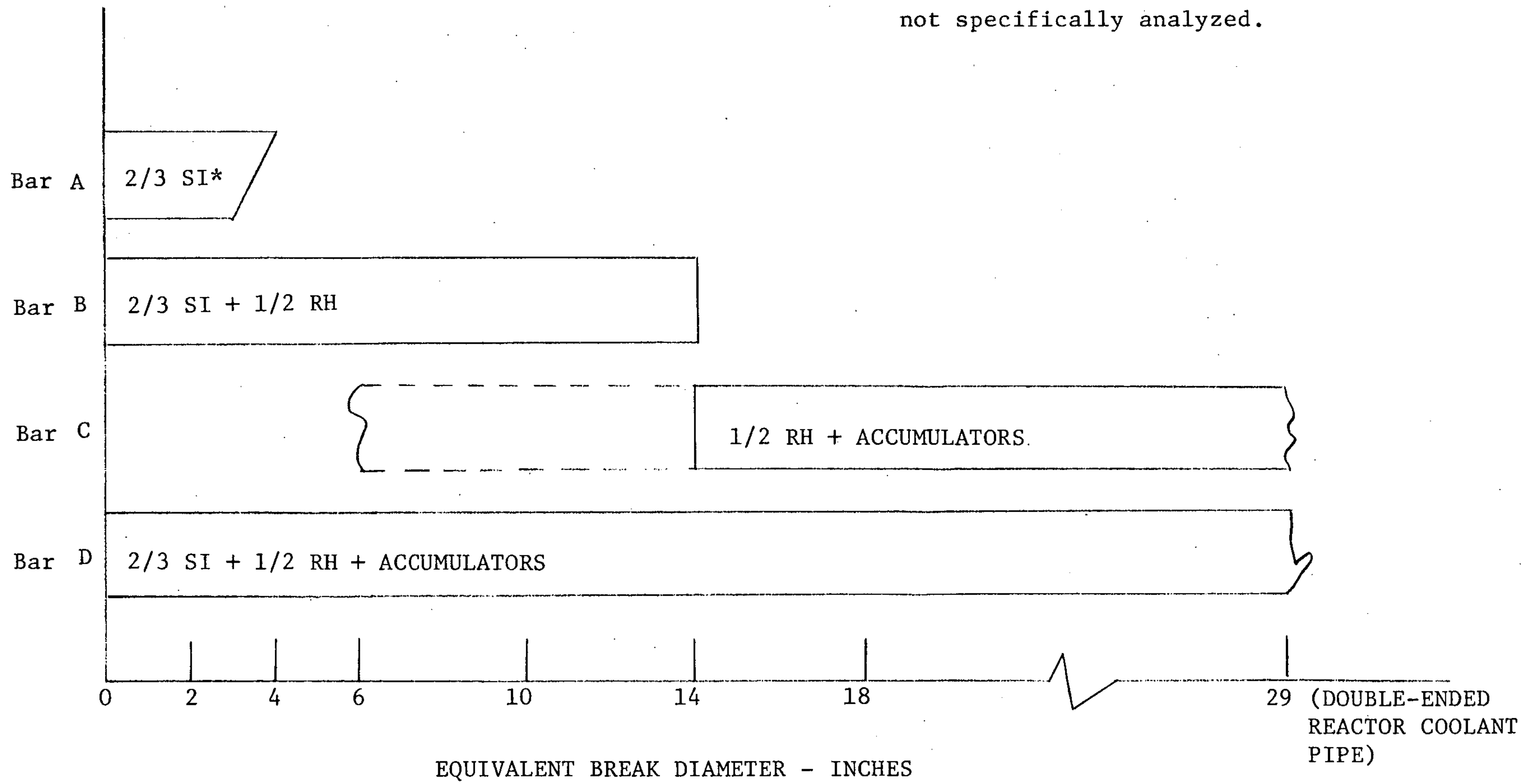
2/3 SI = TWO OF THREE SAFETY INJECTION PUMPS

1/2 RH = ONE OF TWO RESIDUAL HEAT REMOVAL PUMPS

CORE PROTECTION

Solid bar indicates capacity to meet core cooling criterion of no clad melting.

Dashed lines indicate expected performance not specifically analyzed.



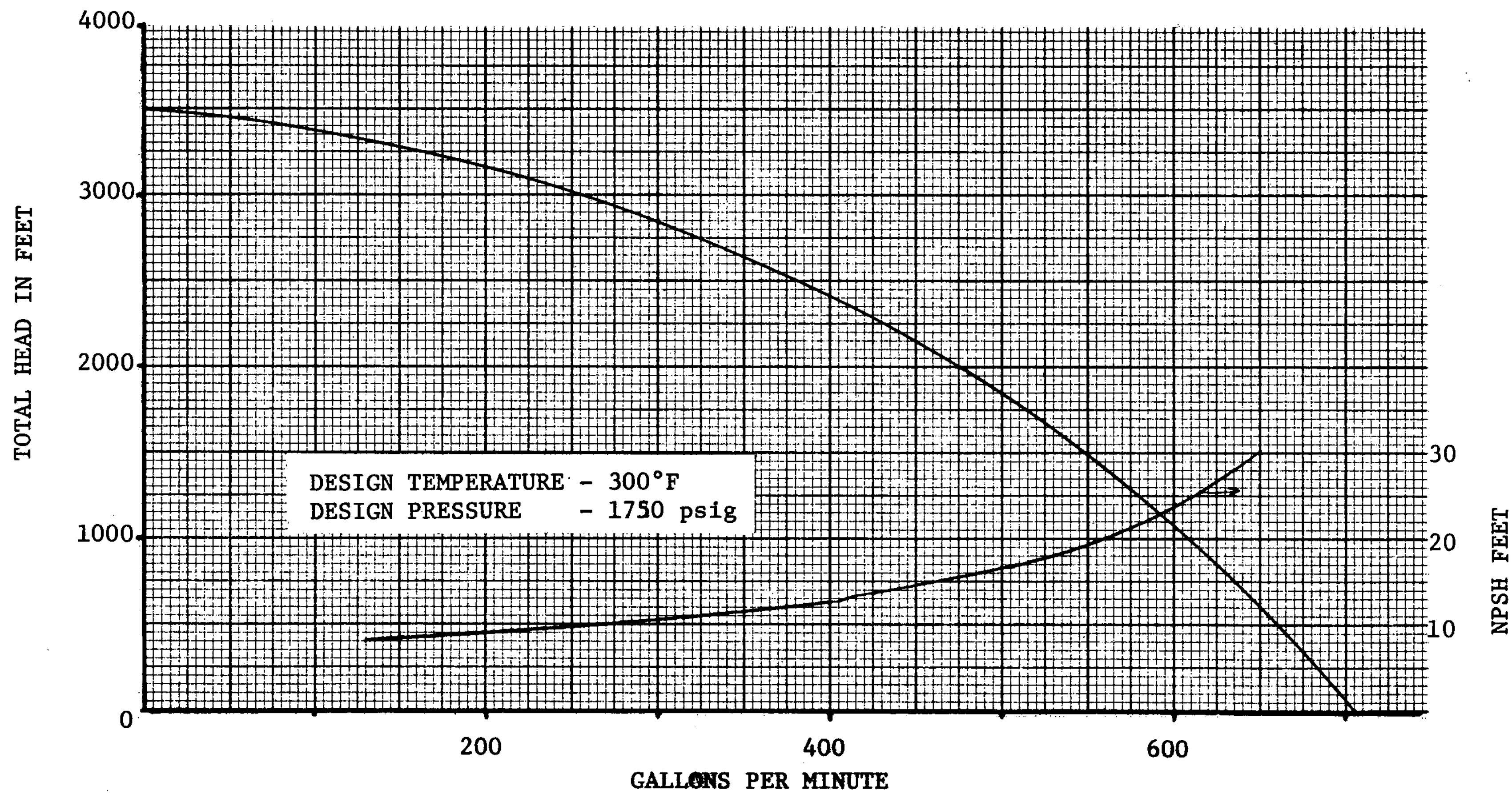
NOTE: FOR ALL CASES ONE OF TWO RECIRCULATION PUMPS REQUIRED FOR RECIRCULATION

RANGE OF PROTECTION BY SAFETY INJECTION SYSTEM

FIG. 1-17

INDIAN POINT #2

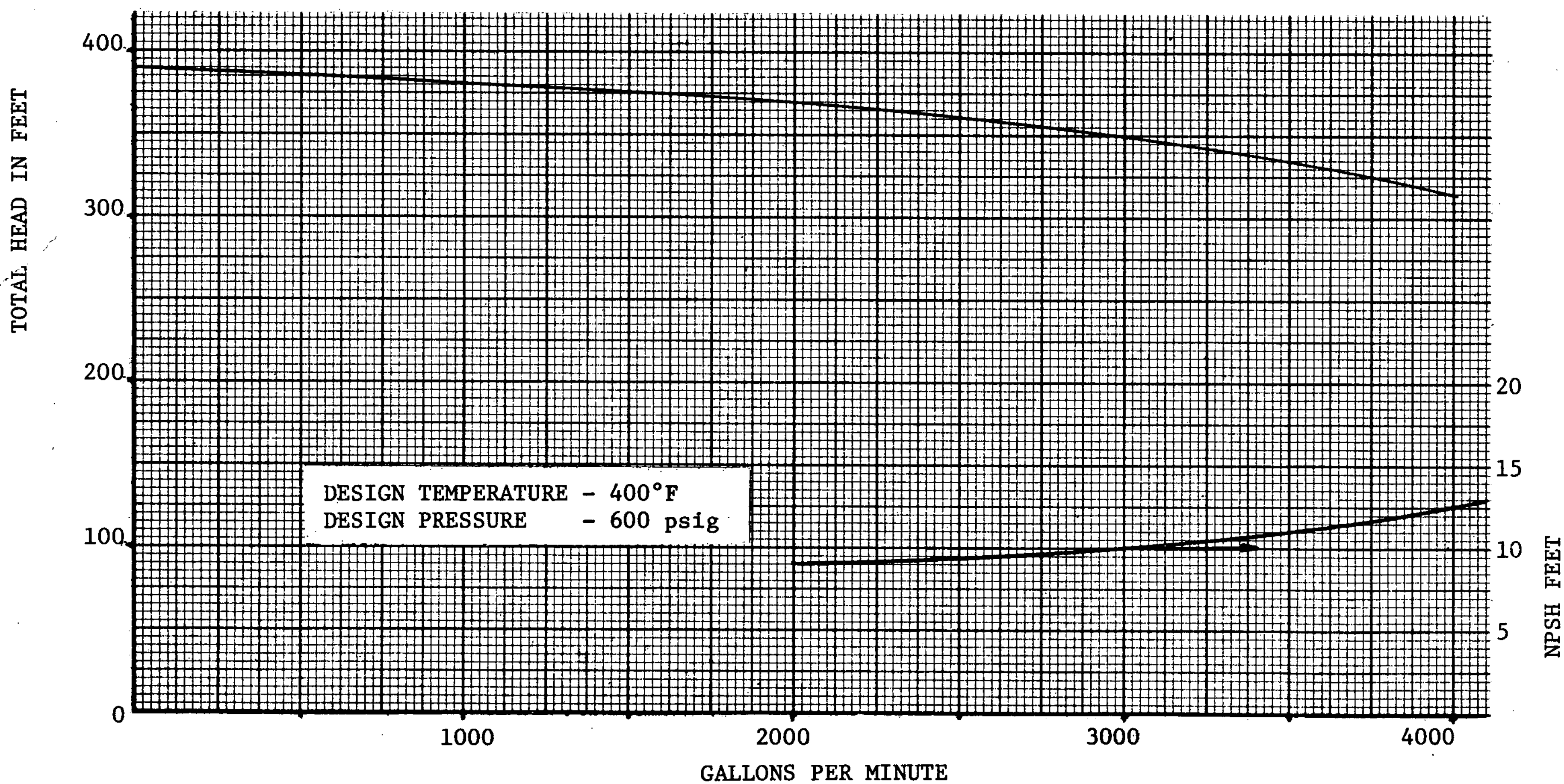
SAFETY INJECTION PUMP PERFORMANCE



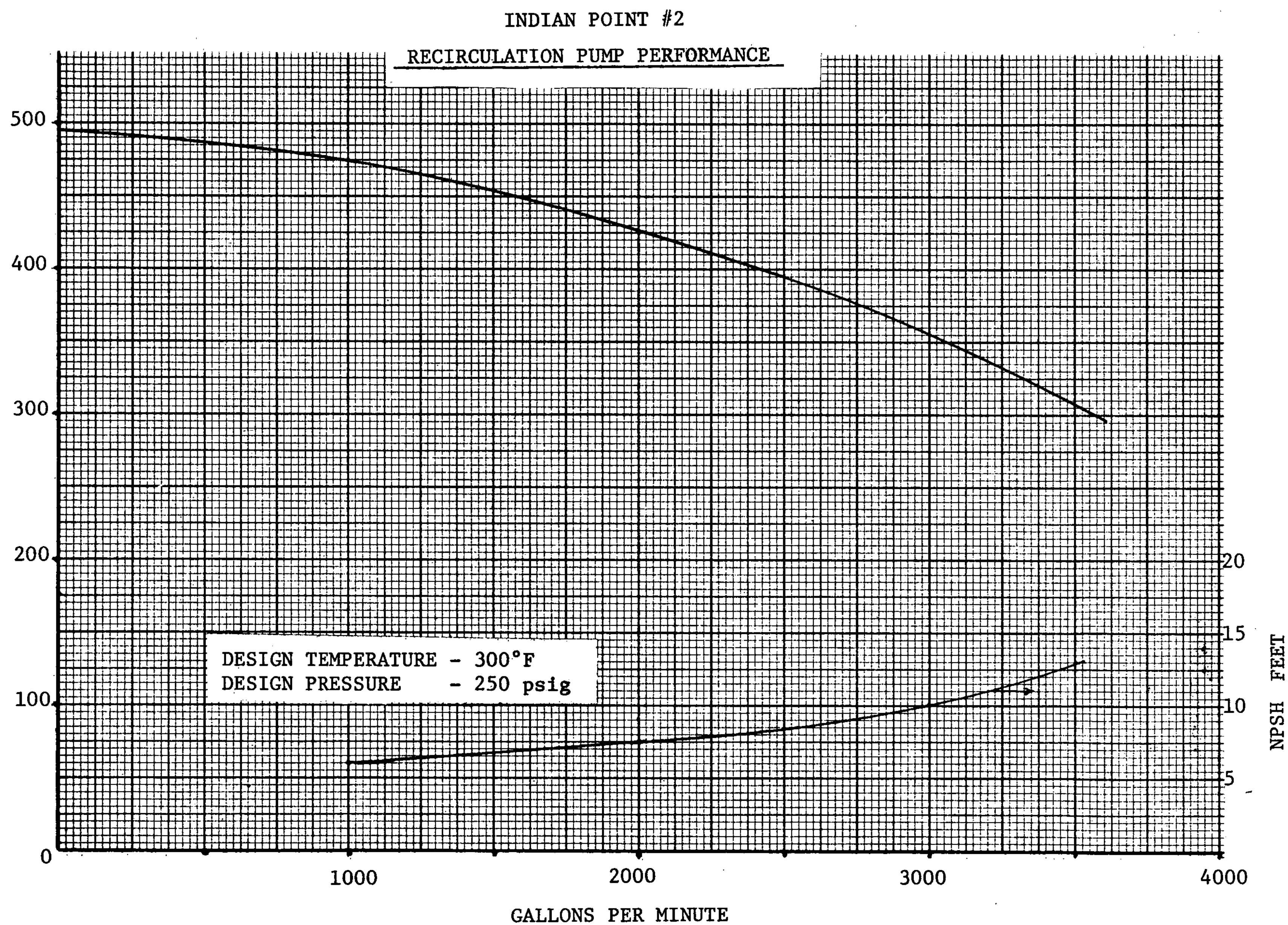
SAFETY INJECTION PUMP PERFORMANCE
FIG. 1-18

INDIAN POINT #2

RESIDUAL HEAT REMOVAL PUMP PERFORMANCE



RESIDUAL HEAT REMOVAL PUMP PERFORMANCE
FIG. 1-19



RECIRCULATION PUMP PERFORMANCE
FIG. 1-20

2. REACTOR INTERNALS

2.1 DESIGN

The reactor internals are described on pages 3.2-46 through 3.2-58 of the Preliminary Safety Analysis Report. A schemetic of the reactor vessel and internals is shown in Figure 2-1 which identifies the major components. The principal structural members of the internals are the lower and upper core support structures shown on Figure 2-2 and 2-3 respectively. These structures are constructed from type 304 stainless steel.

The fuel assembly construction is described on pages 3.2-51 through 3.2-56 of the PSAR. The principal structural members of the fuel assembly are the top and bottom nozzles, the control rod guide thimbles, and the fuel assembly grids as shown on Figure 2-4. All components of the top nozzle are type 304 stainless steel except for the leaf springs and bolts which are Inconel 718 and Inconel 600 respectively. The spring clip grids are Inconel 718, the control rod guide thimble are type 304 stainless.

Since issuance of the PSAR, the number of grids per fuel assembly has been increased from 8 to 9. The effect is to increase the structural rigidity of the fuel assembly.

2.2 REACTOR INTERNALS DESIGN CRITERIA

2.2.1 Normal Operation

The internals and core are designed for normal operation conditons subjected to loads of mechanical, hydraulic and thermal origin. The response of the structure under the so called "design earthquake" is included in this category.

The stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Section 4 has been adopted as a guide for the design of the internals and core with exception of the fuel rod cladding. Seismic stresses are combined in the most conservative way and are considered primary stresses.

The members are designed under the basic principles of: (1) Maintaining distortions within acceptable limits, (2) Keeping the stress levels within acceptable limits, and (3) Prevention of fatigue failures.

To assure the performance of the parts, the intent of Section III of the ASME Code is followed.

2.2.2 Abnormal Operation

The criteria for acceptability blowdown effects due to a pipe break are that the reactor should be able to scram and that the safety injection system for core coling will be able to operate as designed. Consequently, the limitations established on the internals for these types of loads are concerned principally with the maximum allowable deflections. The deflection criteria for structures that are critical are as follows:

	Calculated	Allowable	No Loss of Function
Upper Barrel (expansion/compression)	.072"	3.0	6"
Upper Package (Axial Deflection)	.120"	1.0"	2"
RCC Guide Tube (cross section distortion)	0.	.035"	.072"
(deflection as a beam)	.2	1.0"	1.5"
Fuel Assembly Thimbles (cross section distortion)	0.	.035"	.072"

2.3 EFFECT OF BLOWDOWN FORCES AND PRESSURES ON CORE AND INTERNALS

It is the intent of the following paragraphs to clarify the information presented in Section 3.3 of the Sixth Supplement by referencing the location of the stresses and deflections presented in that document for blowdown forces following a loss of coolant accident.

1. Core

As shown in Figures 3-7a and 3-7b of Supplement Six, the pressure difference across the core for a hot leg break is oscillatory for the first 2 seconds, and predominently upward leveling off to about 50 to 75 psi. The initial peak is largest and causes a compressive force of 6700 pounds per fuel assembly, Figure 2-4. This load is transferred to the upper core plate and the the upper core support structure.

Compressive force on the fuel assembly during a cold leg break is 9900 pounds and is transferred downward to the lower core plate and to the lower core support structure. In comparison to the above compressive loads the force to buckle a fuel assembly is on the order of 85,000 pounds. The effect of the above loadings on the upper core support structure, the lower core support structure, and the control rod guide thimbles are discussed in paragraphs 2, 3 and 6 below respectively.

2. Upper Core Support Structure

For the hot leg break, the initial transient causes an 0.12 inch maximum upward deflection at the upper support plate and a maximum total stress of approximately 12,000 psi in the upper support plate ligaments and grid. After 2 seconds the force on the upper core support structure becomes approximately constant (each fuel assembly exerts an upward force of approximately 2920 pounds) and the above deflection and stress reduce to 0.057 in. and 5200 psi. The location of the maximum deflection and total stress are indicated on Figure 2-3. For the cold leg break there is no significant effect on the upper core support structure.

3. Lower Core Support Structure

The maximum total stress occurs at the lower girth weld and is approximately 15,000 psi for a hot leg break and 11,000 psi for a cold leg break. The lower girth weld is located on Figure 2-2.

4. Thermal Shield

The thermal shield is rigidly connnected to the core barrel as shown on Figure 2-5 and is not affected by either a hot or cold leg break.

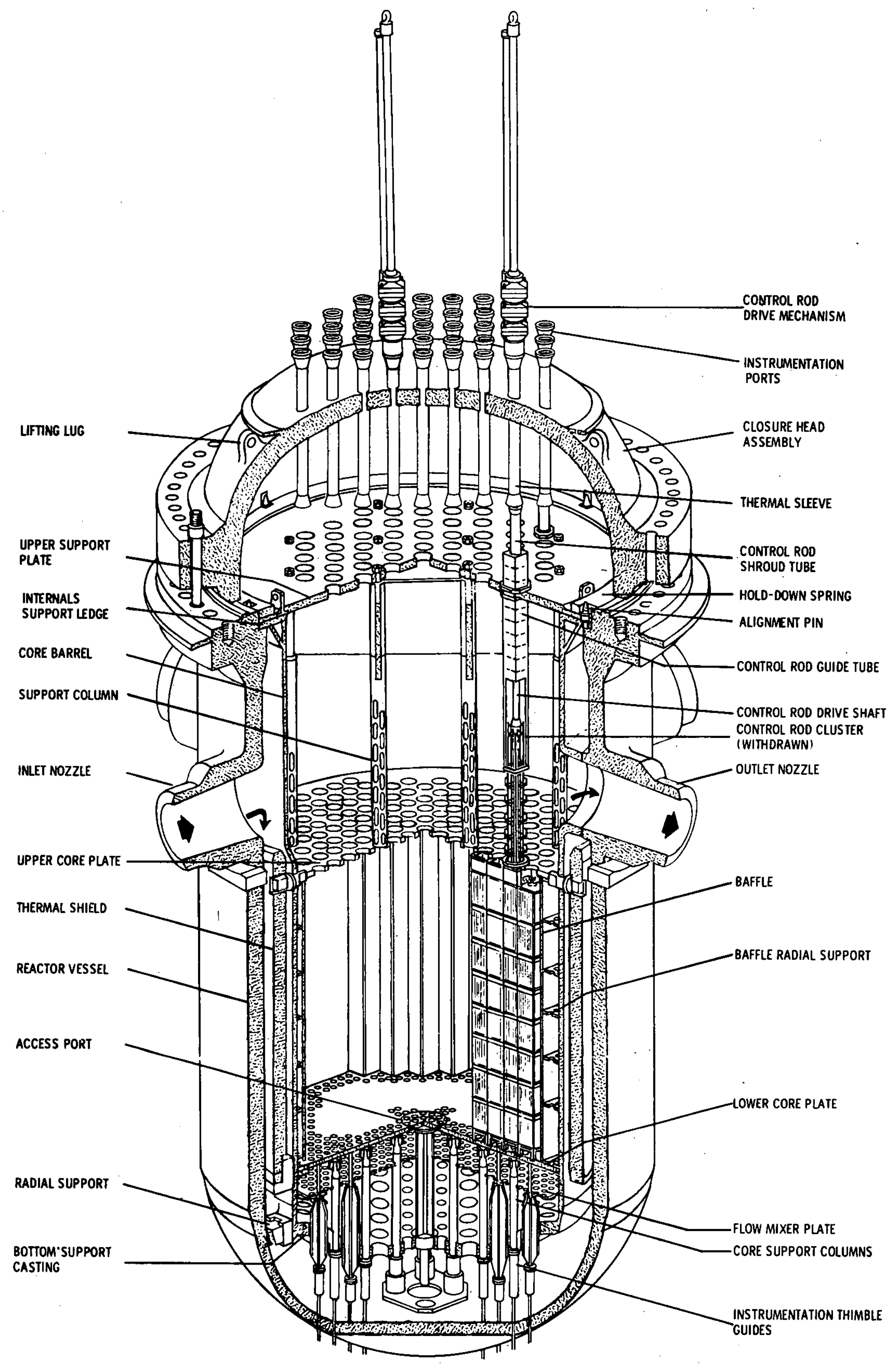
5. Upper Core Barrel

The pressure across the upper barrel is oscillatory during the first 1/2 second following the break. The differential pressure across the core barrel various between ± 400 psi at a frequency of 15 cps for a hot leg break between ± 600 psi at a frequency of 17 cps for a cold leg break. These frequencies are low compared to the natural frequency of the upper core barrel and the corresponding hoop stress varies between ± 13,500 and ± 20,300 psi respectively. The upper core barrel is identified on Figure 2-2.

6. RCC Guide System

Compressive stresses of approximately 16,800 psi occur in the fuel assembly guide thimbles, Figure 2-4 following a hot or cold break. Cross flow to the outlet nozzles will result in a 6,000 psi maximum bending stress in the guide column near the outlet nozzle after a hot leg break as indicated on Figure 2-7.

In each case the stresses and deformations following either a hot or cold leg break are less than those which would adversely effect the integrity of the structures.



REACTOR VESSEL AND INTERNALS
FIG. 2-1

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U. S. Atomic Energy Commission

Docket No. 50-247

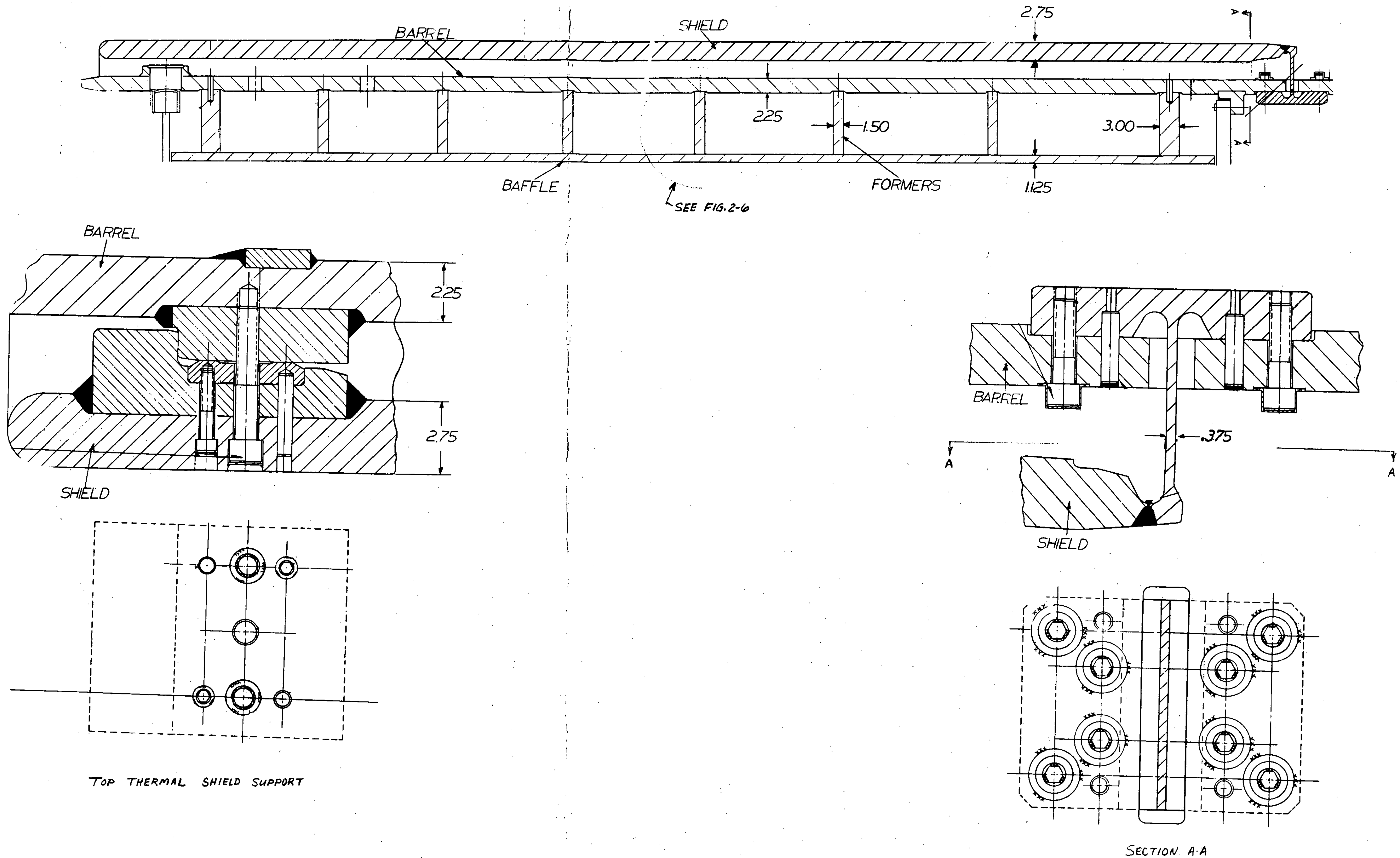
Exhibit B-7

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

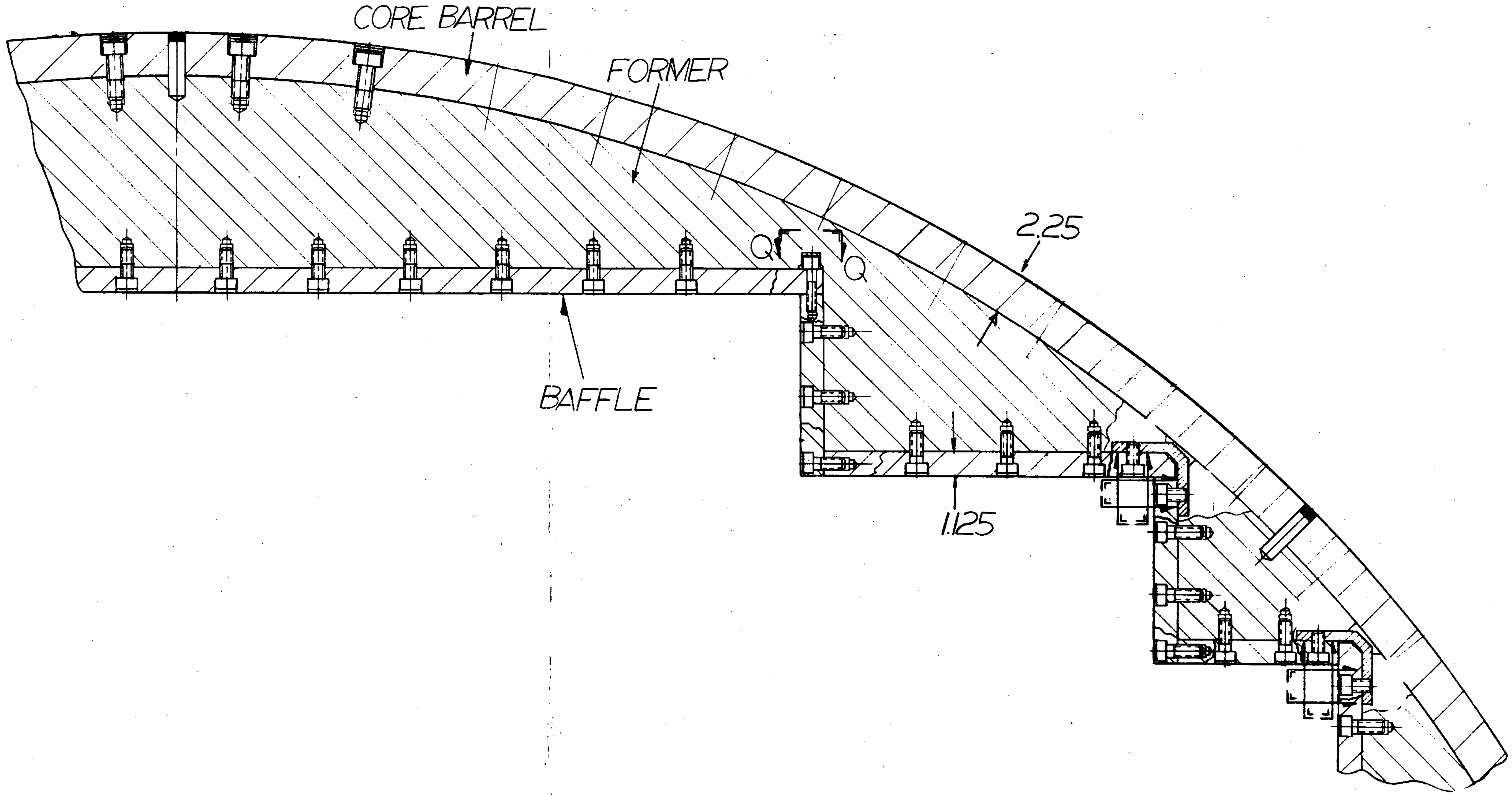
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

SEVENTH SUPPLEMENT TO:

PRELIMINARY SAFETY ANALYSIS REPORT

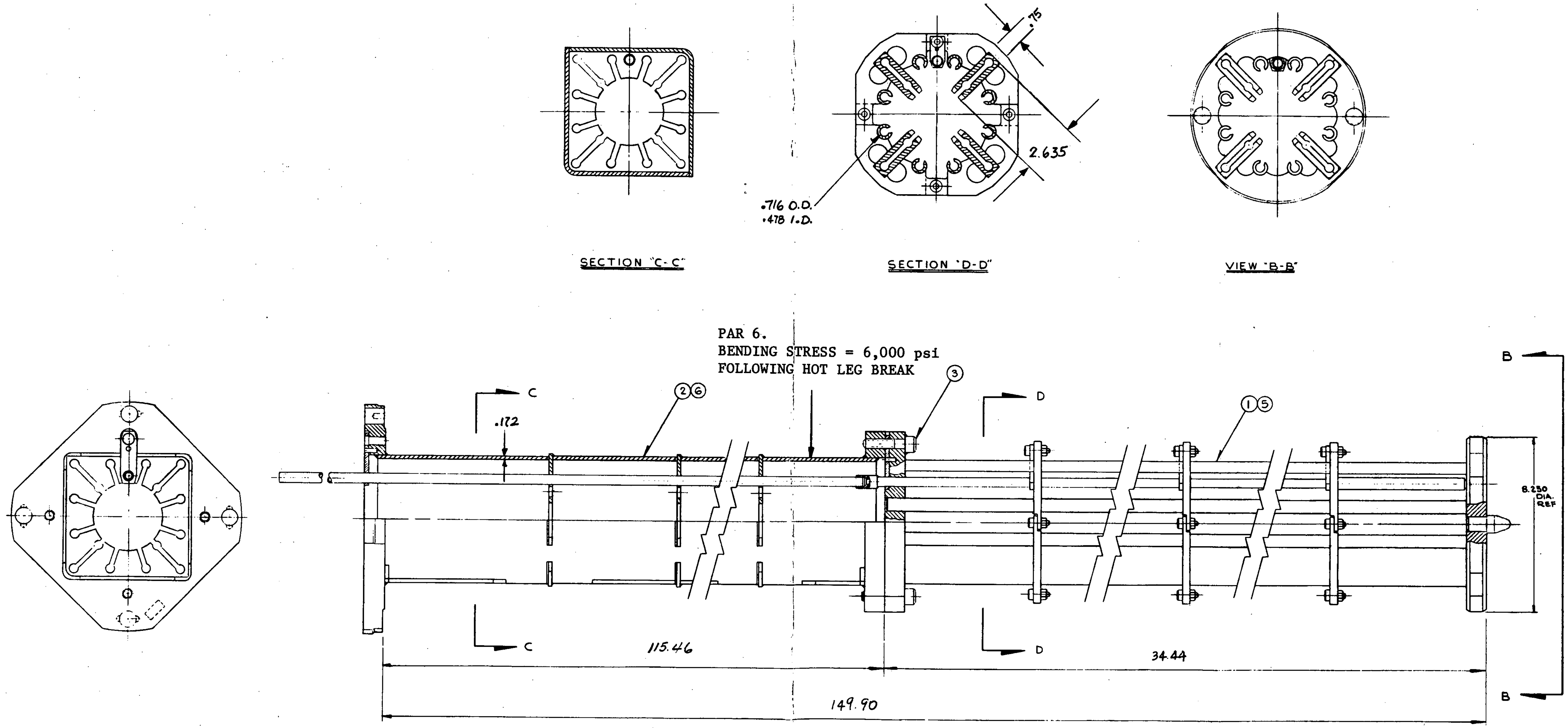


DETAILS OF THERMAL SHIELD TO BARREL CONNECTION
FIG. 2-5



CORE BARREL, FORMER AND BAFFLE
FIG. 2-6

BILL OF MATERIAL						NO. REQ.			
NOTE	ITEM	TITLE	DRAWING & GR. OR IT	MATERIAL SPECIFICATION	EQUIVALENT SPECIFICATION	GR. 1	GR. 2	GR. 3	
	1	LOWER ASSY.				-	1		
	2	UPPER ASSY				-	1		
	3	.500-13 x 1.75 LG. SST				-			
		SOCKET HD. CAP SCREW				4	4		
	4	NEOLUBE				[illegible]	[illegible]		
	5	LOWER ASSY.				1	-		
	6	UPPER ASSY.				1	-		



ASSEMBLY OF ITS. 1, 2 & 3 MAKE UP GR. 2
ASSEMBLY OF ITS. 3, 5 & 6 MAKE UP GR. 1

E.D. SK. 319263-F

CONTROL CLUSTER GUIDE ASSEMBLY
FIG. 2-7

3. ADEQUACY OF PRIMARY SYSTEM DESIGN AND FABRICATION TECHNIQUES

3.1 INTRODUCTION

In development of our specifications for primary loop equipment, the choice of standards for design and fabrication of the equipment was based upon our study of existing applicable industry standards in use at that time. Prior to 1965, the ASME Pressure Vessel Codes, both Sections I and Section VIII were found to have the closest applicability for primary pressure containing equipment. Although Section 1 was used in the design of some of the early plant equipment, Section VIII was ultimately chosen as the standard to be used in lieu of a nuclear code for this equipment. Many code cases were applied to Section VIII to supplement it for applicability to the unique requirements of nuclear vessels as compared to conventional pressure equipment. Ultimately Section VIII and its applicable nuclear code cases were used as a basis for the new Section III Code for Nuclear Vessels. Our choice of Section VIII was based on the investigations we made into its applicability to our design requirements for nuclear vessels and the excellent safety record which had been established with its use on conventional equipment over the years. We found that an outstanding record of safety and reliability had been established on thousands of operating units designed to this code.

In a similar manner, a decision was made to employ the Power Piping Code - ASA B31.1 for all primary system piping. In our study of both codes Section VIII and ASA B 31.1, an investigation was also made to determine the adequacy of the bases on which these codes were established. Through our participation on the Pressure Vessel Research Committee which acts as the research and development organization for these code committees, we followed closely all of the developments which became the basis for these codes over the years. Through our participation in the PVRC and on the various code committees we stay completely abreast of all code changes and improvements and contribute directly to keeping the codes abreast of the state of the act.

3.2 DESIGN

Considerations given to the design of the primary loop components under the rules of the ASME Nuclear Vessel Code Section III Class A and the USA S.I. B31.1 Power Piping Code were previously discussed in the First Supplement, Question 3, page 2, and the Sixth Supplement page 4-1. It is the intent of this section to specifically point out the areas in which Westinghouse primary loop design considerations and equipment specifications are over and above the applicable code requirements.

Design requirements above code requirements are listed below:

1) Independent Reactor Vessel Stress Analysis
   It is Westinghouse practice to independently review the reactor vessel stress analysis. This is not a code requirement.

2) Primary Component Stress Analysis
   The reactor vessel, steam generators, pressurizer and primary coolant pumps receive a complete stress analysis including analysis for cyclic pressure and temperature operation. The ASME Nuclear Vessel Code Section III Class A rules, to which the reactor vessel, steam generators and pressurizer are designed, are generally exempt from cyclic analysis by code paragraph N415.1. Hence, the Westinghouse requirement for cyclic analysis is above the Section III requirement for Class A vessels.

3) Primary Coolant Pump Casing
   The primary coolant pump casing does not come under the rules of the ASME Code. However, Westinghouse has specified that the pump casing be designed to Section III Class A requirements of the ASME Nuclear Vessel Code. In addition the cyclic analysis which is above the Class A requirements is performed on the pump casing.

4) Steam Generator Secondary Side
   The secondary side of the steam generator by definition is classified as a Section III Class C vessel. However, Westinghouse designs the secondary side of the steam generator to the more stringent requirements of Section III Class A vessels.

There are additional areas of conservatism in the design of primary loop components that may or may not be considered to transcend code requirements. These areas are discussed below.

1) Primary System Relief Valves

The primary system will be equippped with three safety valves and two power operated relief valves. The combined capacity of the three safety valves is sufficient to prevent the primary system pressure from exceeding the 2500 psia design pressure by more than 10% during any pressure surge. Hence, the three safety valves alone meet the requirements of paragraph N-910.3 of Section III.

The power operated relief valves are provided to limit the operating frequency of the code safety valves. These valves operate automatically and they have the capacity to relieve the surge rate accompanying an over-power transient. Under paragraph N-911 of Section III credit cannot be taken for these power actuated relief valves. However, their presence in the design is consistent with the conservative Westinghouse primary loop design basis.

2) Design Pressure

The primary loop is designed for a pressure of 2500 psia, and is operated at a pressure of 2235 psig. This is consistent with the code recommendation set forth in paragraph N-441 of Section III that the specified design pressure should provide a suitable margin above the pressure at which the vessel will be normally operated to allow for probable pressure surges. The maximum pressure surge would result from a complete loss of load without an immediate reactor trip. However, the redundancy in the instrumentation and control (I&C) circuitry provided insures that a complete loss of load without an immediate reactor trip, in which case the pressurizer safety valves are not required to operate. Because of the stringent I&C requirements set forth in the Westinghouse design basis, it is felt that the margin provided to accommodate pressure surges is conservative.

3.2 FABRICATION

On page 4-4 of the Sixth Supplement to the PSAR, it was shown that the fabrication of primary loop components is in strict accordance with the applicable codes. However, there are areas where Westinghouse Equipment specifications for primary loop components go beyond the applicable codes.

Fabrication requirements that are above code requirements are listed below:

1) Welding Pre-Heat Requirements
   On the reactor vessel, pressurizer, and steam generators the non-mandatory pre-heat requirements for P1 and P3 material is performed on all weldments.

2) Reactor Vessel Out-of-Roundness Requirements
   In order to insure uniform coolant flow, the Westinghouse out-of-roundness requirements on the cylindrical region in the area of the thermal shield is above code. Section III Class A out of roundness requirements are stated in paragraph N-516. This paragraph states that the difference in inches between the maximum and minimum inside diameters at any cross section shall not exceed the smaller of (D+50)/200 and D/100, where D is the nominal inside diameter in inches at the cross section under consideration. For the Indian Point nominal diameter of 173 inches, the code out-of-roundness requirement is 1.11 inches. Westinghouse requires 0.5% of diameter or 0.865 inches on the cylindrical section of the vessel in the region of the thermal shield.

4. REACTOR SYSTEM INSPECTION

Reactor system inspection includes three important areas, i.e., non-destructive inspection of material and components in the shop prior to shipment, visual inspection of the reactor loops and reactor loop components during the plant's operating life, and provisions in the reactor loop and the reactor loop component design to allow for application of non-destructive test techniques that are presently under development.

Section 4 of the Sixth Supplement to the PSAR summarized the quality assurance program for all reactor system components including the overall surveillance of the program by Consolidated Edison through the services of an independent testing organization. Figure 4-1 of this supplement illustrates graphically this same program and thus shows all of the non-destructive tests and inspections, which are required by Westinghouse specifications on reactor system components and materials for each component. All tests and inspections required by the applicable codes are included. Circled are those tests or inspections that are totally incremental or contain incremental requirements as compared to code requirements.

The detailed explanation of component by component distinctions between special Westinghouse and code requirements is given in Table 4-1 of this supplement.

Westinghouse requires as part of its reactor vessel specification that certain special tests which are not specified by the applicable codes be performed. These tests are listed below:

1) Ultrasonic Testing
   Westinghouse requires that a 100% volumetric ultrasonic test of reactor vessel plate for both shear wave and longitudinal wave be performed. Section III Class A vessel plates are required by code to receive a longitudinal wave ultrasonic test on a 9 in. x 9 in. grid.

2) Measurement of NDTT

Westinghouse requires that the unirradiated or initial NDTT of pressure vessel base plate material be measured by two methods. These methods are the drop weight test per ASTM E208 and the Charpy V-notch impact test (Type A) per ASTM E23. The NDTT is defined in ASTM E208 as "the temperature at which a specimen is broken in a series of tests in which duplicate no-break performance occurs at 10°F higher temperature." Using the Charpy V-notch test, the NDTT is defined as the temperature at which the energy required to break the specimen is a certain "fixed" value. For SA 302B steel the ASME III Table N-332 specifies an energy value of 30 ft-lb. This value is based on a correlation with the drop weight test and will be referred to as the "30 ft-lb-fix." A curve of the temperature versus energy absorbed in breaking the specimen is plotted. To obtain this curve, 15 tests are performed which include three tests at five different temperatures. The intersection of the energy versus temperature curve with the 30 ft-lb ordinate is designated as the NDTT.

3) Radiation Surveillance Program

In attachment 5 to Question 3 of the First Supplement to the PSAR the radiation surveillance program to be incorporated in the operating reactor is described. Samples of material from pressure containing parts are to be placed in irradiation specimen tubes located in the reactor as indicated in Figure 4-2 of the PSAR. Specimens removed from the irradiation tube will be subjected to Charpy tests. Full Charpy curves will be plotted and used to determine the shift in NDTT. Since the specimens, because of their location, will receive a higher irradiation than the reactor vessel, the NDTT determined will be representative of the reactor at a later time in life. This program is not required by the code.

During the design phase of the reactor coolant system careful consideration had been given to allow access for both visual and non-destructive inspection of primary loop components. The following components and areas are available for visual and/or non-destructive inspection:

1) Reactor Vessel - the entire inside surface
2) Reactor Vessel Nozzles - the entire inside surface.
3) Closure Head - the entire inside and outside surface.
4) All reactor vessel studs, nuts and washers
5) Field welds between the reactor vessel and the main coolant piping.
6) Reactor internals
7) Reactor vessel flange seal surface
8) Closure head flange seal surface
9) Fuel assemblies
10) Rod cluster control assemblies
11) Control rod drive shafts
12) Control rod drive mechanism assemblies
13) Reactor coolant pipe external surfaces except for the five foot penetration of the primary shield.
14) Steam generator - the external surface; the internal surfaces of the steam drum, and the channel head internal surfaces
15) Pressurizer - the internal and external surfaces
16) Reactor coolant pump - the external surfaces, motor and the impeller.

The design considerations that have been incorporated into the reactor loop design to permit the above inspections are as follows:

1) All reactor internals are completely removable. The tools and storage space required to permit inspection of the internals will be provided.

2) The closure head will be stored dry on the reactor operating deck during refueling.

3) All reactor vessel studs, nuts and washers are removed to dry storage during refueling.

4) Removable plugs are provided in the primary shield just above the coolant nozzles to provide access to inspect the welds to the reactor coolant pipe. The insulation covering the nozzle welds is readily removable.

5) Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

6) A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

7) The storage stands provide for storage of the internals allow for inspection access to both the inside and outside of the structures.

8) The station provided for transfer of control rod clusters from one fuel assembly to another is specially designed to allow inspection of both fuel assemblies and control rod clusters.

9) The control rod mechanism pressure housing is specially designed to allow removal of the mechanism assembly from above during refueling.

10) Manways are provided in the steam generator steam drum and channel head to allow access for internal inspection.

11) A manway is provided in the pressurizer top head to allow access for internal inspection.

12) All insulation on reactor system components and piping is removable. The piping insulation is removed by undoing the insulation fasteners.

The use of well known non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all reactor loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on the reactor vessel, several features have been incorporated into the design and manufacturing procedures in preparations for non-destructive test techniques which may be employed in the future. These are:

1) Shop ultrasonic examinations are performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate clad bond to allow later ultrasonic testing of the base matal. Size of clad bonding defect allowed is 3/4 inch.

2) The reactor vessel shell in the core area is made with a clean, uncluttered cylindrical surface to permit future positioning of test equipment without obstruction.

3) Various tests are currently underway to determine the effect of clad surface finish on ultrasonic inspectability of vessel material.

The plans for in-service inspection of some Indian Point #2 plant components was previously described in the Second Supplement to the Preliminary Safety Report in Question 1. Table 4-2 summarizes these plans. Plans for inspection of the other reactor loop components is currently in preparation for submittal in the Final Safety Report.

TABLE 4-1
Reactor Coolant System Quality
Assurance Data
Special W Requirements

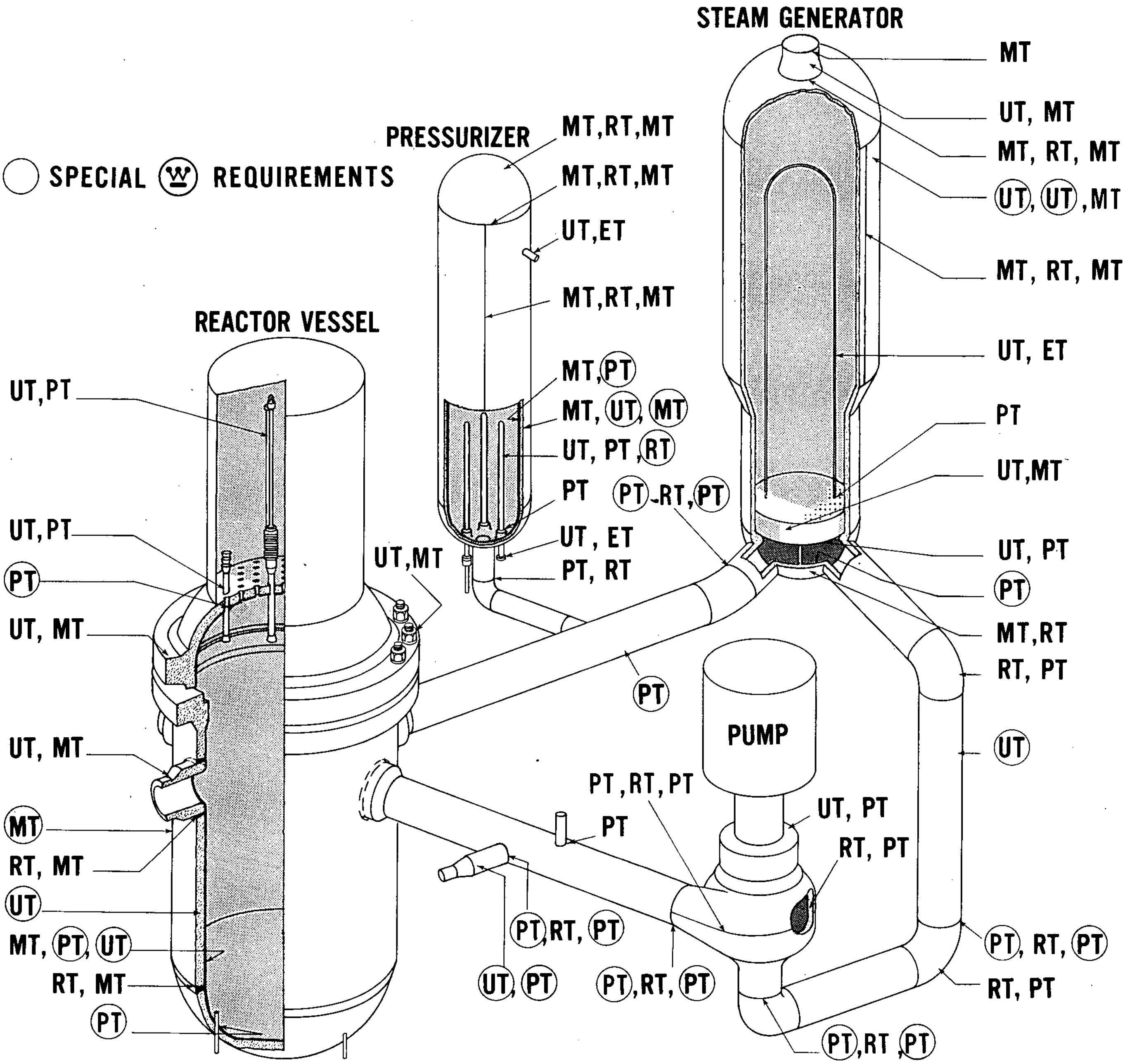
Reactor Vessel	Special W Requirement	Code Requirement
Ultrasonic Testing (UT)	1. 100% volumetric longitudinal and shearwave UT of plate material.	1. Longitudinal UT on a 9"x9" grid pattern
	2. UT of clad bond to a 3/4 in. dia unbonded area repair standard	2. None
Dye Penetrant Testing (PT)	1. PT test every 1/2 in. of weld layers for partial penetration welds to Control Rod Drive Mechanism Head Adapters and Instrument Tube Connections.	PT test of finished weld.
	2. PT test finished surface of cladding.	
Magnetic Particle Test (MT)	1. MT test all outside surfaces of the vessel and head after final stress relief.	1. MT test of weld surfaces
Steam Generator		
Ultrasonic Test (UT)	1. 100% volumetric longitudinal and shear wave UT of plate material	1. Longitudinal UT on a 9"x9" grid pattern
	2. Ultrasonic volumetric longitudinal UT of formed plate after quench and temper	2. None

TABLE 4-1 (Continued)

Dye Penetrant Testing (PT)	1. DP test finished surface of channel head cladding.	1. None
Pressurizer		
Ultrasonic Testing (UT)	1. 100% volumetric longitudinal and shear wave UT of plate material	1. Longitudinal UT on a 9"x9" grid pattern
Dye Penetrant Testing (PT)	1. PT test on finished surface of cladding	1. None
Magnetic Particle Testing (MT)	1. MT test all outside surface of the vessel and head after final stress relief.	1. MT test of' weld surfaces
Radiography (RT)	1. Radiograph heaters to check heater wire positioning.	1. None
Main Coolant Pipe		
Ultrasonic Testing (UT)	1. 100% volumetric longitudinal UT testing of all forged pipe and fittings.	1. None
Dye Penetrant Testing (PT)	1. PT testing of all forged pipe and fittings surfaces.	1. None
	2. Testing of all circumferential pipe weld preparations	2. None circumferential
	3. PT testing of all finished pipe welds	3. None

TABLE 4-2
Consolidated Edison Indian Point #2
In-Service Inspection Plans

	Inspection Area	Method of Inspection	Frequency of Inspection
1.	Reactor Vessel Head	Direct Visual and limited dye penetrant	Each refueling
2.	Reactor Vessel Studs	Magnetic Particle and Ultrasonic	Each refueling (all studs)
3.	Reactor Vessel Cladding between Closure Flange and Reactor Coolant Outlet Nozzles	Remote Visual	The 2nd refueling and every 4th refueling thereafter
4.	Reactor Vessel Head Adaptors	Direct Visual Dye Penetrant	The 2nd refueling and every 4th refueling thereafter
5.	Bottom Instrument Tubes	Direct Visual Dye Penetrant	The 2nd refueling and every 4th refueling thereafter
6.	Nozzle Safe Ends	Direct Visual Dye Penetrant	The 2nd refueling and every 4th refueling thereafter



REACTOR COOLANT SYSTEM QUALITY ASSURANCE DATA
FIG. 4-1

5 REACTOR COOLANT SYSTEM LEAK DETECTION

General

Two radiation sensitive instruments provide the capability for detection of leakage from the Reactor Coolant System. The first, the containment air particulate monitor, is quite sensitive to low leak rates and can be used to alarm the presence of new leaks, if desired. The second is the containment radio gas monitor which is much less sensitive but can be used as a backup to the first.

A leakage detection system is included which determines leakage losses from all water and steam systems within the containment including that from the Reactor Coolant System. This system collects and measures moisture condensed from the containment atmosphere by the cooling coils. It relies on the principle that all leakages up to sizes permissible with continued plant operation will be evaporated into the containment atmosphere. This system provides a dependable and accurate means of measuring integrated total leakage including leaks of the cooling coils which are part of the containment boundary.

A third instrument used in leak detection is the humidity detector. This provides a backup means of measuring overall leakage from all water and steam systems within the containment but furnishes a less sensitive measure. Both the condensate collection and the humidity monitoring methods provide backup to the radiation monitoring methods.

Engineering work by Westinghouse to establish recommended routines for leak detection is in preliminary stages, therefore, more details of the techniques and routines will be presented in the final safety report.

Leakage Prevention

Primary system commponents are manufactured to exacting specifications that exceed normal code requirements as outlined in Section 3 of this Supplement. Because of this, the welded primary system construction, and the extensive non-destructive testing programs outlined in Section 4 of this Supplement, leakage through metal surfaces or welded joints is considered very unlikely.

Valve stems are examples of possible leakage sources. The many small instrument valves, which are located close to the reactor coolant loop pipe, are manually operated and of the diaphragm type. These valves have no stem leakage and experience has proven them highly reliable against failures permitting escape of reactor coolant. The other valves are as described on Page 4-11 of the PSAR. The large valves are equipped with backseats and a leakoff, located between the two sets of stem packing, which connects to the Waste Disposal System. Small valves, except the instrument and control valves, have backseats and stem packing which is at least 1.5 times the stem diameter. Control valves have a leakoff located between two sets of packing.

The seals for the reactor coolant pumps permit some leakage from the Reactor Coolant System. These seals have been developed, refined, and are now in use after the extensive development and testing program by Westinghouse. The PSAR gives details on page 4-7.

Where it is important to permit expeditious disassembly of components for internal servicing, gasketed joints are provided. The reactor coolant pumps, the reactor vessel, and the control rod drive mechanisms have joints of this kind. For the pumps, the motor assembly, including the sealing system, and impeller can be removed from the impeller casing by disassembling a bolted and Flexitallic gasketed joint. For the reactor vessel, two hollow, Inconel, self-energizing o-rings are used.

A leakoff equipped with a temperature monitor provides the means to determine leakage. Any leakage is conducted to the reactor coolant drain tank so is excluded from measurements of Reactor Coolant System leakage. One of the joints sealing each mechanism to its adapter uses o-rings so that the joint can be disassembled. All other mechanism joints are sealed by welding.

The steam generator channels, an access opening for the pressurizer, and all flanged joints in general are sealed by Flexitallic gaskets. Provided it is properly installed, this type of flexible stainless steel asbestos-packed gasket has proved to be successful for the pressure and temperature conditions experienced in PWR reactor systems.

Seals for instrumentation include those at joints for tubing for instruments which sense pressure (and flow), those for temperature sensing devices, and those for flux monitor thimbles. Tubing connects from the root valve near the reactor coolant piping to pressure transducers and utilizes swage type fittings. Temperature sensing detectors are installed in a joint arrangement which utilizes o-rings and provides for seal weld backup. Thermocouples for in-core instrumentation are sealed to an adapter by swage type fittings and the adapters, each of which hold a group of thermocouple penetrations, are in turn sealed to the reactor coolant barrier by a Conoseal gasket.

The seats of the spring-loaded safety and the air-operated relief valves that assure protection against overpressure in the reactor coolant system are difficult to seal against leakage. Since these devices relieve into the pressurizer letdown tank which in turn is connected to the reactor coolant drain tank, any leakage is controlled and does not appear in the containment atmosphere.

Although these many sealed joints are potential sources of leakage, care has been taken in each case to select the most appropriate sealing device. Because of the great number of joints and the difficulty of assuring complete freedom from leakage, a small integrated tolerance is considered acceptable.

Locating Leaks

Experience has shown that hydrostatic testing is quite successful in locating most of the leaks in the pressure containing systems. With the addition of other operating conditions on these systems new leaks can appear.

Methods of leak location for these include visual observation for escaping steam or water or for the presence of white stains near the leak. These are caused by boric acid crystals transported outside the Reactor Coolant System in the leaking fluid and then left behind by the evaporation process. Portable sonic detectors sensitive to ultrasonic frequencies provide another means for locating small leaks.

Other, less sensitive means of detecting leakage into the containment are available as backup to the basic leak detection methods.

Containment Air Particulate Monitor

The containment air particulate monitor is the most sensitive instrument of those available for detection of reactor coolant leakage into the containment. This instrument is capable of detecting particulate radioactivity in concentrations as low as 10^{-9} μc/cc of containment air.

The air particulate monitor sensitivity to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal base-line leakage into the containment. The sensitivity is greatest where base-line leakage is low as has been demonstrated by the experience of Indian Point Unit No. 1, Yankee Rowe and Dresden Unit 1. Where containment air particulate activity is below the threshold of detectibility, operation of the monitor with stationary filter paper would increase leak sensitivity to a few cubic centimeters per minute. Assuming a low background of containment air particulate radioactivity, a reactor coolant corrosion product radioactivity (Fe, Mn, Co, Cr) of 0.2 μc/cc (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactive solids into the containment air, the air particulate monitor is capable of detecting leaks as small as 0.013 gal/min (50 cc/minute) within thirty minutes after they occur. If only ten per cent of the particulate activity is actually dispersed in the air, the threshold of detectable leakage is raised to 0.13 gpm (500 cc/minute).

For cases where base-line reactor coolant leakage falls within the detectable limits of the air particulate monitor, the instrument can be adjusted to alarm on leakage increases from two to five times the base-line value.

Condensate Measuring System

The principle that the condensate collected by the cooling coils matches, under equilibrium conditions, the leakage of water and steam from systems within the containment applies because conditions within the containment promote complete evaporation of leaking water from hot systems. The air and internal structure temperatures are normally held at 120°F, the air is dry, the cooling coils provide the only surfaces at or below the dew point temperature and there are no internal bodies of water exposed to the atmosphere during plant operation.

The containment cooling coils are designed to remove the sensible heat generated within the containment. The resulting large coil surface areas means that the exit air from the coils has a dew point temperature which is very nearly equal to the cooling water temperature at the air exit.

Measurement of the condensate drained from the cooling coils is made to determine collection rate and thus leak rate. About one-half hour after the occurrence of a leak, the equilibrium condition is established in which the amount of the leakage change is matched by a change in the cooling coil condensation rate. Measuring the condensate flow is done with sufficient accuracy for the operator to decide if operating limits are being met.

Containment Radio Gas Monitor

The containment radiogas monitor is inherently less sensitive (threshold at 10^{-6} μc/cc) than the containment air particulate monitor, and would function only in the event that sufficient reactor coolant gaseous activity exists from fuel cladding defects. Assuming a reactor coolant gas activity of 0.3 μc/cc, the occurence of a leak of two to four gpm would double the background (predominantly argon-41) in about an hour's time. In these circumstances this instrument would be a useful backup to the air particulate monitor.

Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the containment. This instrumentation has not nearly the sensitivity of the air particulate monitor, but has the advantage of being sensitive to vapor originating from all sources, the reactor coolant, the steam, and the feedwater systems. Plots of containment air dew point variations above a base-line maximum established by the cooling water temperature to the air coolers should be sensitive to incremental leakage equivalent to 2 to 10 gpm.

The sensitivity of this method is a dependent on cooling water temperature. With the least sensitivity, based on peak summer cooling water temperatures, it is estimated that an increase of 2 gpm in leak rate will cause a rise in containment dew point temperature of one degree Fahrenheit.

6. CORE DESIGN

6.1 MODIFICATIONS TO THE CORE DESIGN

Since the submittance of the PSAR a number of modifications have been incorporated in the core design. These changes are a result of refinements in the core physics and core design calculations; the addition of part length absorber rods, and the addition of static burnable poison rods. Changes in the core design parameters resulting from these changes are presently being finalized. The new core parameters will be submitted in the FSAR.

The fuel loading arrangement planned for the initial cycle was shown in Figure 3.2-32 of the PSAR. In order to improve the second cycle power distribution, the checkerboard initial cycie fuel loading arrangement shown in Figure 6-1 will be employed. In this initial cycle, the core is divided into three regions of different enrichments; an outer region containing 64 fuel assemblies, and a central region containing 129 fuel assemblies in a checkerboard array. During subsequent cycles fresh fuel will be added to the core periphery and the inner two regions will be arranged to give an acceptable power distribution.

Detailed physics studies indicate that the core may be subject to axial, diametral and azimuthal Xenon oscillations with the first cycle positive moderator coefficient originally planned. Part length absorber rods will be used in the Indian Point Unit No. 2 core to control axial Xenon oscillations, and fixed burnable poison rods will be used in the first cycle to maintain a negative moderator coefficient at operating temperature. The addition of the fixed burnable poison rods insures that diametral and azimuthal Xenon oscillations will be damped. In succeeding cycles the moderator coefficient is negative and the xenon oscillations are damped without the use of the burnable poison rods.

WCAP-7072, which is attached to this supplement as Appendix A describes the use of part length rods in Westinghouse pressurized water reactors. These rods will be used to dampen axial Xenon oscillations; and to shape the axial flux, thereby maintaining hot channel factors within acceptable limits. Detailed physics and control studies are still in progress. The FSAR will contain a description of the following;

1. Xenon Oscillations
   Additional pertinent results of the physics studies on the characteristics of Xenon Oscillations will be presented.

2. Control Strategy
   The control and safety systems to be employed will be described in detail.

3. Operator Function
   The indication and control elements to be provided to the operator, and the frequency of the control function will be described.

WAPD studies indicate that the part length rods can adequately dampen Xenon oscillations with the control described in Appendix A with minor, if any, modifications.

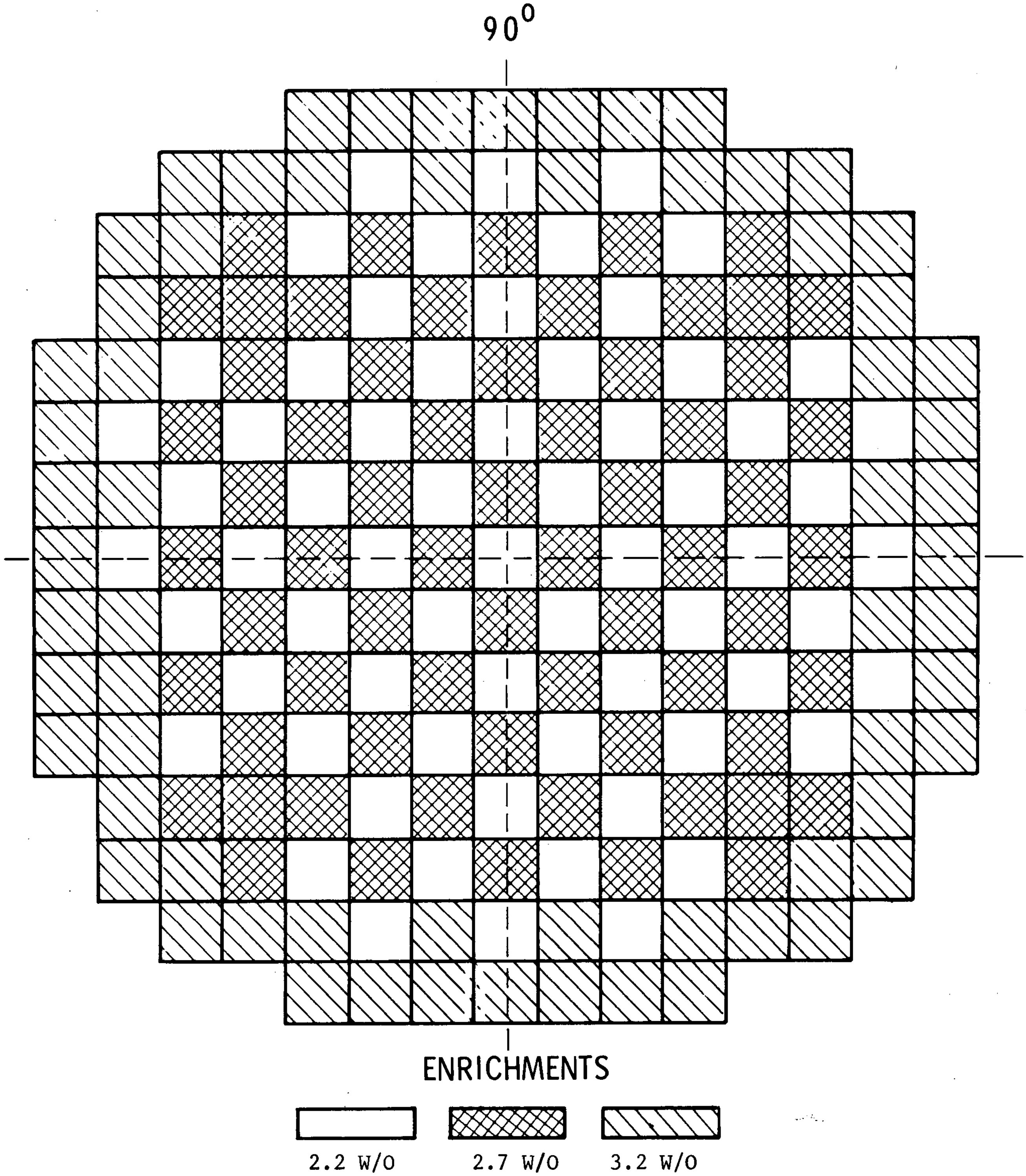
WCAP-7113, which is attached as Appendix B describes the use of static burnable poison rods in Westinghouse pressurized water reactors. The negative temperature coefficient obtained by employing burnable poison rods eliminates the concern over moderator voiding during postulated excursions. The details of the static burnable poison rods final design will be presented in the FSAR.

6.2 REACTIVITY STUDIES

Design studies have been performed to determine the potential for reactivity insertion as a result of fuel rod bowing. As a consequence of the application of burnable poison the reactivity effects from rod bowing are greatest at the end of cycle life. This is the condition under which the analysis was performed. Any displacement of a fuel rod which is merely a realignment of its position relative to its neighbors, resulting neither in a net gain nor a net loss of moderator is found to result in a slight negative reactivity effect.

If the fuel rod bowing results in assembly repositioning, there is a buildup of tolerance between assemblies which can result in a reactivity gain at end of cycle life. This tolerance buildup is limited by the core baffle which is rigidly connected to the core barrel by former plates, and by the fuel assembly grids which maintain fuel rod geometry within an assembly. If the tolerance buildup is assumed to redistribute from a condition of minimum (not uniform tolerance distribution) reactivity to that of maximum reactivity, there will be no greater gain than 1/3%. This is the maximum reactivity insertion due to fuel rod bowing.

Because of the addition of burnable poison rods, the moderator temperature coefficient is no longer positive. Although it is true that regions of the core still retain a positive coefficient when the overall coefficient is zero, the potential reactivity insertion is trivial.



CORE LOADING ARRANGEMENT
FIG. 6-1

7. REACTOR PIT CRUCIBLE

7.1 SUMMARY DESCRIPTION

The reactor pit crucible was proposed in Supplement 5 to serve as back-up to the emergency core cooling system in the event that the core might melt and deposit in the reactor vessel cavity. The reactor pit crucible, Figure 7-1, is a refractory lined vessel with a sloped bottom supported and elevated from the cavity floor by structural members which allow free flow of water beneath the vessel and steam separation by way of the space between the vessel sides and the concrete reactor cavity walls. The crucible is located directly below the reactor vessel and in-core instrumentation guide thimbles, and extends into the access tunnel as shown on Figure 7-2. The capacity of the crucible is sufficient to contain all of the fuel (UO_2), fuel assembly grids (Inconel), fuel assembly end fixtures (stainless steel) and portions of the lower core support structure and reactor vessel bottom head.

The addition of the accumulators to the redundant and independently protected emergency core cooling systems relegates the reactor pit crucible back up to a role which will never be needed, even under the worst accident hypothesis. Core thermal transient studies (Section 1, Supplement Six) analytically demonstrate the capability of the emergency core cooling system, with its increased capacity, in preventing any clad melting for a loss of coolant accident even when considering complete rupture of a reactor coolant pipe.

The engineering approach to design of the reactor pit crucible was based on the following generalized assumptions.

1. The reactor pit is submerged with borated water from the break in the reactor coolant system and introduction of containment spray water from the refueling water storage tank (350,000 gals capacity).

2. Molten agglomerate including UO_2 solidifies on contact with the water cooled refractory.

With solid external boundarys of the agglomerate mass, an insulating barrier is obtained which essentially lowers the interface temperatures in the refractory to plate region.

Thermal analysis presented in Supplement 5 indicated refractories with service temperatures in the range of 3000°F are adequate. Of the many materials available today, silicon carbide brick and a high alumina brick are presently being considered for use in Indian Point No. 2. Chemical and Physical characteristics of the above materials are shown in Table 7-1.

Both materials are available in the same fundamental geometeric shapes and are suitable for attachment to steel back up plates. It is presently planned to cover the refractory with a ten or twelve gauge stainless steel liner to preclude material damage and moisture absorption during plant life time.

TABLE 7-1

CHEMICAL AND PHYSICAL CHARACTERISTICS

OF REFRACTIES

CHEMICAL COMPOSITION	Percent Alumina Brick	Silica Carbide
Al_2O_3	89.0	0.78
SiO_2	9.0	11.75
SiO	-	86.10
Fa_2O_3	0.25	1.05
TiO_2	0.10	-
CaO	0.02	0.21
MgO	0.02	0.10
Na_2O	-	Trace
K_2O	-	None
B_2O_3	-	-
PHYSICAL CHARACTERISTICS		
Bulk Density		
gms/cc	3.01-3.06	2.57-2.65
lbs/ft^3	188-191	161-168
Porosity-Percent	14-16	14-17
Modulus of Rupture-psi	2800-3600	3000-4000
Cold Crushing Strength-psi	14-18000	14-18000
Use Temp.-°F	3100	3100
Coef. of Thermal Expansion (per °F - 212 to 1800°F)	$4.2x10^{-6}$	$2.6x10^{-6}$
Thermal Conductivity ($BTU/hr/ft^2/in/°F$)	$\cong$ 16	105

7.2 DESIGN BASIS

The crucible is designed to contain the residue from core meltdown and vessel melt through, thereby preventing contact of the meltdown residue with the containment.

The load criteria for the design of the refractory lined crucible are based on a core and reactor vessel meltdown residue of 512,000 lbs. determined as follows:

		lbs.
1.	Reactor fuel (UO_2)	220,217
2.	Zr (as ZrO_2) Fuel assembly girds (Inconel) Fuel assembly end fixtures (Stainless steel)	61,100
3.	Lower core support structure (Stainless steel)	166,000
4.	Reactor vessel bottom head (Carbon steel)	64,500
	Total	511,817

The limiting case for crucible structural design is the assumption of meltdown residue collecting in a cone shaped mound with a 1:1 height to diameter ratio and a density of 750 lbs/ft^3 which results in a core volume of 683 ft^3. The weight of the meltdown cone over its area results in a load of approximately 8300 psf on the crucible under the reactor, a value well within the strength of the structure at its use temperature.

The dynamic loading of the crucible structure associated with vessel melt through and Class I seismic criteria, as defined in Supplement Two to the Preliminary Safety Analysis Report, have been applied to the structural design.

The cooling mechanism of the crucible was analyzed in Section 2 of Supplement No. 5 to the Indian Point Unit No. 2 Preliminary Safety Analysis Report. The principle consideration for crucible thermal integrity is that the steel vessel be maintained at temperatures below which its strength satisfies the load requirements previously listed. The major item affecting temperature in the steel vessel is the heat flux which determines surface temperature and temperature rise through the metal. The heat flux is dependent on the "melt" volumetric heat generation rate, and conductivities of the solid and liquid UO_2 phases which with the UO_2 vaporizing temperature and the refractory melting temperature establish the thickness of the solid and liquid UO_2 layers which conduct heat into the refractory.

In Section 7.4, conservative estimates for the above parameters have been used in a steady state thermal analysis which establishes that the mode of heat transfer at the vessel water interface is nucleate boiling and the temperature in the carbon steel crucible vessel would be maintained below 800°F. In Supplement 5 the transient effect of initially molten UO_2 contacting the crucible was determined to result in a monotonic heat up of the system verifying the applicability of the steady state thermal analysis.

7.3 MECHANICAL AND STRUCTURAL DESIGN

7.3.1 CRUCIBLE (See Figures 7-2 and 7-3)

The bottom of the crucible is composed of 12 inch wide flange sections sitting on concrete pads to form a trough with a minimum angle of 20° from the horizontal up to a maximum of 37° at toe of boot. One inch carbon steel plates are welded to the wide flange sections; this plate supports the refractory.

The side walls of the crucible under the incore instrumentation leads consist of 18 inch wide flange columns resting on concrete and bearing against concrete side walls. Where crucible side walls taper down to 8 inch wide flange sections due to clearance required for incore instrumentation, the columns are 1 inch plate box sections. One inch steel plates are welded to these vertical columns. The side walls extend approximately 11 feet (up to El. 29'-0", top of crucible) above the lowest point of the concrete.

The side walls under the reactor, consist of 2 inch thick plate against the concrete, 18 inch wide flange sections welded to the plate and then a 2 inch layer of plate welded to the 18 inch wide flange columns. These walls extend approximately 12 1/2 ft above the lowest point of concrete. The side walls will have 2 inch T-bars welded vertically at 2 ft. center to center spacing to support the 4-1/2 inch refractory.

The bottom plate is designed for approximately 8300 psf under the reactor and based on approximately 685 cubic feet of molten material. A loading of approximately 4300 psf is assumed at the toe of the boot. The temperature of the plate as a result of meltdown is considered in determining allowable plate stresses.

There are four - 8 inch diameter pipes encased in concrete around the reactor section that will carry water from elevation 46' - 0" to the underside of the crucible by gravity flow.

The side walls of the crucible will have a baffle located around the top extending out from concrete wall to direct falling material inward toward the crucible and away from the cavity annulus.

All structural steel sections and plates will be in accordance with ASTM A36 specification.

All welding will be in accordance with the latest edition of American Welding Society D1.0-66.

7.4 THERMAL ANALYSIS

Section 2 of the Fifth Supplement to the PSAR describes the mechanisms of heat transfer which govern the design of the reactor pit crucible. A hypothetical accident is postulated in which total failure of core cooling systems results in melting and migration of the core material, lower core support and reactor vessel bottom head, to the crucible. In this configuration, the core mass is surrounded by water resulting from reactor coolant loss and containment spray. Self heating of the core mass is only partially dissipated by conduction through the solidified outer crust. The remainder is transferred by expulsion of UO_2 vapor and liquid through breaks in the crust created by UO_2 vapor pressure. Quenching of these eruptions by the surrounding water causes steaming in the reactor cavity which is relieved through the access openings to the reactor loop compartment where it is ultimately condensed by containment spray or the air recirculation coolers.

It is the specific purpose of the crucible in this hypothetical situation to maintain insulation of the foundation mat and liner from the core mass. A water space is provided surrounding the crucible shell for the purpose of cooling the steel plate and refractory insulation of the crucible. Boiling at the plate surface maintains a temperature gradient through the crucible structure which is definable in terms of the volume heat source, melting points and thermal conductivities of the materials. A satisfactory evaluation is obtained when the average temperature of the steel plate is within the range where adequate structural strength is assured and the heat flux at the water/steel interface is within the nucleate boiling range.

Heat Transfer at the Crucible Wall

The transient and steady-state heat transfer models with which the temperature distribution in the crucible wall was determined are described in Supplement Five, pages 2-4 to 2-10. Further consideration of the problem during subsequent engineering studies has confirmed the applicability of these models, subject to confirmation of the thermal properties of the materials used. In the present supplement, the results of a recalculation of temperatures with parameters reflecting design are presented.

The following refinements of input parameters are here incorporated:

1. Internal heat generation rate (q''') of the UO_2 has been selected as 0.4×10^6 Btu/ft^3 hr based on the residual heat rate (1.9% of the full power value) at an elapsed time of 2000 seconds after shut down. Noble gases, halogens, and 5% of the cesium fission products are assumed stripped from the fuel. The remaining decay heat sources are assumed to be present in concentrations representative of the leading 10% of the core, having an average power density which is 1.6 times the core average.

2. A value of 5.2 Btu/ft-hr-F° was assumed for the effective conductivity of the molten UO_2 below the vaporization temperature. This was the "worst case" assumption used in the Fifth Supplement analysis.

3. Solid UO_2 conductivity was taken as 2.5 Btu/ft-hr-F°, again the worst case value, to maximize the heat flux to the crucible wall.

4. A maximum UO_2 liquid temperature of 8000°F was assumed, corresponding to the vaporization temperature of UO_2 at 4000 psia.

5. Refractory conductivity values of 1.33 and 8.75 Btu/ft-hr-F° were assumed, representing alternate material selections of Cerox 700 (alumina) and Kellog 3 AD (silicon carbide), respectively. In both cases the thermal analysis was based on the assumption that above 3000°F the refractory was removed from the slab, reducing the thickness of the refractory layer until only material below 3000°F remained. In this manner a conservative allowance was made for possible eutectic formation which might reduce the protection of the steel afforded by the firebrick.

The maximum heat flux to water estimated in these calculation is 1.28×10^5 Btu/ft^2 hr. No higher value can be expected, because:

a) Higher conductivity of refractory is not obtainable with the materials under consideration.

b) Maximum values of UO_2 solid and liquid conductivities are assumed.

c) **Maximum** value of UO_2 liquid temperature is assumed.

d) Lower values of the parameters listed in (a-c) give rise to less UO_2 contained in the conducting layer, hence a lower heat flux.

The maximum heat flux of 1.28×10^5 Btu/hr ft^2 is within the range where pool boiling heat transfer will maintain the water-side wall surface temperature below 330 °F with a conservative heat transfer coefficient of 4000 Btu/ft^2-hr-°F. DNB heat flux at this condition would be expected to be approximately 200,000 Btu/ft^2 hr. The calculation has thus established a stable heat transfer mode capable of dissipating the largest heat flux which can be delivered to the crucible/fuel interface. Table 7-2 summarizes the region thicknesses and interface temperatures for both the insulating materials under consideration.

At a heat flux of 1.28×10^5 Btu/hr ft^2, the calculated steam separation rate at the periphery of the crucible is a maximum of 0.122 lb/sec. per foot of perimeter. A minimum of 18-in radial clearance is provided at all points, sufficient to pass this flow of steam with a superficial velocity of about 1 ft/sec, hence there would be no tendency to vapor-bind the region under the crucible.

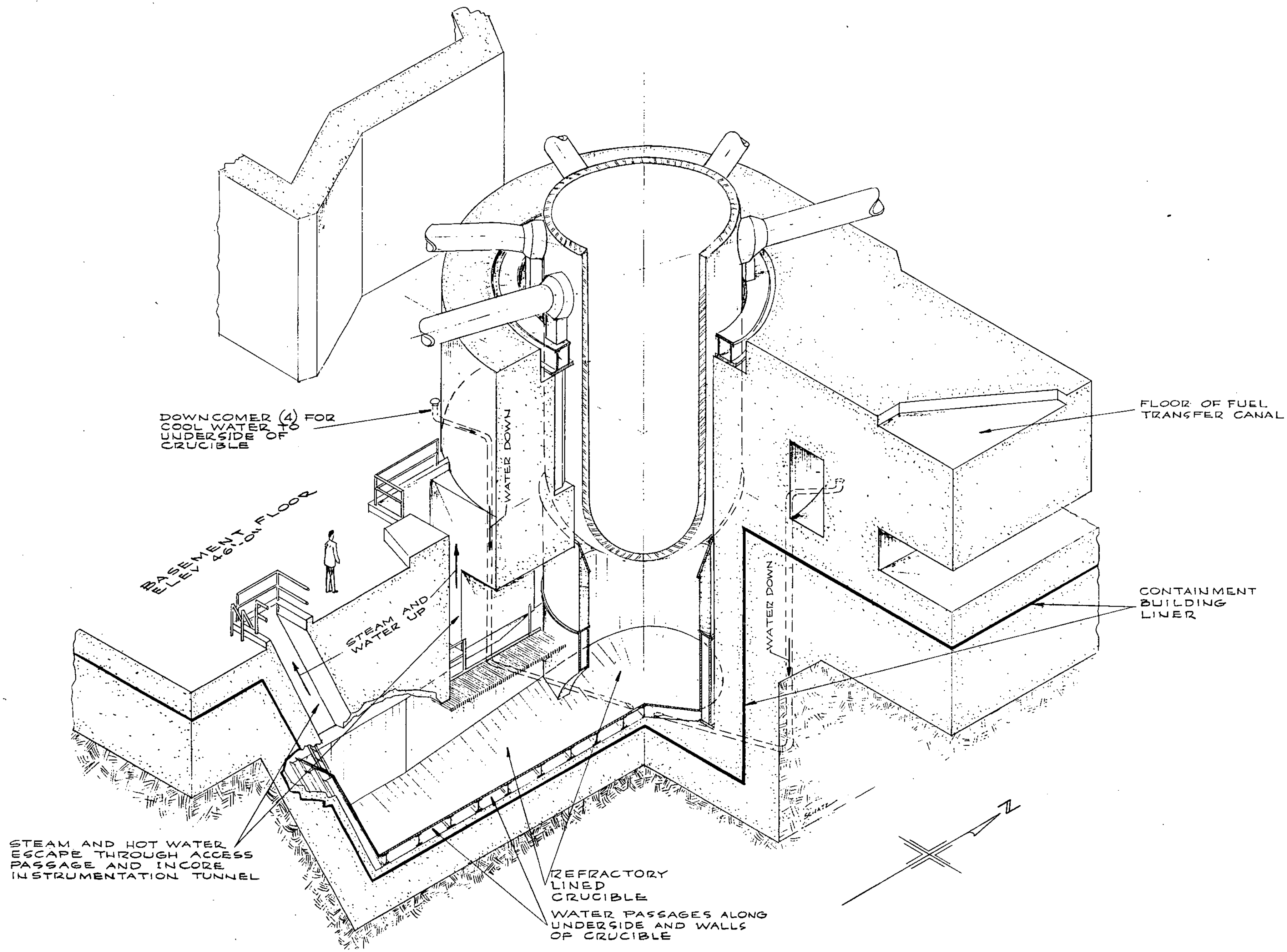
The total rate of steam formation from the entire core mass, if it were contained in the crucible cavity and assuming saturated water were returned from the containment floor to the cavity, would be about 65 lb/sec.

The available area to pass this steam flow is approximately 30 sq. ft., comprising the openings for personnel access and in-core detector conduit into the reactor cavity. The superficial velocity of steam rising through these openings is about 32 ft/sec. The calculated pressure drop for this steam flow is less than one inch of water, hence the back pressure created is of no consequence in limiting the natural convection flow of water through the downcomers in the reactor cavity.

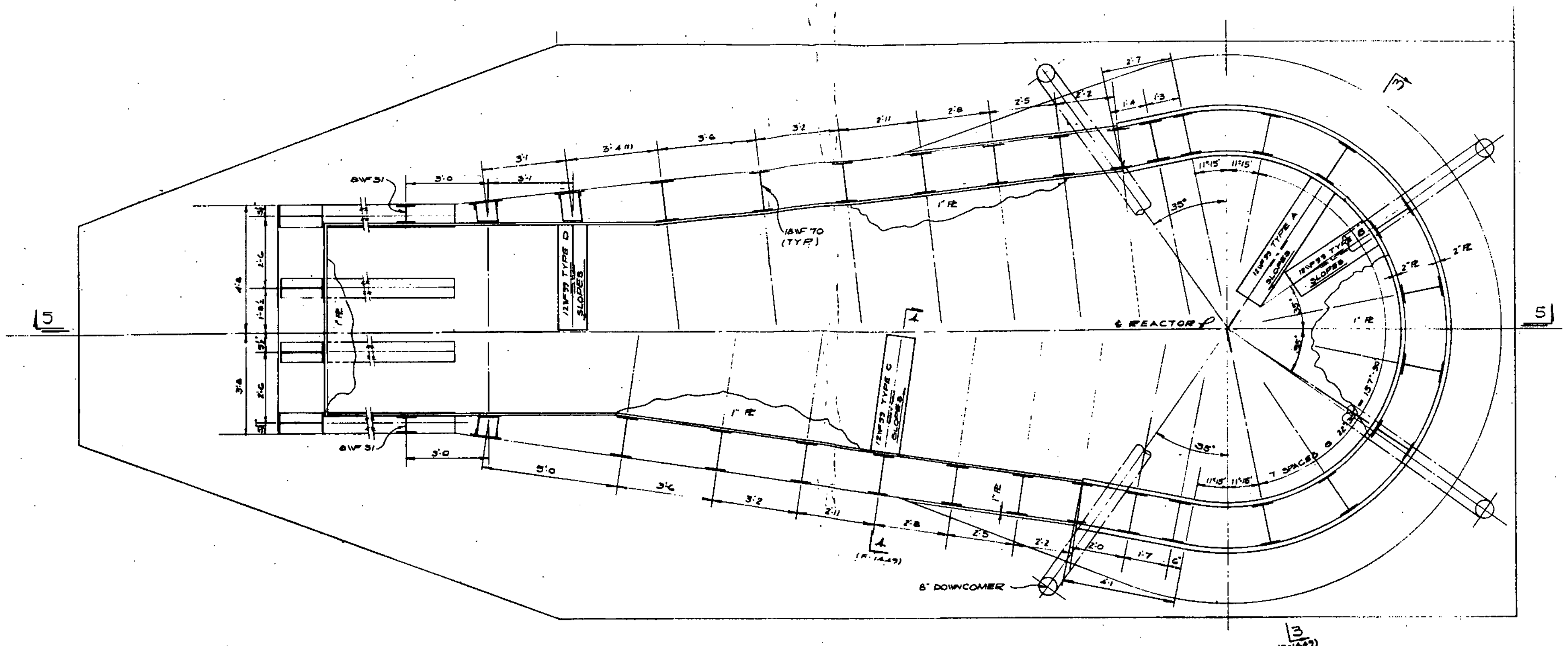
TABLE 7-2

SUMMARY OF THERMAL ANALYSIS

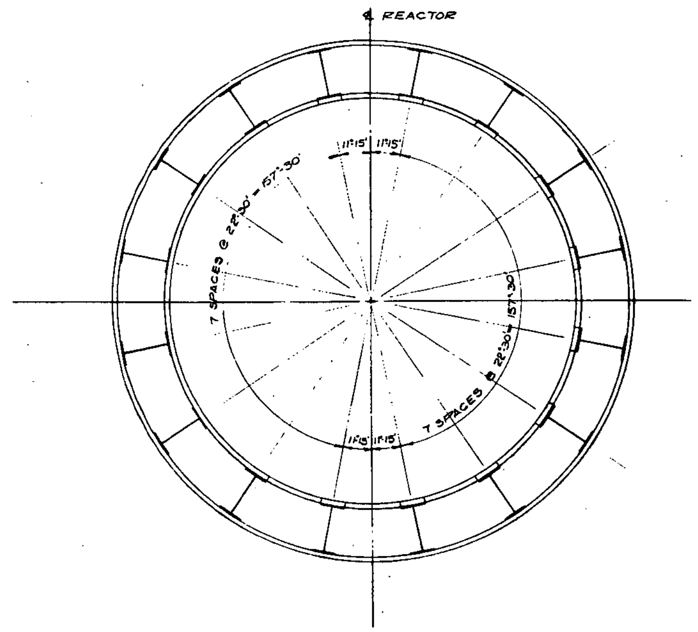
Volumetric Heat Generation Rate (Btu/ft^3-hr)		400,000
UO_2 Vapor Temperature (°F)		8000
Liquid UO_2 Conductivity (Btu/ft-hr-°F)		5.2
Liquid UO_2 Thickness below Vapor Temperature (in.)		3.4
Solid UO_2 Conductivity (Btu/hr-ft-°F)		2.5
Solid UO_2 Thickness (in.)		0.5
Heat Flux Through Crucible (Btu/ft^2-hr-°F)		128,000
Refractory Material	Cerox-700	Kellog 3 AD
Refractory Conductivity (Btu/hr-ft-°F)	1.3	8.8
Refractory Thickness Below 3000°F (In.)	1.88	0.29
Maximum Steel Plate Temperature (°F)	800	800



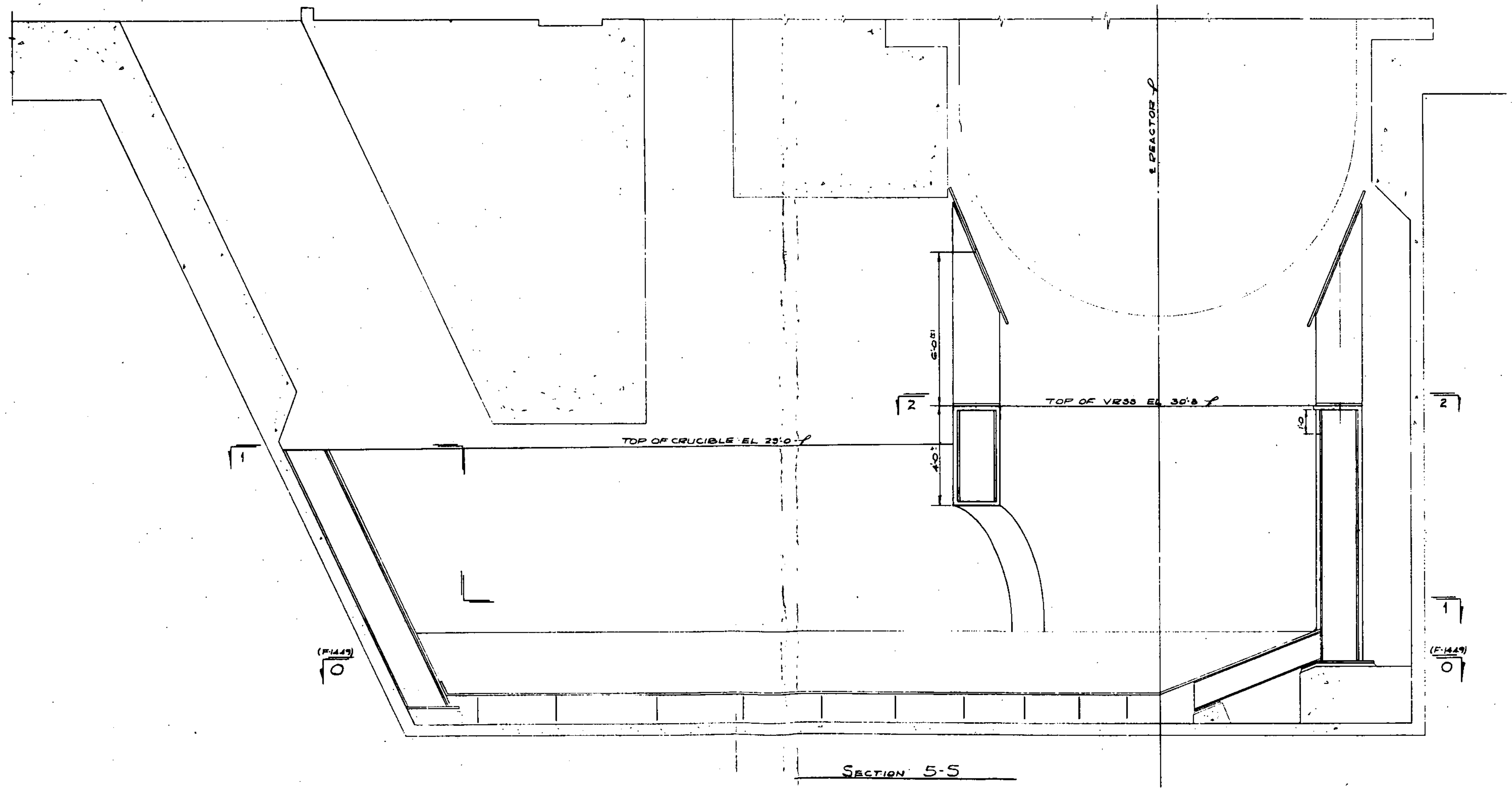
REACTOR PIT CRUCIBLE
FIG. 7-1



SECTION 1-1
SCALE 1/4" = 1'-0"

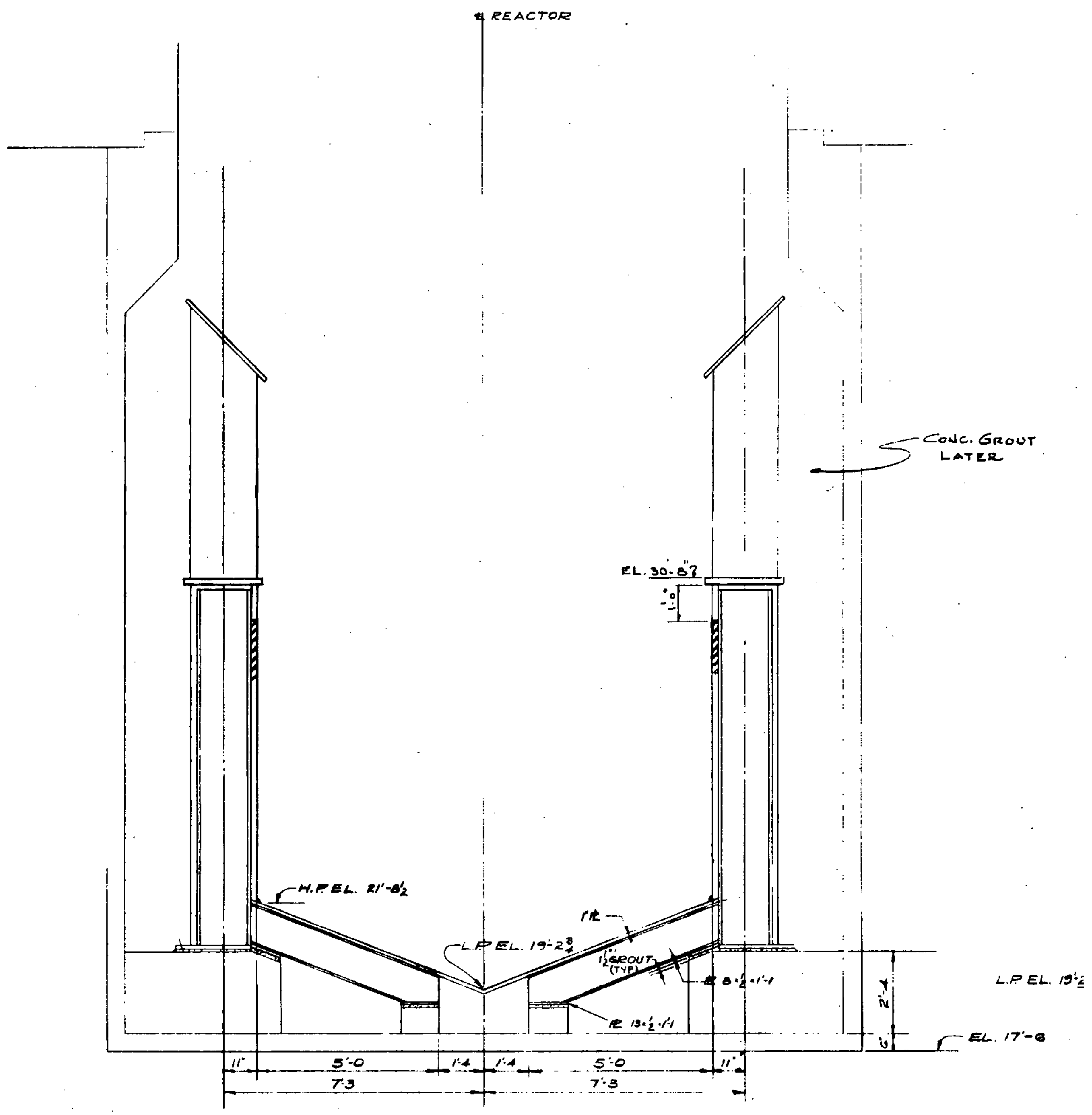


SECTION 2-2



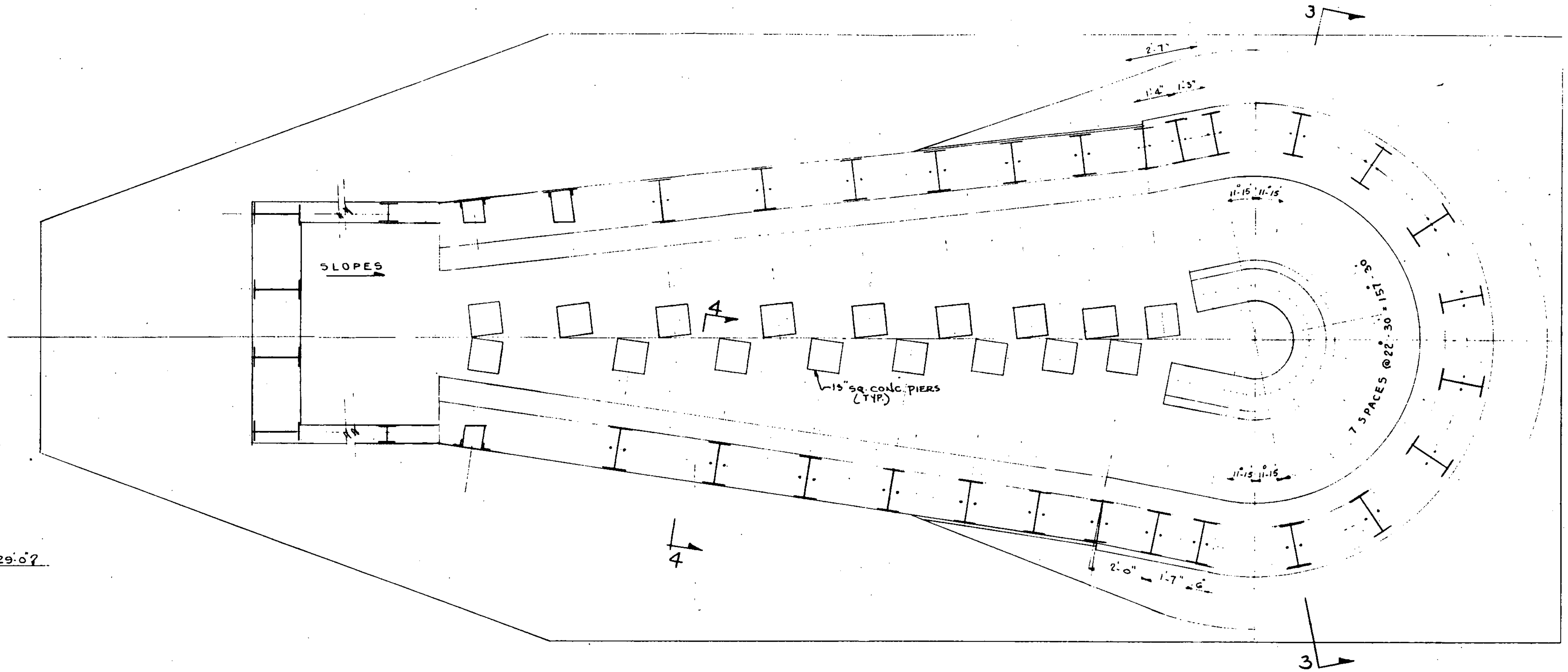
SECTION 5-5

REACTOR PIT CRUCIBLE STEEL DETAILS
FIG.

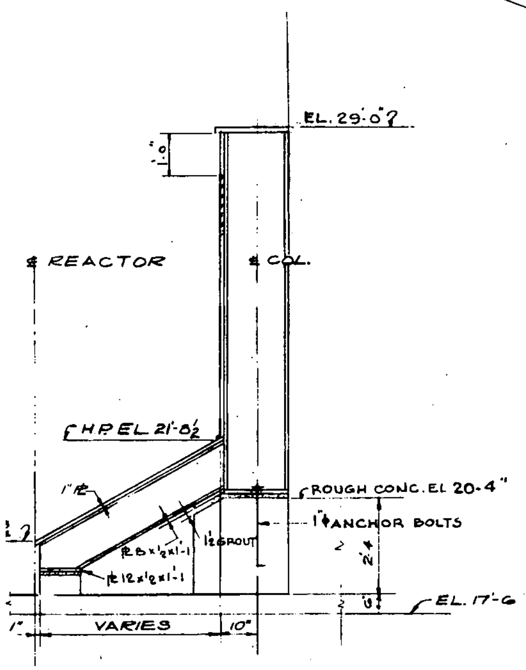


℄ REACTOR
CONC. GROUT LATER
EL. 30'-8⅞
1'-0
H.P. EL. 21'-8½
L.P. EL. 19'-2¾
PL
1½" GROUT (TYP)
PL 8·½·1'-1
PL 13·½·1'-1
L.P. EL. 19'-2
2'-4
6"
EL. 17'-6
11"
5'-0
1'-4
1'-4
5'-0
11"
7'-3
7'-3

SECTION 3-3



PLAN O-O (F-1447)



SECTION 4-4

REACTOR PIT CRUCIBLE SECTIONS AND CONCRETE OUTLINE
FIG. 7-3

APPENDIX A

WCAP-7072

USE OF PART LENGTH ABSORBER RODS
IN WESTINGHOUSE PRESSURIZED WATER REACTORS

June 1967

WCAP 7072

USE OF PART LENGTH ABSORBER RODS IN WESTINGHOUSE PRESSURIZED WATER REACTORS

June 1967

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TABLE OF CONTENTS

1 INTRODUCTION AND SUMMARY

1.1 GENERAL

Part length control rods will be provided in the reactor in addition to the normal control rod system. The function of these rods, which are similar to standard control rods but have control material in only the bottom three feet, is to shape the axial power distribution and to control axial xenon oscillations. Eight of these part length rods will be used. Their tentative locations in the core are shown in Figure 3.1-2. These rods can be located in any control rod location and studies to select their optimum location are in progress.

Calculations discussed in Section 4.2 show that the part length rods are effective in controlling axial xenon oscillations. In addition, they are beneficial in flattening the axial power distribution. The hot channel factors for this reactor are $F_{\Delta h}$ = 1.70 ($F_{\Delta h}$ nuclear = 1.58) and F_q = 2.82. The physical design limits on power capability are linear fuel rod power density (kw/ft) and the DNB (departure from nucleate boiling) ratio. The DNB ratio is calculated with the W-3 DNB correlation and will be equal to or greater than 1.30.

The hot channel factors F_Z and $F_{\Delta h}$ do not represent physical design limits but must be considered in their relationship to F_q and DNBR limits. As will be discussed in detail in Section 4.3, a reduction in F_Z by the action of the part length rods will allow an increase in $F_{\Delta h}$. In the present design most of the design conservation is in F_Z although calculated values of $F_{\Delta h}$ are well below design values.

The part length rods will be controlled manually by the operator from the control console using information displayed on the console. The primary information will be the comparison between the readings of the upper and lower sections of the long ion chambers external to the core. The part length rods will be moved as a bank in a manner to make the upper and lower ion chamber readings approach a prescribed relationship.

Motion of the part length rods is necessary only at several hour intervals when the plant is operating at steady state. When major changes in load are made some administrative control will be required to maintain the part length rods within a prescribed allowable region of travel.

The adequacy of the instrumentation system to provide information for the control of the part length rods is discussed in Section 6. It is concluded that design limits on F_q and the DNB ratios are never exceeded even though the part length rods are not located properly so long as the following conditions are met.

1. The part length rods are not moved into the upper part of the core above an administratively imposed upper travel limit. This upper limit is a function of the control bank location.

2. The part length rods are moved periodically (every 3-5 hours) to damp out xenon oscillations. A time period of the order of 24 hours without control would be required for axial oscillations to become serious.

Since an inadvertent trip of the part length rods could result in the addition of 0.2% reactivity, provisions will be made to prevent them from tripping.

The power supply for control of the part length absorber control rods will be provided from multiple and independent sources such that failure or loss of any one power source will not result in a tripping of all part length rods. These sources of power supply will also be independent of the power source for the standard rod control clusters.

Trip of one or more part length rods is considered an accident condition and is discussed in Section 6.

Other reactivity accidents (credible and hypothetical) involving undesired part length rod motion will be evaluated during detail design and will meet criteria of no departure from nucleate boiling (DNB) or no vessel overpressurization as outlined in Section 7.

2 RESEARCH AND DEVELOPMENT

No research and development programs are required concerning the mechanical design aspects of the part length rods since except for the absorber sections they are the same as standard control rods.

The SENA reactor is equipped with the out-of-core long ion chambers similar to those that used for control of the part length rods. From tests conducted in this reactor it will be possible to correlate the power distributions in this core determined by the in-core instrumentation with the information provided by the out-of-core long ion chambers. These tests are expected to demonstrate the adequacy of the out-of-core instrumentation to provide information for the control of the part length rods.

Analytical studies are in progress to investigate the behavior of the reactor under various operational conditions to determine the optimum way to operate the control system.

3 MECHANICAL DESIGN AND EVALUATION

3.1 PART LENGTH ABSORBER ROD DESIGN

The external appearance and design of the part length absorber rod cluster assemblies will be identical to the standard control rod assemblies which incorporate full length absorber material. An outline of the part length absorber rod cluster is shown in Figure 3.1-1.

There will be eight part length control rod clusters located in the core as shown in Figure 3.1-2. These tentative locations are expected to minimize the perturbations of the radial power distribution by the part-length absorber rods. However, optimization studies may alter this arrangement if warranted.

The part-length absorber rods will have cadmium-indium silver alloy only in the bottom 36 inches of the absorber tube. The remainder of the tube, excluding the end expansion and void space, is filled with either pelletized or extruded aluminum oxide which has a very small neutron absorption cross-section.

The tubing is sealed at the bottom and the top of welded end plugs. Sufficient diametral and end clearance are provided to accommodate relative thermal expansion and to limit the internal pressure to acceptable levels.

The absorber and aluminum oxide stack is held in position within the tube for shipment by a carbon steel wire spring which is not required to function under reactor operating conditions.

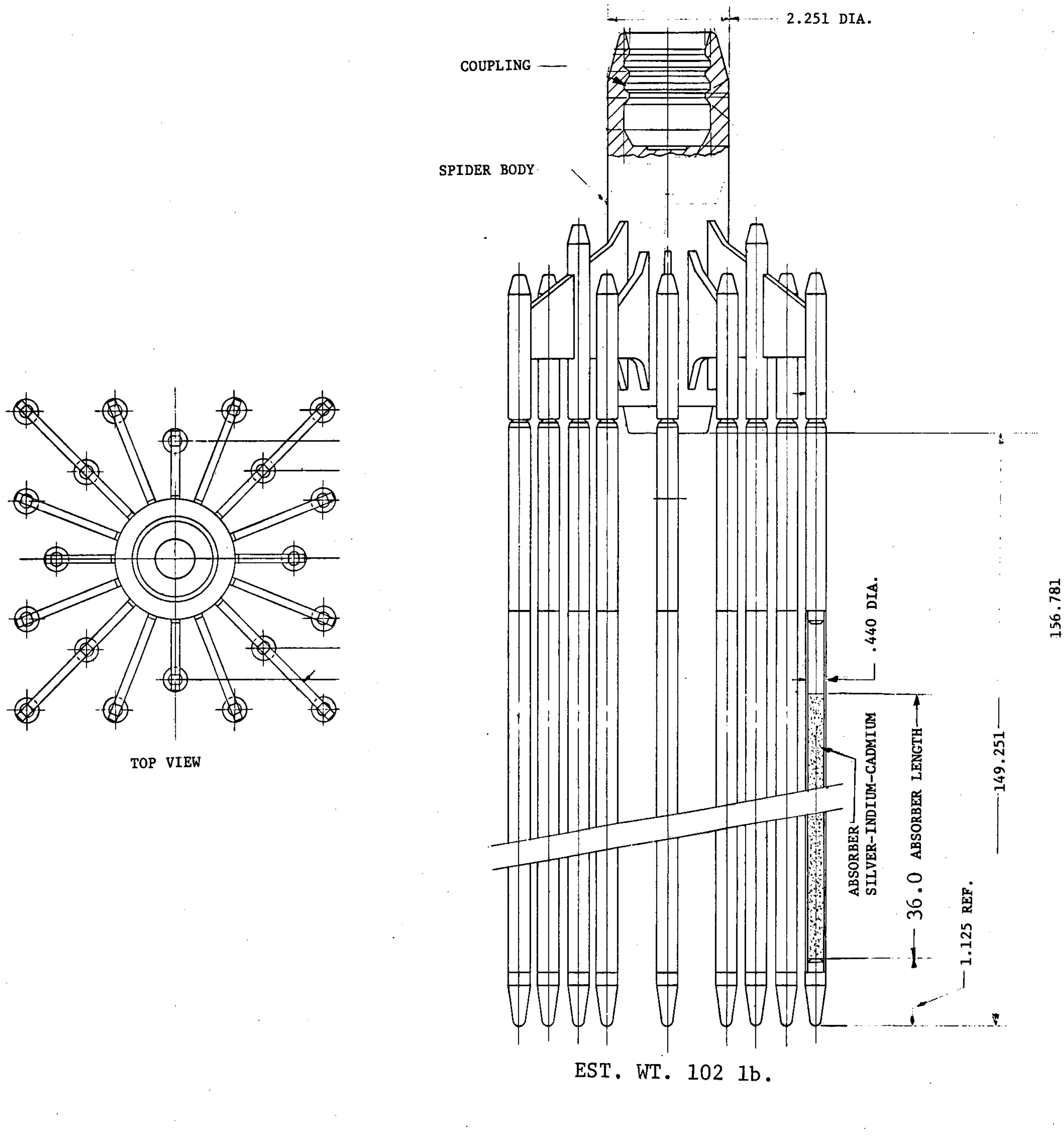
The assembly consists of a cluster of twenty individual part length absorber rods, grouped by fastening to a common hub called a spider. The part length absorber rods are inserted directly into the fuel assembly where guide thimbles are provided to assure free passage. This arrangement is similar to that used for the standard RCC assemblies.

The only physical characteristic of the part length rod absorber assembly that is different from the standard RCC is that each assembly will be approximately 65 pounds lighter than the standard RCC design.

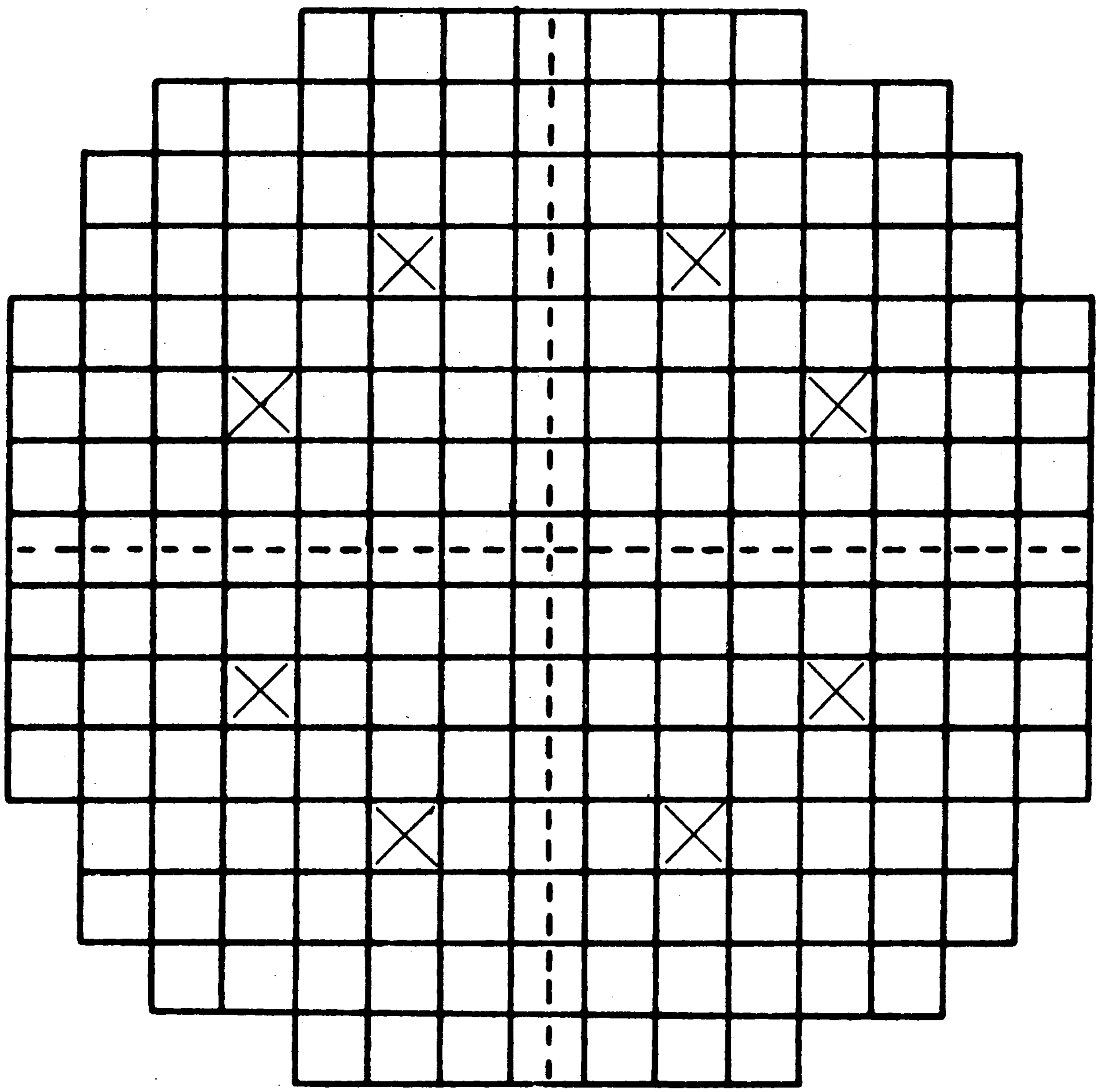
Prototype tests have shown that the standard RCC assemblies are very easily inserted and not subject to binding even under conditions of severe misalignment. Performance of the part length rods will be the same as the standard rods.

All other components, including the stainless steel cladding, end plugs, spider, and the methods of joining and attachment are identical to the standard full length absorber control rod assembly.

Stainless steel clad part-length silver-indium cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions.



ROD CLUSTER CONTROL ASSEMBLY
FIGURE 3.1-1



Part Length Rods

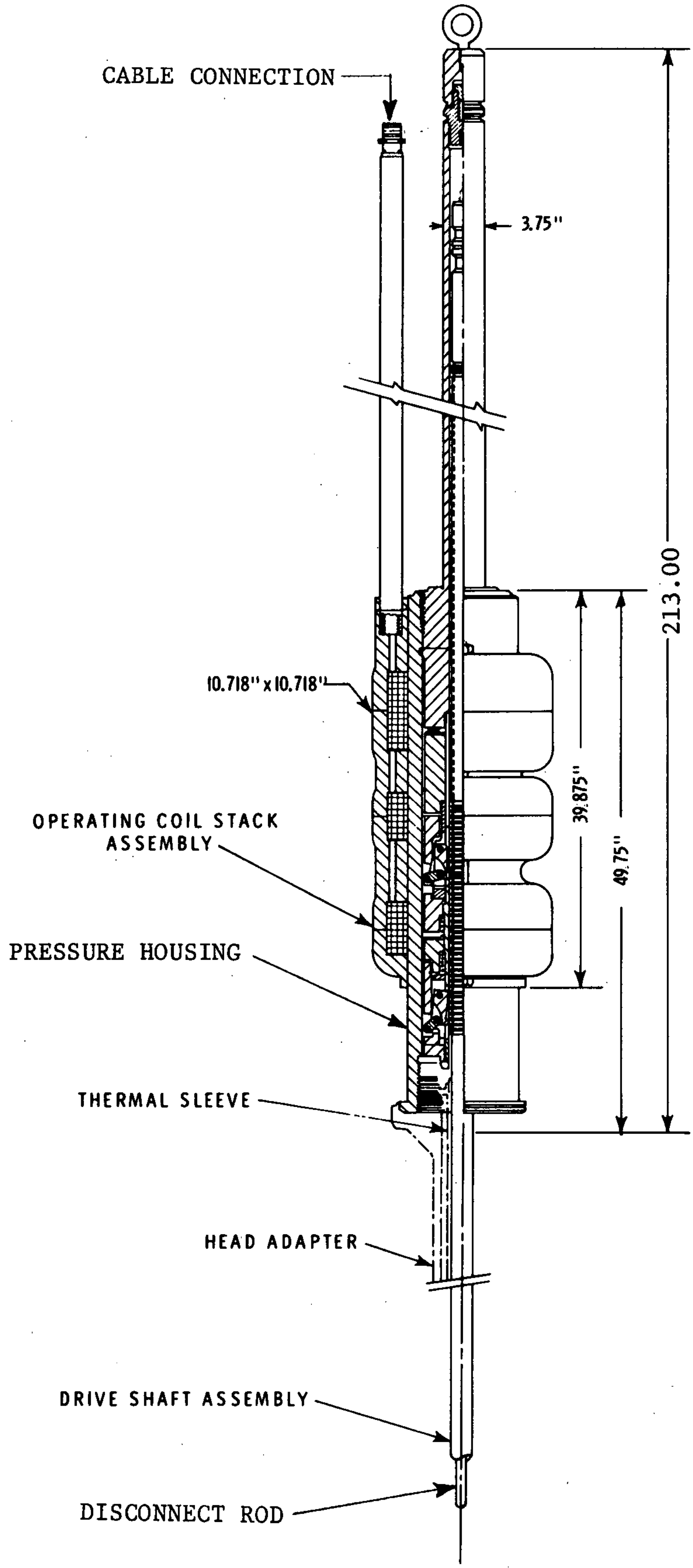
FIGURE 3.1-2

Location of Part Length Control Rod

3.2 CONTROL ROD DRIVE MECHANISM

Standard control rod drive mechanisms are used for withdrawal and insertion of the part length absorber rod cluster control assemblies into the reactor core and to provide sufficient holding power for stationary support.

These drive mechanisms are identical to those used for the standard absorber rods. The complete drive mechanism, shown in Figure 3.2-1 consists of the internal (latch) assembly, the pressure housing, the operating coil stack, the drive shaft assembly, and the position indicator coil stack. Operation of the mechanisms is similar to that of the standard rods except that provisions are made in the control circuitry to prevent trip of the part length rods.



CONTROL ROD DRIVE MECHANISM ASSEMBLY

FIGURE 3.2-1

4 NUCLEAR DESIGN

4.1 REACTIVITY EFFECTS

There will be eight part length control rods located in the core as shown in Figure 3.1-2. The reactivity worth of eight full length rods at these positions would be about 1% Δk_{eff}. The reactivity worth of the eight part length rods is dependent on their axial position but in their most reactive position is on the order of 0.3% Δk_{eff}. Dropping the part length rods to the bottom of the core may increase reactivity by as much as 0.2% Δk_{eff}. The power supply for the part length rods and provisions to prevent inadvertent reactivity additions are discussed in Section 6.

4.2 CONTROL OF AXIAL XENON OSCILLATIONS BY PART LENGTH RODS

Under certain circumstances the action of control rod banks and the shift of the axial water density distribution with power can induce a xenon oscillation in the axial direction. These oscillations can be controlled and excessive axial hot channel factors avoided by the use of part length control rods.

To illustrate the ability to control an oscillation with part length rods, a series of calculations were performed. The calculations were made with the FAB-8 code with the following assumptions.

1) The power and flux in the z direction were calculated using a one-dimensional two-group representation with 20 core regions.

2) The nuclear constants at each mesh point were adjusted to include the effects of density changes related to the enthalpy rise at that point. (Integral of source up to that point.)

3) The Xe-135 and I-135 concentrations are calculated at each point as a function of time.

4) The fuel temperature coefficient (Doppler effect) was included by adjusting the group one absorption coefficient at each point. The fuel temperature coefficient has a $T^{-1/2}$ behavior with a full power value of $-1.0 \times 10^{-5}/°F$.

5) The calculation proceeds in time by a series of discrete (2-4 hr.) steps with the flux distributions assumed constant over the time interval. Zero time steps are made to initialize new conditions.

6) Long term burnup is represented by regionwise burnup in each of the 20 core regions using burnup tables.

To illustrate an uncontrolled oscillation, the following case (A) was considered: The core was burned in 1000 MWD/MT time steps with a power level of 3250 MWt for 12,000 MWD/MT with the enthalpy feedback in the calculation. The resulting power distribution is somewhat displaced to the bottom of the core and has a peak value F_z of 1.17. The core at this stage in burnup is in the condition of maximum instability to axial xenon oscillations. A 1% control bank is inserted half way into the core and power reduced to make the system critical. After two hours the control bank is removed, and the core returned to full power. The power shape and xenon distributions were calculated at four-hour intervals. The axial hot channel factor (F_z) as a function of time is shown in Figure 4.2-1. Without control the system exhibits divergent oscillations.

To illustrate that the part length rods will control the oscillation the preceding case was repeated (Case B) but with part length rod control imposed after the control bank was removed. The control strategy involved moving the part length rods every four hours in course steps (8 in) to equalize the power in the top and bottom halves of the core. The axial hot channel factor (F_z) is shown as a function of time in Figure 4.2-2. Note that the oscillations shown in Figure 4.2-2 are now completely damped out and the calculated value of F_z is always less than 1.35 after the control bank is removed. In a third case (Case C) the preceding case was repeated but with part length rod control imposed 20 hours before the system is perturbed by the control bank. For this case it is necessary to impose the administrative control that the part length bank be kept between the control bank and the bottom of the core. The axial hot channel factor F_z is shown in Figure 4.2-3. When the control bank is inserted, the axial hot channel factor is only 1.40 with the reactor power reduced to 65%. After the control bank is removed the hot channel factor is never above 1.25.

The above cases illustrate the ability of the part length rods to damp a xenon oscillation induced by a simple load change encountered in base load operation. Additional studies are in progress to determine what if any administrative controls are necessary to allow load changes under various conditions.

The part length rods are not related to the control of azimuthal or X-Y oscillations. The stability of the core to X-Y oscillations is discussed in a report on burnable poisons which is in preparation.

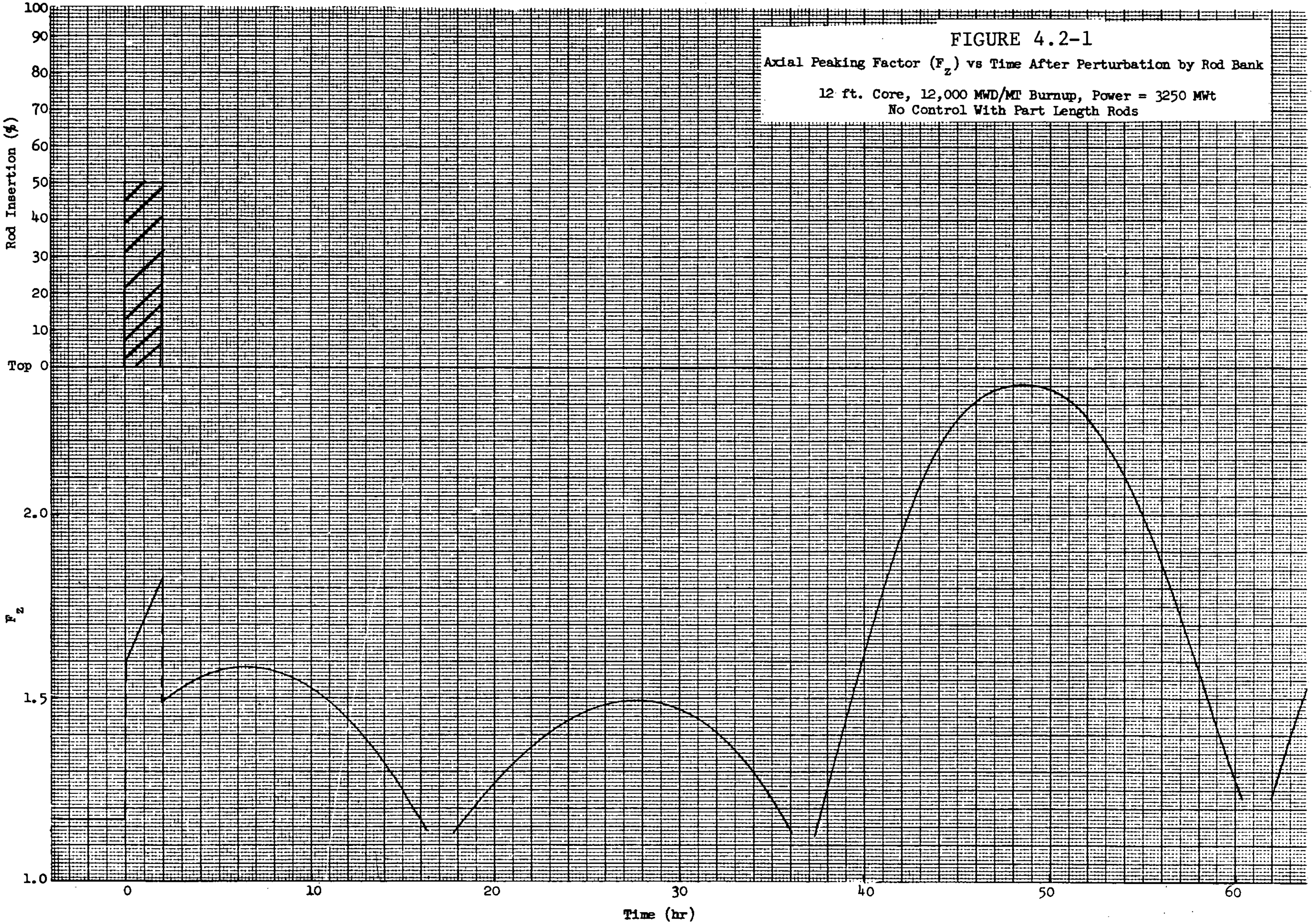


FIGURE 4.2-1
Axial Peaking Factor (F_z) vs Time After Perturbation by Rod Bank
12 ft. Core, 12,000 MWD/MT Burnup, Power = 3250 MWt
No Control With Part Length Rods
Rod Insertion (%)
100
90
80
70
60
50
40
30
20
10
Top 0
F_z
2.0
1.5
1.0
0
10
20
30
40
50
60
Time (hr)

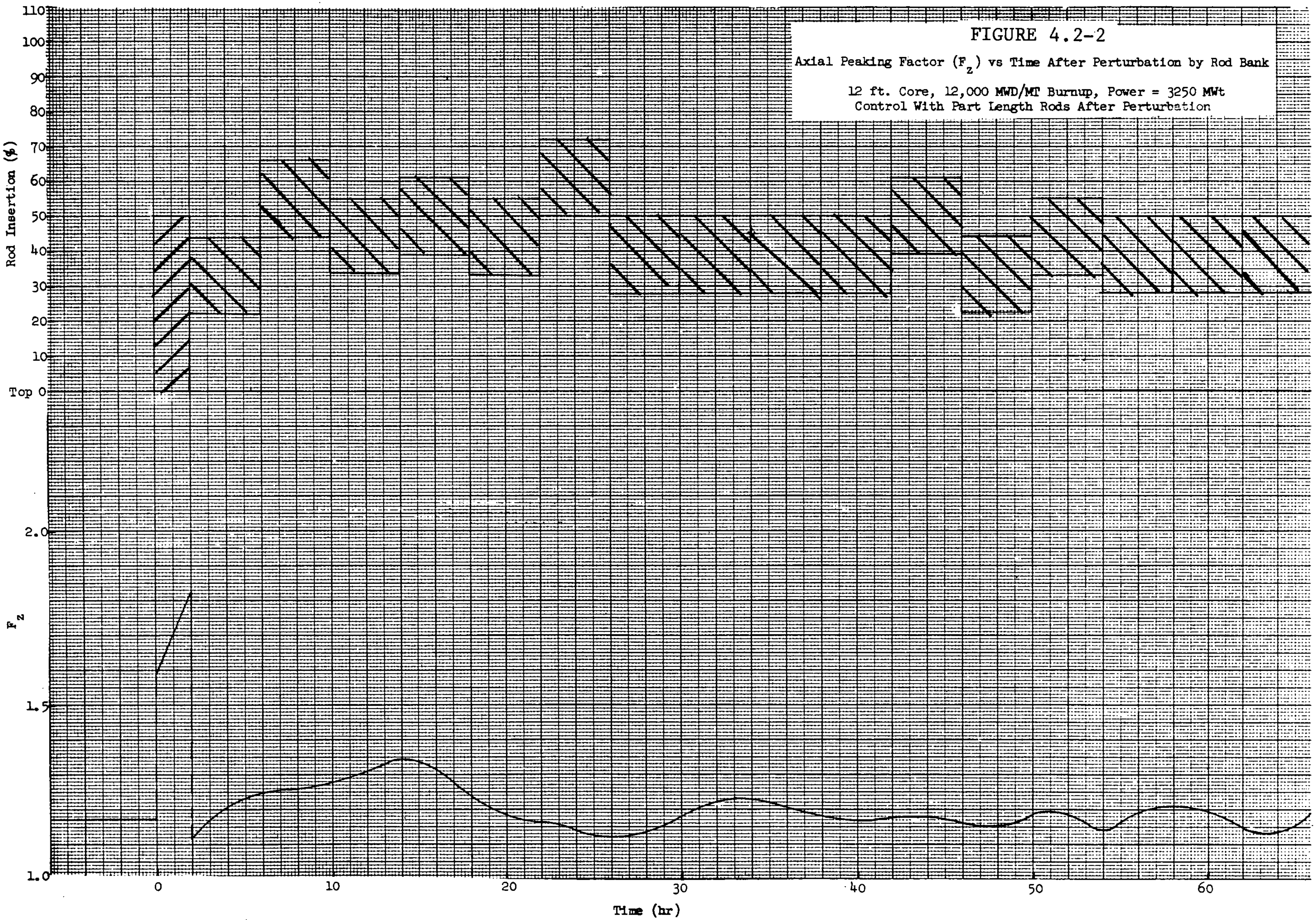


FIGURE 4.2-2
Axial Peaking Factor (F_z) vs Time After Perturbation by Rod Bank
12 ft. Core, 12,000 MWD/MT Burnup, Power = 3250 MWt
Control With Part Length Rods After Perturbation
Rod Insertion (%)
110
100
90
80
70
60
50
40
30
20
10
Top 0
F_z
2.0
1.5
1.0
0
10
20
30
40
50
60
Time (hr)

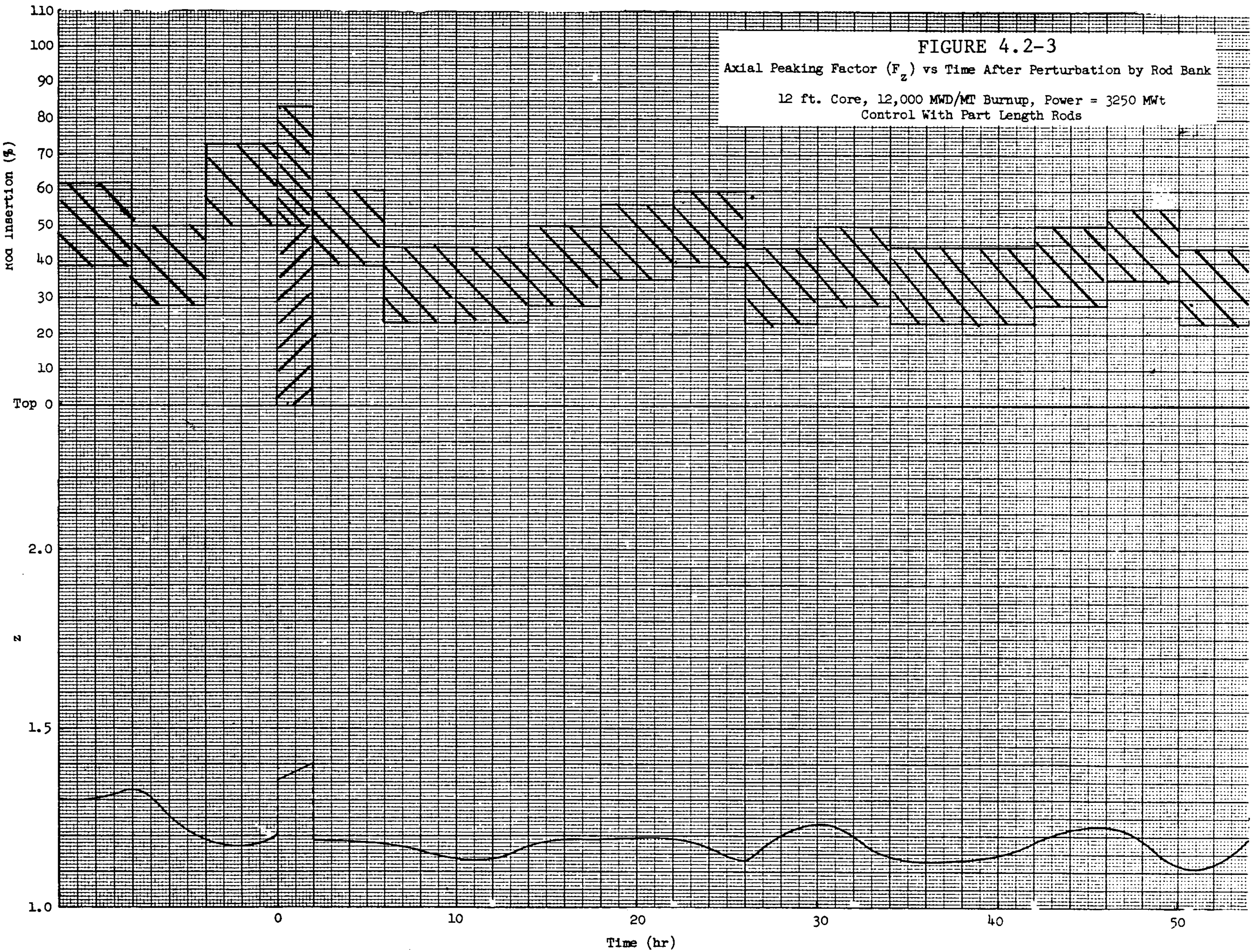


FIGURE 4.2-3
Axial Peaking Factor (F_z) vs Time After Perturbation by Rod Bank
12 ft. Core, 12,000 MWD/MT Burnup, Power = 3250 MWt
Control With Part Length Rods
Rod Insertion (%)
110
100
90
80
70
60
50
40
30
20
10
Top 0
2.0
z
1.5
1.0
0
10
20
30
40
50
Time (hr)

4.3 POWER SHAPING AND HOT CHANNEL FACTORS

The hot channel factors for the reactor $F_{\Delta h}$ = 1.70 ($F_{\Delta h}$ nuclear = 1.58) and F_q = 2.82 are lower than those quoted for some earlier reactors. A major factor in attaining these lower hot channel factors is the incorporation of part length rods in the design. These rods provide a means of obtaining a better axial hot channel factor and controlling the location of the peak power. A power peak at the bottom of the core gives a higher DNB ratio than one at the top of the core, for the same peak to average ratio.

The hot channel factors F_z and $F_{\Delta h}$ do not themselves represent design limitations but must be considered in their relationship to the physical design limits of Kw/ft of fuel rods (dependent upon F_q) and the DNB (departure from nucleate boiling) ratio.

The part length rods provide a means of lowering the axial hot channel factor substantially below the design value. With a lower F_z a higher value of $F_{\Delta h}$ can be allowed and still maintain design margins. The relationship between acceptable values of F_z and $F_{\Delta h}$ is shown in Figures 4.3-1 for the reference design case of a cosine axial distribution where the DNB ratio is 1.38 and F_q is 2.82 or less. (The minimum allowable DNBR is 1.30). For this case a reduction of 3% in F_z and $F_{\Delta h}$ is more nearly 1.5 to one. The relationship between F_z and $F_{\Delta h}$ for a typical power distribution with a peak at the bottom of the core (P – μ sin μ) is shown in Figure 4.3-2. The DNB ratio for this case is 1.52.

Calculations of the axial power distribution for various conditions illustrate the effectiveness of the part length rods.

In the beginning of life F_z for the unrodded core would be 1.4. With the part length rod inserted near the center of the core the value of F_z would be 1.27. As the core is depleted the axial distribution tends to flatten and the unperturbed F_z would be less than 1.2.

It is only when the control bank is inserted in the core a substantial distance that axial peaking is of concern. In typical base load plant operation the power is reduced when control rods are inserted more than about 20% of the way into the core. The axial peak power location is always in the unrodded portion of the core where the radial hot channel factor is low.

Detailed analysis of the power distribution under various transient conditions is in progress. Results to date indicate that for all normal operating cases F_z is less than 1.50. Calculated values of $F_{\Delta H}$ are less than 1.50 in the unrodded core and 1.65 in the rodded portion of the core. This conbination of hot channel factors gives values of F_q below and DNB ratios well above design limits.

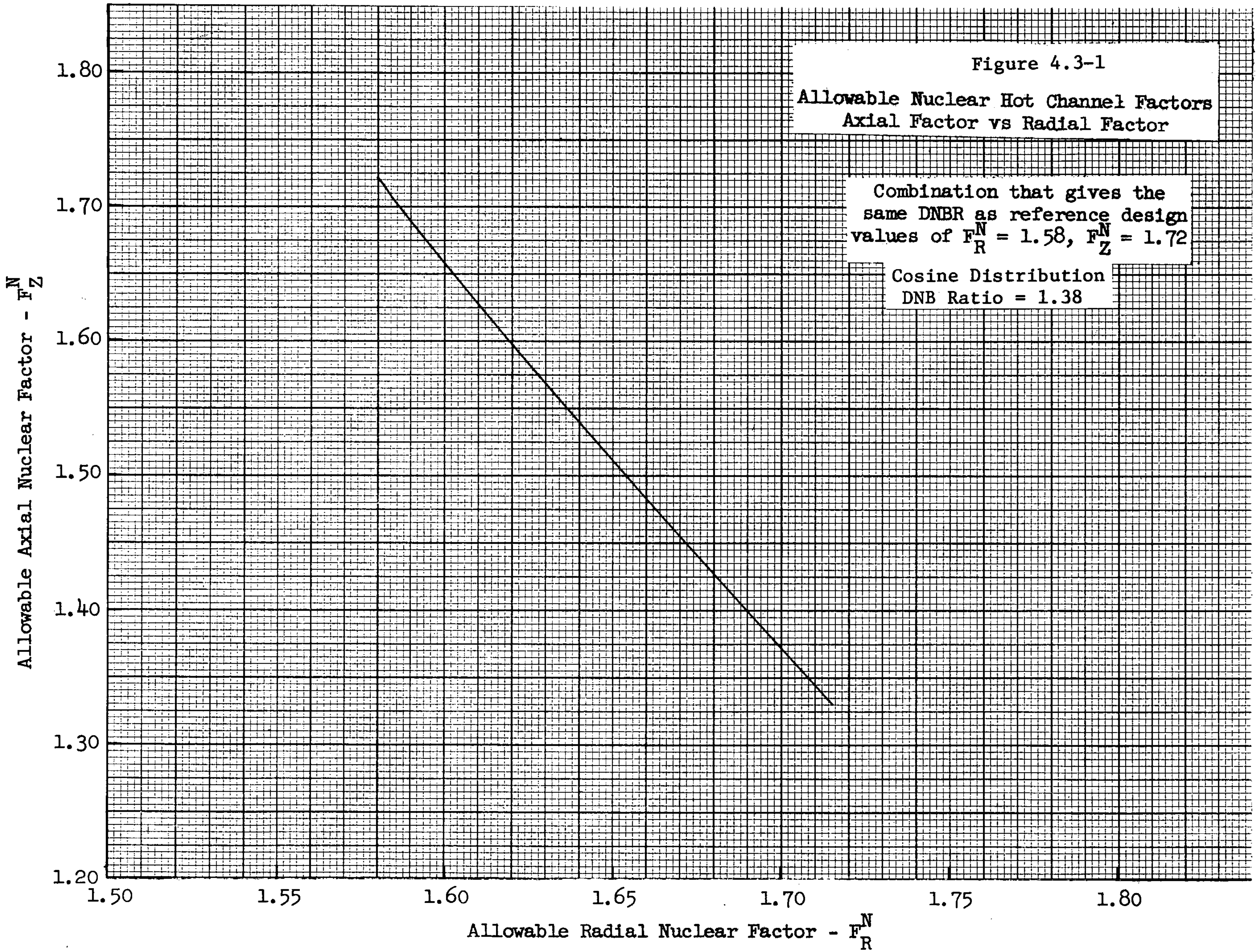


Figure 4.3-1
Allowable Nuclear Hot Channel Factors
Axial Factor vs Radial Factor
Combination that gives the same DNBR as reference design values of $F_R^N = 1.58$, $F_Z^N = 1.72$
Cosine Distribution
DNB Ratio = 1.38
1.80
1.70
1.60
1.50
1.40
1.30
1.20
Allowable Axial Nuclear Factor - F_Z^N
1.50
1.55
1.60
1.65
1.70
1.75
1.80
Allowable Radial Nuclear Factor - F_R^N

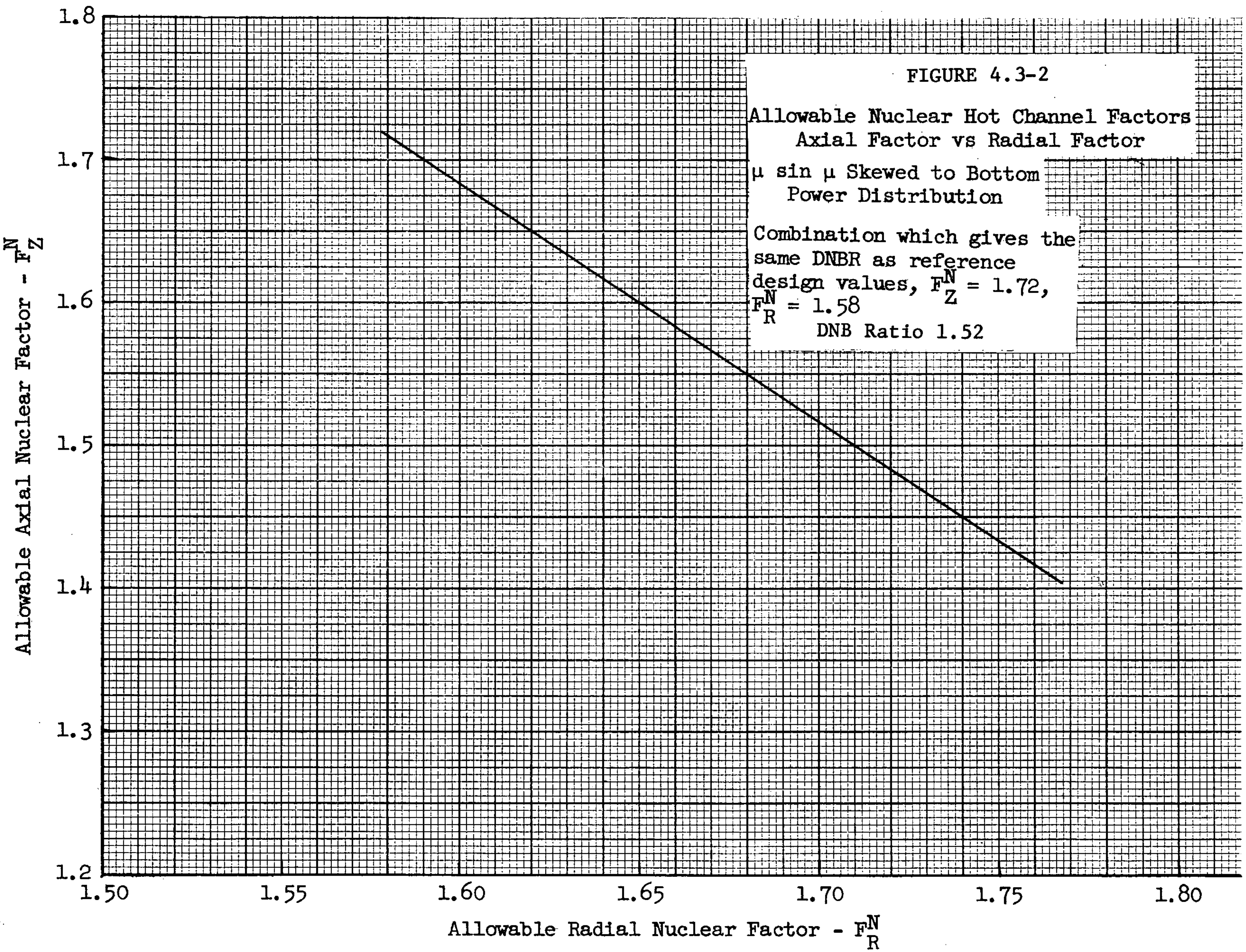


FIGURE 4.3-2
Allowable Nuclear Hot Channel Factors
Axial Factor vs Radial Factor
μ sin μ Skewed to Bottom
Power Distribution
Combination which gives the same DNBR as reference design values, $F_Z^N = 1.72$, $F_R^N = 1.58$
DNB Ratio 1.52
1.8
1.7
1.6
1.5
1.4
1.3
1.2
1.50
1.55
1.60
1.65
1.70
1.75
1.80
Allowable Axial Nuclear Factor - F_Z^N
Allowable Radial Nuclear Factor - F_R^N

5 THERMAL AND HYDRAULIC DESIGN AND EVALUATION

5.1 DESIGN

The design length absorber rods will be designed such that they will perform within acceptable temperature and stress limits. In addition, the presence of these rods will not adversely affect the thermal and hydraulic conditions in the fueled regions in the core.

The primary hydraulic criterion for core design is to prevent departure from nucleate boiling (DNB) during normal operation and includes such conditions that may occur during normal system perturbations. A safety margin from DNB during normal operation is maintained by setting a minimum allowable DNB ratio. The reactor control and protection system is designed to provide actuation of automatic reactor trip to prevent expected plant transients from producing core conditions which would give a DNB ratio lower than 1.30.

A secondary criterion is that there shall be no melting of the fuel during any anticipated operating condition.

5.2 EVALUATION

The fuel temperatures, DNB ratios, coolant void fractions, flow velocities and distributions will be evaluated to insure that thermal and hydraulic design criteria established are not violated.

The adequacy of the standard RCC design has been demonstrated in development tests including in-pile Saxton tests and in operation of the SCE reactor. From a thermal and hydraulic standpoint, it is not expected that the partially heated length of the part length absorber rods would introduce any problems. Sufficient coolant flow inside the thimble provides adequate cooling for these rods. The by-pass flow required is not excessive insofar as the fuel bundle flow requirements are concerned. No thermal problems caused by neutron flux peaking are expected in the fuel bundle during normal operation or in loss of coolant accidents due to the addition of the partial length absorber rods.

6 CONTROL AND INSTRUMENTATION

6.1 PART LENGTH ROD CONTROL SYSTEM

The control system for part length rods is composed of two principle equipment groups, the detection and information group and the part length rod controller and power supply group. Actual operation of the part length rods is performed manually by the operator.

The control system design will be in accordance with the IEEE "Standards for Nuclear Power Plant Protection."

6.2 DETECTION AND INFORMATION GROUP

Basic Control Scheme

The basic control scheme employs out of core nuclear instrumentation in the form of split long ion chambers. Upper and lower ion chamber currents are compared, properly normalized, and displayed to the operator. The part length rods are positioned by the operator to maintain a prescribed relationship between the upper and lower chamber readings. The part length rod travel is limited to a prescribed region to maintain a monotonic relationship between part length rod position and ion chamber readout information. Alarms will be used to inform the operator when these conditions are violated.

The principle behind the control scheme is quite straight forward and is as follows: If the axial power shape is controlled to balance the power generated in the upper and lower halves of the core, the perturbations imposed on axial power distribution by xenon can be essentially eliminated. In a similar manner, if the relation between full length rod insertion and part length rod travel is controlled to maintain the effectiveness of the part length rods, then the perturbations imposed on axial power distribution by rod insertion can also be controlled. The effectiveness of part length rods in reducing the axial peaking factor has already been demonstrated. In these studies the part length rods were positioned to achieve a minimum in F_Z. Since the information for control is not F_Z the results will be somewhat different than shown in Section 4.2. This difference is exhibited in the following.

Assume the comparison between upper and lower ion chambers is the ratio of the upper chamber current to the lower chamber current. Assume also that the long ion chamber currents bear a one to one correspondence with the axial power generation. The behavior of the core in terms of F_Z max, axial position of peak power generation and xenon concentration, and ratio of power generated in the upper half to the lower half are given in Figure 6.2-1. This case is the uncontrolled case discussed in Section 4.2. These curves demonstrate the effect of the xenon perturbation on axial power generation both in terms of peak power and power sharing. The sensitivity of F_Z max to the power ratio is shown more clearly in Figure 6.2-2. From this result one can make the argument that controlling the relationship of ion chamber currents to some prescribed relationship should maintain F_Z max within a certain bound.

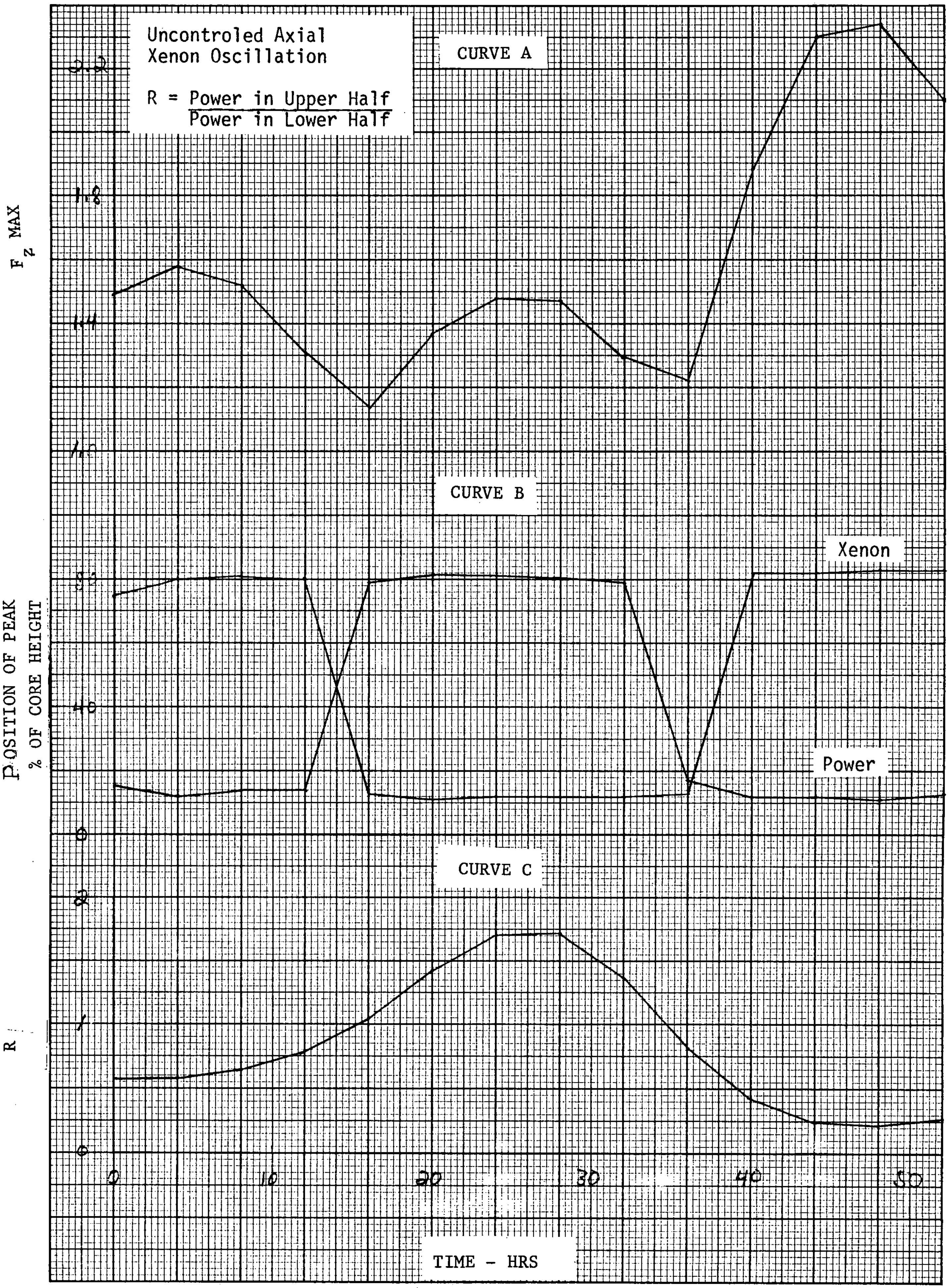
Results of an analytical check on the conceptual control system are given in Figure 6.2-3. The case shown is the same as shown in Figure 4.2-3. A comparison between results given in that figure and the results for the control scheme where a balance in power sharing is maintained is given in Figure 6.2-3a. This comparison shows that while the F_Z is somewhat larger for the control scheme it is still well contained and the xenon oscillations are, for all intensive purposes, completely damped.

Of interest also is the behavior of F_Z at the extreme limits of part length rod motion. For this study the partial length rod motion was restricted to the center third of the core, that is the center of the part length rod was maintained between $\pm$ 2 feet of the core center. Results in Figure 6.2-3b show that F_Z max while considerably improved over the uncontrolled case is higher than desired when the part length rods are at their top limit. However as can be seen from Figure 6.2-3c, the power ratio is very low. Restricting the minimum power into 0.8 yields a max F_Z of 1.5 or less which is within the desired value. It should be emphasized here that the principle mode of part length rod control studied was the long ion chamber readings and that the position restriction was imposed to eliminate ambiguous information in the ion chamber readings.

Long Ion Chamber Effectiveness

The preceeding discussion has indicated the desirability of establishing a correlation between axial power distribution and out-of-core detector readings. Experience on operating power plants with in-core instrumentation, namely, Yankee and Selni, has shown qualitative agreement between out-of-core detector readings and in-core power behavior. When the axial power generation in the core is balanced the out-of-core detector readings are also balanced. There is reasonable repeatability to these results. The detector readings also show the relative relationship of power generation between the upper and lower core halves for unbalanced conditions. The establishment of a definite quantitive relationship between in-core distribution and out-of-core readings would permit more flexibility in the control system. In this regard Westinghouse is presently conducting tests at the the SENA plant to determine the correlation between out-of-core nuclear instrumentation readings and in-core power generation.

Initial results from the SENA program show that for a condition where the ratio of upper and lower core power generation the gross core ratio was 0.45 the long ion chambers indicated 0.51 or about 13% less unbalance. These results were obtained with the control group inserted about 55% and are about as expected since the long ion chambers should see a flatter power shape than the core average axial shape. Sufficient information has not been obtained to permit a correlation between axial peaking factor and the long ion chamber readings. At this time the conclusion can be drawn that the results look reasonable.



Uncontroled Axial
Xenon Oscillation
R = Power in Upper Half / Power in Lower Half
CURVE A
2.2
1.8
1.4
1.0
F_z MAX
CURVE B
Xenon
Power
80
40
0
POSITION OF PEAK
% OF CORE HEIGHT
CURVE C
2
1
0
R
0
10
20
30
40
50
TIME - HRS

FIGURE 6.2-1

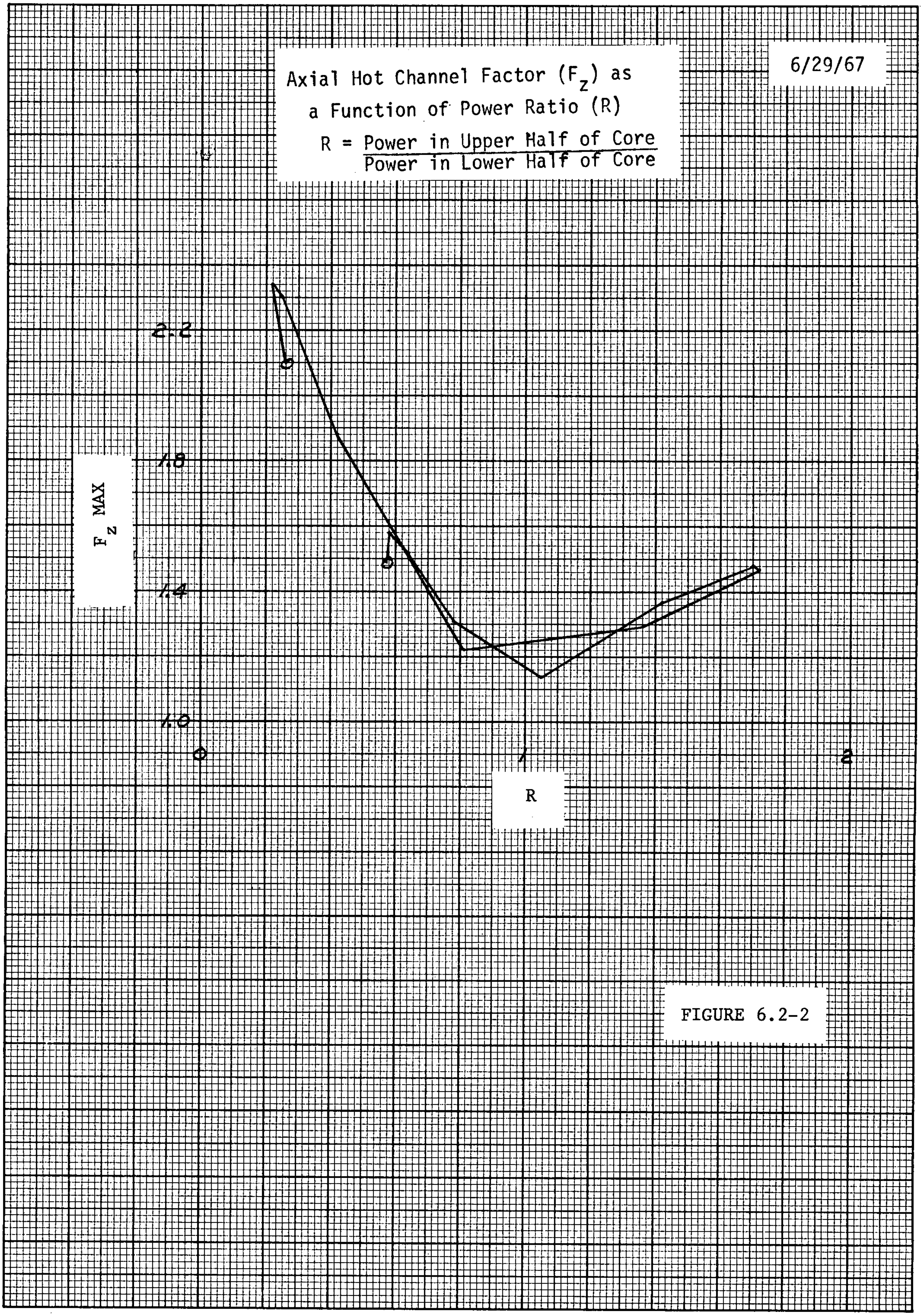
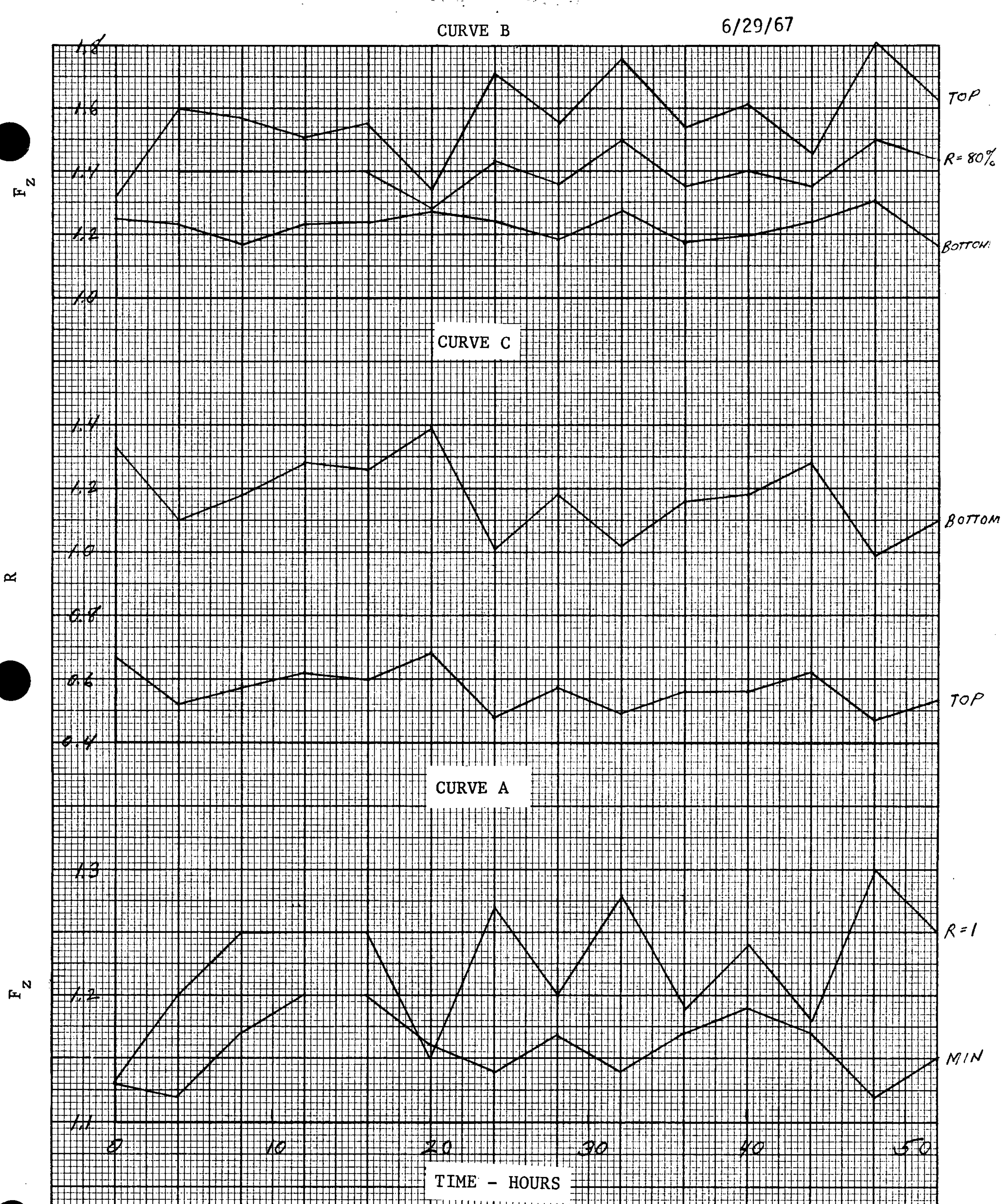

6/29/67
Axial Hot Channel Factor (F_z) as
a Function of Power Ratio (R)
$R = \frac{\text{Power in Upper Half of Core}}{\text{Power in Lower Half of Core}}$
2.2
1.8
1.4
1.0
F_z MAX
0
1
2
R

FIGURE 6.2-2



Axial Hot Channel Factor and Power Ratio (R)

VS Time With Part Length Rod Control

$$R = \frac{\text{Power in Upper Half of Core}}{\text{Power in Lower Half of Core}}$$

FIGURE 6.2-3

6.3 ROD CONTROLLER AND POWER SUPPLY GROUP

The part length rods are operated by the conventional Westinghouse magnetic jack mechanism that is used to control the full length rods.

Since the part length rods will normally be in a region of fairly high neutron worth tripping the rods will add reactivity and cause a power excursion. The rod controller and power supply for the part length rods will be arranged so that single failures or loss of a single power source will not result in a trip of all the part length rods.

In particular these requirements will be met by

1) Use of two independent energy storage supplied separate from the supplies for the full length rods.

2) Redundant holding circuits when the rods are not in motion both the movable gripper and the stationary gripper are energized and from separate sources

3) Staggered movement for groups of part length rods.

With these features a single failure will result in the tripping of only two out of the eight part length rods and then only if it occurs simultaneous with rod motion.

The energy sources will be monitored and rod motion will not be permitted if either energy source shows a failure.

7 SAFETY ASPECTS

7.1 SUMMARY AND SAFETY CONSIDERATIONS

As discussed in the previous sections, the part length rods are employed to improve the axial power distribution as well as to control potential axial xenon oscillations.

Calculations will be performed based upon a range of full length rod positions and power levels to determine the effectiveness of the part length rods. Periodic in-core measurements will be performed to verify these calculations as well as to correlate the out-of-core long ion chamber readings with respect to in-core conditions.

The axial distribution will be continuously monitored by the upper and lower sections of out-of-core ion chambers. Out-of-limit signals will be generated should the relative readings differ by a preset amount. Corrective action will be taken, e.g., stop loading or power cutback to assure that core limits are maintained in the event proper rod motion is not executed.

Since the part length rods are employed to assist in controlling the axial power distribution, consideration must be made of the potential dropping of the part length group. This could lead to potential positive reactivity insertion of as much as 0.2% and an axial flux distortion. This small increase in power coupled with the power distribution could lead to possible DNB. The part length rod drive power supply will be designed to prevent a trip of the part length rods in the event of a reactor trip. Detailed analysis of the consequences of dropped part length rod(s) will be made and the power supply will be designed such that no core damage will occur for a single failure in the system.

Detailed calculations are in progress to establish by analysis the relationship between the in-core and out-of-core flux distribution and to develop the ground rules for operator control of the part length rods. On-line tests are being performed on the SENA reactor to evaluate the long ion chamber performance.

During detailed design of the core, credible and hypothetical accident conditions will be evaluated. Typical accidents and criteria particular to part length rods are listed below.

TYPICAL ACCIDENTS AND CRITERIA

	Accident	Type	Criteria
a)	Dropped Rod	Credible	No DNB
b)	Undesired Rod Insertion	Credible	No DNB
c)	Ejected Rod	Hypothetical	No vessel overpressure
d)	Undesired Rod Withdrawal	Credible	No DNB

8 IDENTIFICATION OF CONTRACTORS

Westinghouse Atomic Power Division (WAPD) will be responsible for the design, manufacture and assembly of the part length absorber rod clusters and associated control, instrumentation and drive mechanisms, and for the preparation of administrative guides for operator control of the part length absorber rods during initial tests and normal plant operation.

APPENDIX B

WCAP-7113

USE OF BURNABLE POISON RODS IN
WESTINGHOUSE PRESSURIZED WATER REACTORS

October 1967

WCAP - 7113

USE OF BURNABLE POISON RODS
IN
WESTINGHOUSE
PRESSURIZED WATER REACTORS

October 1967

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USE OF BURNABLE POISON RODS IN WESTINGHOUSE PRESSURIZED WATER REACTORS

1. INTRODUCTION AND SUMMARY

Burnable poison rods will be added to the core to eliminate a positive moderator reactivity temperature coefficient at operating temperature conditions during the initial portion of the first fuel cycle. With the addition of the burnable poison rods the spatial power distribution will be stable with respect to azimuthal and diametral xenon oscillations, and moderator reactivity addition that could otherwise occur in the event of a loss of coolant or rod ejection accident during the early portion of the first core cycle will be eliminated lessening the severity of these accidents.

The burnable poison rods will be in the form of borated pyrex glass tubes clad in stainless steel. There will be approximately 1144 of these rods grouped in clusters which will be distributed throughout the core in vacant rod cluster control guide tubes. These rods will initially control about 7.2% $\frac{\Delta k}{k}$ of the installed excess reactivity and their addition will result in a reduction of the initial operating boron concentration in the coolant to 1200 ppm. The moderator coefficient is negative at this boron concentration with the burnable poison rods installed. The poison is depleted as the core accumulates burn up during operation. The poison depletion rate is sufficiently slow that the boron requirement in the coolant and hence the moderator temperature coefficient will be less than their initial values throughout the entire initial cycle. In succeeding cycles, the core boron requirement is less due to the partially depleted fuel loading and the poison rods are not required to maintain the moderator coefficient negative and will be removed.

The following paragraphs describe the evaluation and development of the preliminary design for the poison rods. The numerical calculations presented in Sections 3.1 and 3.2 were performed with initial enrichments of 2.25, 2.8, and 3.42 wt% U-235. Since these calculations were performed, the enrichments have been changed to 2.2, 2.7,and 3.2 wt% U-235 as a result of recent data from operating reactors. The effect of this change will be a 3% increase in flux level and a reduction of initial boron concentration from 1300 ppm to 1200 ppm which will make the moderator temperature coefficient $0.1 \times 10^{-4}/°F$ more negative. The net result of this change should improve stability.

2. RESEARCH AND DEVELOPMENT

An evaluation of nuclear requirements and material properties for the burnable poison rods has been completed and borosilicate glass has been selected as the reference design burnable poison material.

A series of critical experiments has been performed at the Westinghouse Reactor Evaluation Center to evaluate the reactivity worth and effect on power distribution of borosilicate glass. The experiments were performed using 2.72% enriched UO_2 clad in Zircaloy. The reactivity worth was measured for solid glass rod and two thicknesses of glass tubing, for several ratios of glass to fuel. The B_2O_3 content of the glass was 12.8 wt%. The fuel element configuration with glass rods was simulated in the critical experiments and the power distribution measured.

These experiments are presently being evaluated to test the adequacy of calculational techniques. Each critical loading will be calculated using two dimensional diffusion theory with methods identical to those used in the core design. Preliminary results from the evaluation of the experiments indicate that the design methods predict the reactivity worth of the glass to within 0.1% Δk_{eff}.

The configuration and mechanical design of the poison rod has been selected. A program for in-pile testing two of the rods in the Saxton Reactor has been initiated to verify mechanical performance of the burnable poison material and rod configuration in a power reactor environment.

3 REACTOR

3.1 DESIGN BASIS

The function of the burnable poison rods is to absorb neutrons which would otherwise be absorbed by soluble poison in the coolant. In effect, the poison rods will "control" a portion of the excess reactivity normally "controlled" by the boric acid in the coolant and the boric acid requirement will be less. The result will be that changes in density of the coolant will have less effect on the density of poison and the moderator reactivity temperature coefficient will be reduced. The details of the quantity, com-position, location, and structure of the burnable poison rods will satisfy the following limits on the design.

a) The moderator temperature coefficient of reactivity will be negative at operating coolant temperature.

b) Induced aazimuthal and diametral xenon oscillations will be damped.

c) Power distribution, shutdown margin, fuel and clad temperature, and minimum DNB ratio will be within the design limits previously established.

d) Materials, construction, and support of the poison rods will insure integrity of the poison rods for both normal operating conditions, and accident conditions.

e) The poison rods will be held down in place by a spider assembly compressed beneath the upper core plate to ensure that they cannot be lifted out of the core by flow forces.

Table I summarizes design and operating characteristics of the burnable poison rods.

Xenon Oscillations in the X-Y Plane (Diametral and Azimuthal Oscillations)

Because the coolant flow is transverse to the X-Y plane, changes in the moderator temperature distribution influence the stability of the core to xenon oscillations in the X-Y plane. As the power shifts to one side of the core the average temperature in that side increases. If the moderator temperature coefficient of reactivity is negative this results in a negative feedback which tends to stabilize the power distribution. On the other hand, if the moderator temperature coefficient is positive, the feedback is destabilizing and the core is more subject to oscillations.

An extensive program sponsored by the AEC and Euratom (AEC Contract AT(30-1)-3680) has been initiated to develop methods for the control of xenon instabilities in large PWR's. Methods to incorporate the effects of coolant temperature distribution into a two dimensional calculation are being developed. As a survey tool a relatively simple one-dimensional method has been developed to investigate the effect of variables on stability. This one-dimensional model has been compared to two dimensional cases and gives a conservative result.

TABLE I

PRELIMINARY DESIGN AND OPERATING CHARACTERISTICS OF BURNABLE POISON ROD CORE

Mechanical Design	
Borosilicate Glass - Composition (wt%)	SiO_2-80.5, Al_2O_3-2.2, Na_2O-3.8, K_2O-0.5, B_2O_3-12.5
I.D.	.243
O.D.	.396
Length	142.7
External Clad	
I.D.	.4005
O.D.	.4395
Internal Support Clad	
I.D.	.2235
O.D.	.2365
Guide Thimble	
I.D.	.515
O.D.	.545
Total Number of Poison Rods	1128
Poison Rod Location (See Figs. 3-5 and 3-6)	
Poison Rods per Assembly	6, 12, 16
Nuclear Design	
Poison Rod Boron Content (gm/cm)	.043
Initial Worth of Poison Rods-hot full power (% Δk/k)	7.2
Initial worth of single poison rod (% Δk/k)	.007
Initial worth of single poison rod cluster (% Δk/k)	.11
Poison Rod Depletion Rate (See Fig. 3-9)	
Moderator Reactivity Temperature Coefficient (at power)	-0.5×10^{-4} to $-3\times10^{-4}\,°F^{-1}$
Initial Moderator Boron Requirements	1200 ppm
First Cycle Enrichments (wt% U-235)	2.2, 2.7, 3.2

TABLE I (Cont'd)

Installed Reactivity	K_{eff} Without Poison Rods	K_{eff} With Poison Rods
Cold, No Power	1.291	1.225
Hot, No Power	1.244	1.172
Hot, Full Power	1.220	1.148
Hot, Full Power, Xe and Sm	1.178	1.106

Rod Thermal Design

Average B_{10} (n, α) reaction heating (kw/ft)	0.467
Peak B_{10} (n, α) reaction heating (kw/ft)	1.32
Average gamma heating (watts/gm)	1
Peak gamma heating (watts/gm)	3
Peak glass temperature (°F)	1190

Summary of Operating Clad Stress and Pressure for Burnable Poison Rods	Beginning of Life	End of Life
Primary Pressure Stress, psi	-24,870	2650
Secondary Pressure Stress, psi	±7610	±440
Maximum Pressure Stress, psi	-32,480	-3090
Thermal Stress, psi	1280	380
Operating Differential Pressure, psi	-2250	-240
Critical Buckling Differential Pressure, psi	-4650	-4650
Safety Factor Based on Buckling	2.07	11.34

The calculational method assumes a one dimensional slab core having the same physical width as the diameter of the actual core and with a fuel enrichment distribution to give a flattened power distribution. The flux and power distribution in the core is calculated using FAB-8, a two group diffusion theory code. The fuel temperature coefficient (doppler effect) is taken into account by adjusting the fast absorption cross section at each point in the mesh to correspond to the relative power at that point. The calculation is iterative with the Doppler correction recalculated until a convergent solution is obtained. Similarly, the effects of water density are taken into account by adjusting the nuclear constants at each point to correspond to the relative enthalpy at that point. The coolant density and effects of xenon-135 and iodine-135 are calculated at each mesh point.

To study the stability of the core, a perturbation is introduced at one side of the core by adding a fixed poison equivalent to one control rod and letting the perturbed core operate for 1 hour to redistribute the xenon. The perturbation is then removed and the power shape in the reactor is calculated at two or three hour time intervals, with the feedbacks from moderator temperature, xenon, and Doppler effects included. The results of the calculations with the two-three hour time steps are corrected back to a zero time step using the results of Poncelet, et. al.[1]

It is possible to adjust the moderator temperature coefficient in the calculation by adjusting the fixed absorption cross section of the core and letting the code search for a critical boron concentration.

The behavior of the diametral hot channel factor as a function of time after the perturbation is shown in Figure 3-1 for the case of a core with a positive moderator temperature coefficient ($\alpha_m = 0.10 \times 10^{-4}/°F$) and

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(1) Poncelet, C. G., Christie, A. M., "The Effect of Finite Time Step on Calculated Spatial Stability Characteristics," Trans. Am. Nuc. Soc., Vol. 10, No. 2, 1967.

in Figure 3-2 for the case of a negative moderator coefficient ($\alpha_m = -0.35 \times 10^{-4}/°F$) at 3250 Mwt. Note that the core with the negative coefficient shows convergent oscillations while the core with the positive coefficient shows divergent oscillations.

The damping factor is shown as a function of moderator temperature coefficient in Figure 3-3 for two power densities, 84.5 kw/liter equivalent to 2758 MWt and 99.6 kw/liter equivalent to 3250 MWt.

The threshold of stability is $\alpha_m = -0.07 \times 10^{-4}/°F$ for the 3250 MWt power level and $+0.4 \times 10^{-4}/°F$ for the 2758 MWt power level.

The damping factor, b, is defined by the equation:

$$P(t) = P_o + Ae^{b(t-t_o)/T} \sin \omega(t-t_o)$$

where:

P_o = unperturbed $F_{\Delta h}$

$P(t) = F_{\Delta h}(t)$

T = period

ω = angular frequency

t = time

b is negative for convergent oscillations and positive for divergent oscillations.

The program to study the X-Y xenon oscillation problem is still in progress but the one dimensional result, demonstrates the dependence of the stability of the core to diameteral oscillations on the moderator reactivity temperature coefficient. Hence, the burnable poison rods will be incorporated in the core during the first cycle to improve stability.

The moderator coefficient of reactivity in the second and subsequent cycles is sufficiently negative to ensure stability.

3.2 REACTOR DESIGN

3.2.1 Nuclear Design and Evaluation

After a review of the elements available for use as burnable poison, it was concluded the boron was the most desirable for this application. Self-shielded Gadolinium was investigated as a possibility, but it was concluded that the rapid decrease of its cross section in the thermal region gives rise to a positive moderator coefficient contribution that more than offsets the improvement obtained by lowering shim boron concentration. The incorporation of boron or its compounds in the UO_2 fuel was investigated and found not to be practical with normal manufacturing practice. It was concluded that the most straightforward way of incorporating burnable poison in the core is to clad some boron containing material in rods and to introduce these rods into the RCC water holes not used by the RCC assemblies. Borosilicate (Pyrex) glass has been selected as the reference design burnable poison material.

A study was performed to select a boron loading which would provide a suitably negative moderator temperature coefficient with the number of burnable poison rods and the fuel cost penalty from residual poison as variables. A boron loading of 0.043 gm B/cm length was selected as near optimum. The moderator temperature coefficient of reactivity for the initial core at 572.9°F was calculated as a function of the number of burnable poison rods in each of 96 fuel assemblies (a checkerboard pattern) and is shown in Figure 3-4. The coolant boron concentration was adjusted to keep the reactor just critical. Since the burnable poison rods act as leakage surfaces, the effect on the moderator temperature coefficient is greater than the effect that would have been anticipated from the reduction in shim boron alone. The reduction in moderator coefficient from shim boron reduction alone is shown in Figure 3-4.

With the borosilicate glass (Pyrex) as burnable poison, a boron loading of 0.043 gm/cm of length is feasible since Pyrex is 12.5 wt% B_2O_3. Since the B_2O_3 is in soolution in the glass, it is homogeneous and particle self-shielding is not a factor.

Power Distribution Studies

From the results of the temperature coefficient studies, a burnable poison loading of approximately 12 rods in every other fuel assembly was chosen. A series of two dimensional (PDQ) calculations of the power distribution was made to find an arrangement that resulted in a satisfactory radial hot channel factor. The final arrangement is shown in Figure 3-5. In this figure, the numbers represent the number of burnable poison rods in the assembly. The location of these rods in the water holes of the assemblies are shown in Figure 3-6. In 24 fuel assemblies, 16 burnable poison rods were used to depress local peaking. In the fuel assemblies at the edges of the core only 6 burnable poison rods were used. The initial assembly-wise power distribution in the core is shown in Figure 3-7. The calculated radial peaking factor ($F^N_{\Delta H}$) is 1.314 compared to the design value of 1.58. The reactivity held down by the burnable poison is 7.2% $\Delta k/k$.

Moderator Temperature Coefficient

The moderator temperature coefficient at 572.9°F was calculated using a two dimensional diffusion code (PDQ), for the arrangement shown in Figure 3-5. The hot, operating coolant boron concentration was 1300 ppm. The calculated value of the coefficient was -0.62×10^{-4}/°F. This value is slightly more negative than the uniform array results presented in Figure 3-4, primarily because the assemblies that have 16 burnable poison rods are in a more important part of the core from a nuclear standpoint. The design value of the moderator temperature coefficient with the burnable poison rods is -0.5×10^{-4}/°F at full power.

Depletion Studies

The behavior of the core with burnable poison during the first cycle has been calculated using the two dimensional depletion code TURBO*. The core was described with a two dimensional array of approximately 7500 mesh points. The fuel isotopic composition and burnable poison concentration are calculated pointwise as a function of burnup. The power distribution in the core was recalculated at 1000 MWD/MT intervals during the first 5000 MWD/MT and then at 2000 MWD/MT intervals. The burnable poison self-shielding factor was changed as a function of burnup.

The coolant boron concentration is shown as a function of burnup in Figure 3-8. The boron concentration drops from an initial value of 1300 ppm to 960 ppm as Xenon-135 builds up. The coolant boron concentration stays relatively constant for 4000 MWD/MT and then decreases nearly linearly.

The concentration of boron in the burnable poison rods is shown in Figure 3-9 as a function of fuel burnup. The average B-10 concentration decreases by a factor of 10 in 10,000 MWD/MT and the residual B-10 at the end of the cycle is only 1/30 of the initial value.

Since the neutron flux at the center of the core is higher than in the outer region, the burnable poison is depleted more rapidly at the center. This causes some redistribution of power toward the center of the core during the early part of the first cycle. The calculated radial hot channel factor ($F_{\Delta H}^{N}$) increases from 1.32 to 1.48 during the first 1000 MWD/MT and then decreases gradually to 1.30 at 4000 MWD/MT. The effects of the fuel temperature coefficient on power distribution were neglected in this calculation and would tend to reduce the calculated radial hot channel factor to 1.42.

Effect of Burnable Poison Addition on Other Design Parameters

Because the coolant boron conccentration is decreased, the rod worth increases slightly. With burnable poison rods, the initial coolant boron requirement is 1300 ppm and the 53 control rods are worth 8.5% as compared to an initial boron requirement of 1960 ppm and 53 control rod worth of 8% without burnable poison rods. At the end of the cycle there is no difference and the control rods are worth 8%.

Since this study was performed the enrichments in the three regions have been changed to 2.2, 2.7, and 3.2 wt% U-235 and the initial boron concentration reduced to 1200 ppm. The corresponding multiplication factors with and without the burnable poison are tabulated in Table I.

3.2.2 Thermal and Hydraulic Design and Evaluation

The core hot spot initially occurs near the core center with the poison rod loading and shifts outward as the poison rods deplete. The hot channel factors and minimum DNB ratio remain within values given in the PSAR. The overall effect of the poison rod addition on coolant flow distribution and core pressure drop is not significant.

The gamma and (n,α) reaction in the poison rods will result in a peak heat generation rate of 1.32 kw/ft. in the poison rods. This heat generation rate will decrease as the poison rod B_{10} depletes. The coolant flow for cooling the poison rods is provided by the flow annulus between the rod and the guide thimble and the exit coolant temperature is maintained below system saturation temperature. The peak temperature in the glass occurs on initial rise to power. Figure 3-10 shows the initial peak axial temperature distribution in the peak poison rod. The Pyrex temperature will decrease rapidly for the following reasons: less power generation due to B_{10} depletion; better gap conductance as He produced diffuses to gas gap; and decreasing external gap due to creep of the Pyrex.

Test specimens of borosilicate glass irradiated at KAPL (Ref. 1) show evidence of high diffusivity of the He generated (greater than 80% He release). Sufficient void has been designed into the poison rod inner cylinder to maintain internal pressure less than external pressure. Assuming complete release of the He generation the calculated internal pressure at end of exposure is approximately 2000 psi.

3.2.3 Mechanical Design and Evaluation

In the reactor cores, the burnable poison rods would be statically suspended and positioned in vacant RCC thimble tubes within the fuel assemblies at nonrodded core locations. The poison rods in each fuel assembly would be grouped and attached together at the top end of the rods by a flat spider plate which fits with the fuel assembly top nozzle and rests on the top adaptor plate. The spider plate (and the poison rods) are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. This ensures that the poison rods cannot be lifted out of the core by flow forces.

In construction, the poison rods will consist of pyrex glass tubes contained within type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass will also be supported along the length of its inside diameter by a thin wall type 304 stainless steel tubular inner liner. A typical burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3-11.

The rods have been designed in accordance with the standard fuel rod design criteria; i.e., the cladding will be free standing at reactor operating pressures and temperatures and sufficient cold void volume has been provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass as a result of the B_{10} (n,α) reaction. The large void volume required for the helium is obtained through the use of glass in tubular form which provides a central void along the length of the rods. The resulting clad stresses at temperature and pressure are given in Table I. With the selected clad diameter and wall thickness the maximum net external pressure that the cladding can withstand is limited by the elastic stability of the tubing.

The critical buckling pressure as established by the elastic stability and the corresponding factor of safety in the clad design based on the net external pressure imposed on the cladding at operating conditions is also listed in Table I.

Based on available data on properties of Pyrex glass and on nuclear and thermal calculations for the rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube is expected to occur but will continue only until the glass comes into contact with the inner liner. The inner liner is provided to maintain the central void along the length of the glass and to prevent the glass from slumping or creeping into the void as a result of softening at the hot spot. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping but to collapse locally before rupture of the exterior cladding if large volume changes due to swelling or cracking should possibly occur. The top end of the inner liner is open to receive the helium which will diffuse out of the glass.

To ensure the integrity of the burnable poison rods, the tubular cladding and end plugs will be procurred to the same specifications and standard of quality as used for stainless steel fuel rod cladding and end plugs. In addition, the end plug seal welds will be checked for integrity by visual inspection and x-ray and the finished rods will be helium leak checked.

4. SAFETY ASPECTS

Analyses of the hypothetical double ended coolant loop rupture loss of coolant accident show that the safety injection system will meet its design criterion of no clad melting with a slightly positive moderator reactivity temperature coefficient. Hence the design coefficient of $-0.5 \times 10^{-4}/°F$ with the burnable poison rods assures margin in clad temperature in meeting this criteria. Similarly the design coefficient is sufficiently negative that any unexpected azimuthal or diametral xenon oscillation would be damped and the radial hot channel factor will remain within design values.

The burnable poison rods are positively positioned in the core inside RCC assembly guide thimbles and held down in place by attachment to a spider assembly compressed beneath the upper core plate and hence cannot be the source of any reactivity transient. Due to the low heat generation rate, and the conservative design of the poison rods there is no possibility for release of the poison as a result of helium pressure or clad heating during accident transients including loss of coolant.

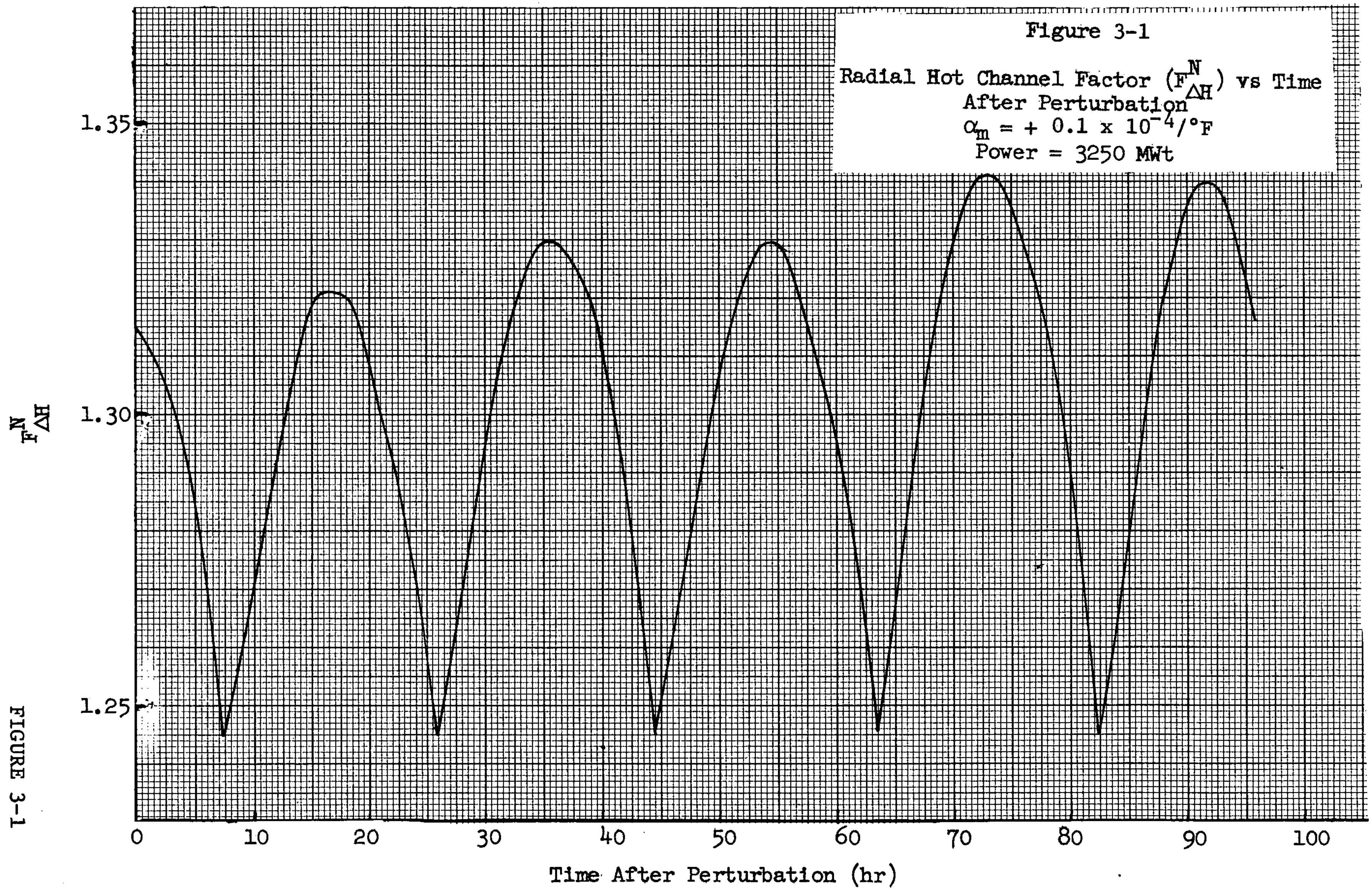


Figure 3-1
Radial Hot Channel Factor ($F^N_{\Delta H}$) vs Time After Perturbation
$\alpha_m = + 0.1 \times 10^{-4}/°F$
Power = 3250 MWt
1.35
1.30
1.25
$F^N_{\Delta H}$
0
10
20
30
40
50
60
70
80
90
100
Time After Perturbation (hr)

FIGURE 3-1

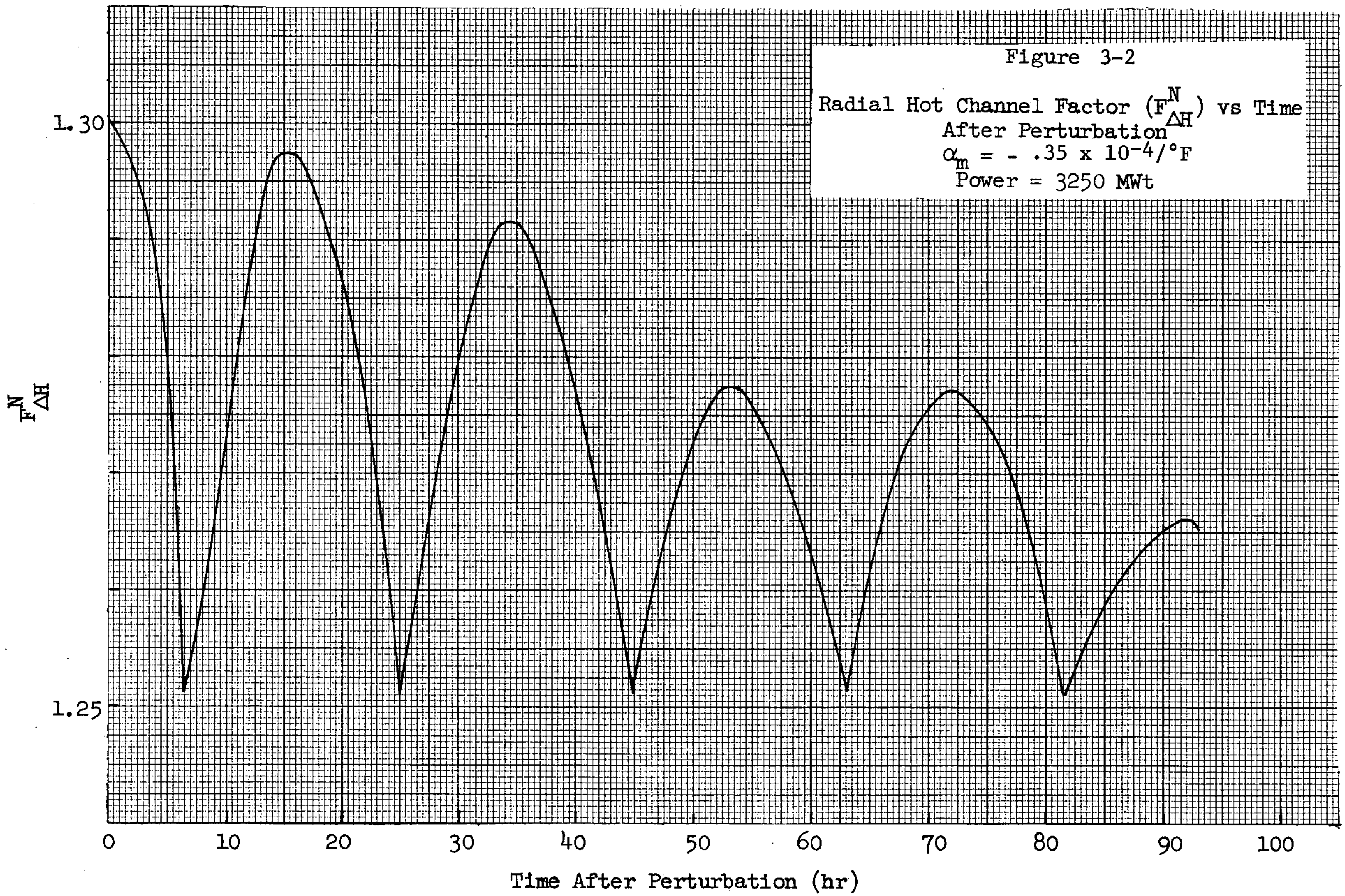


Figure 3-2

Radial Hot Channel Factor ($F^N_{\Delta H}$) vs Time After Perturbation

$\alpha_m = - .35 \times 10^{-4}/°F$

Power = 3250 MWt

FIGURE 3-2

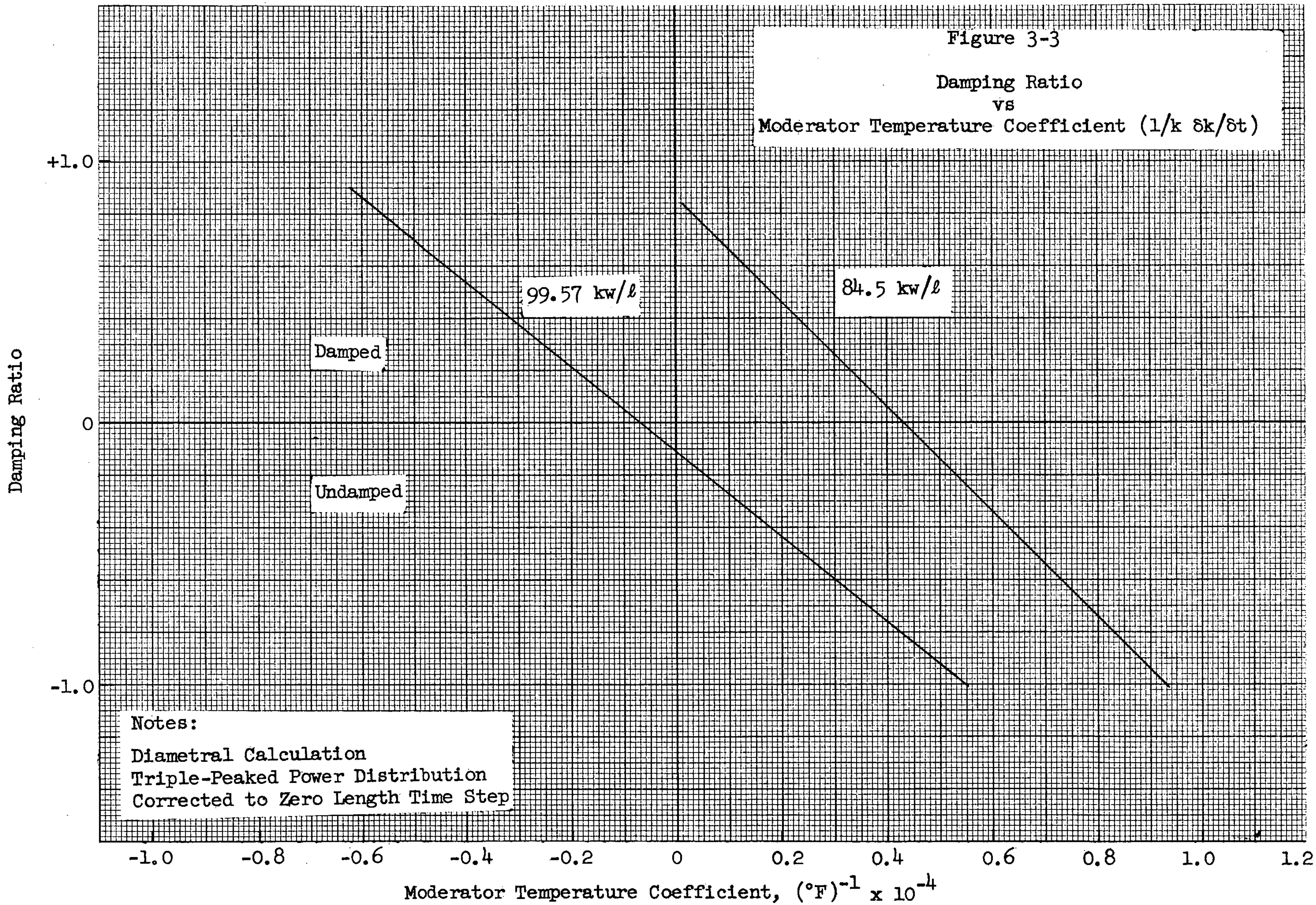


Figure 3-3
Damping Ratio
vs
Moderator Temperature Coefficient (1/k δk/δt)
99.57 kw/ℓ
84.5 kw/ℓ
Damped
Undamped
Damping Ratio
+1.0
0
-1.0
Notes:
Diametral Calculation
Triple-Peaked Power Distribution
Corrected to Zero Length Time Step
-1.0
-0.8
-0.6
-0.4
-0.2
0
0.2
0.4
0.6
0.8
1.0
1.2
Moderator Temperature Coefficient, (°F)$^{-1}$ x 10^{-4}

Figure 3-4

Moderator Coefficient of Reactivity at Beginning of First Cycle versus Number of Burnable Poison Rods in Each of 96 Fuel Assemblies.

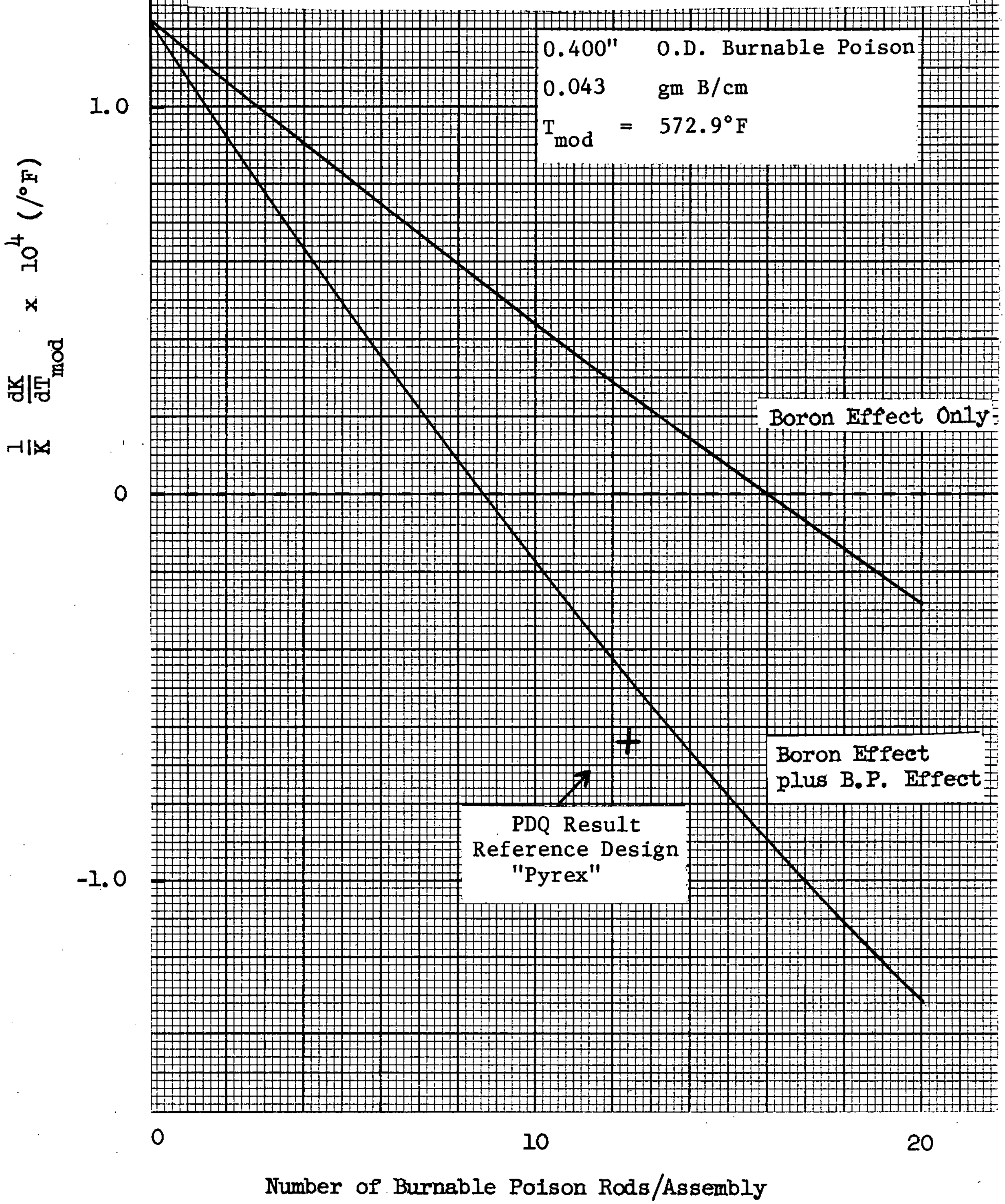


FIGURE 3-4

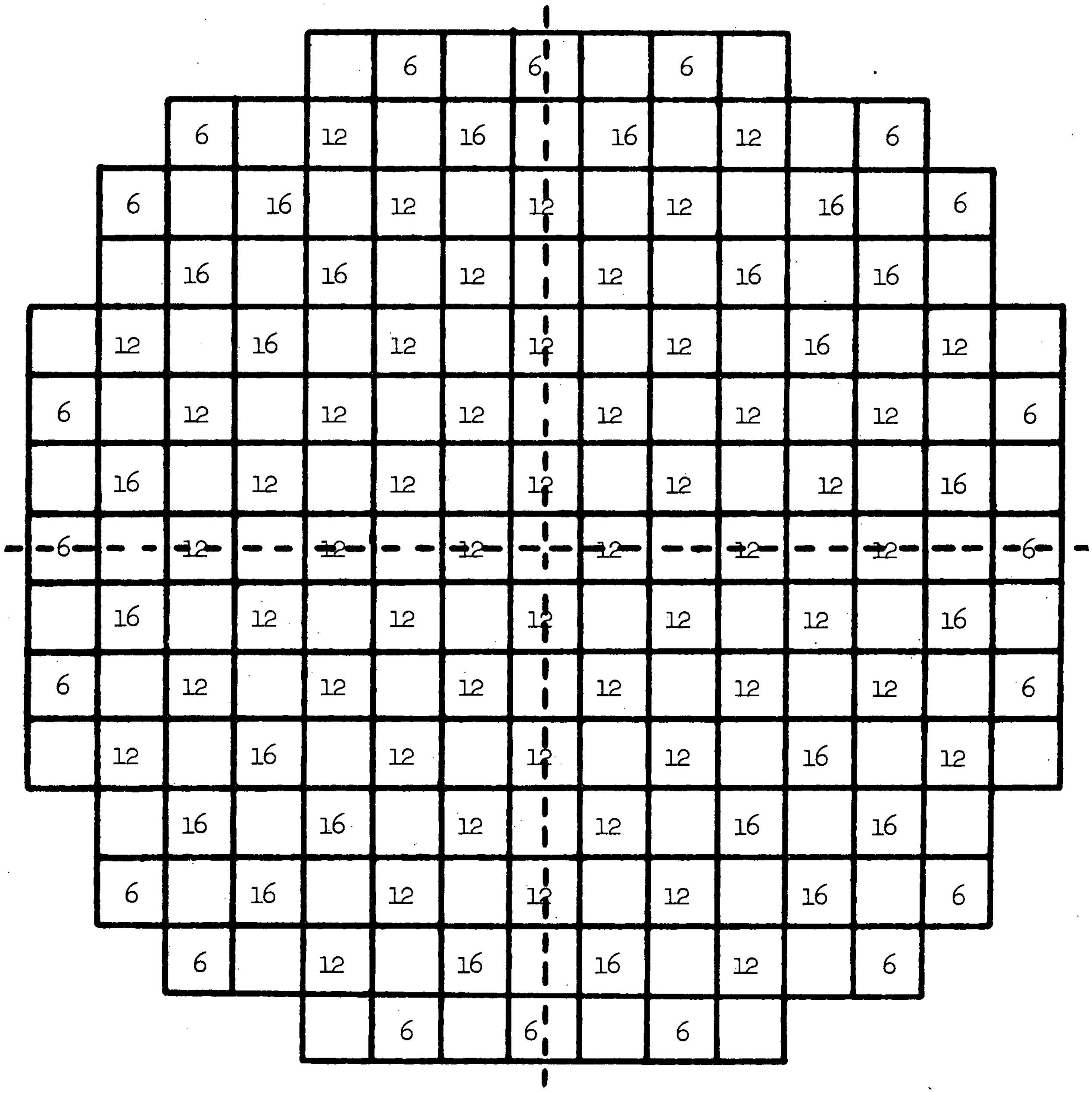
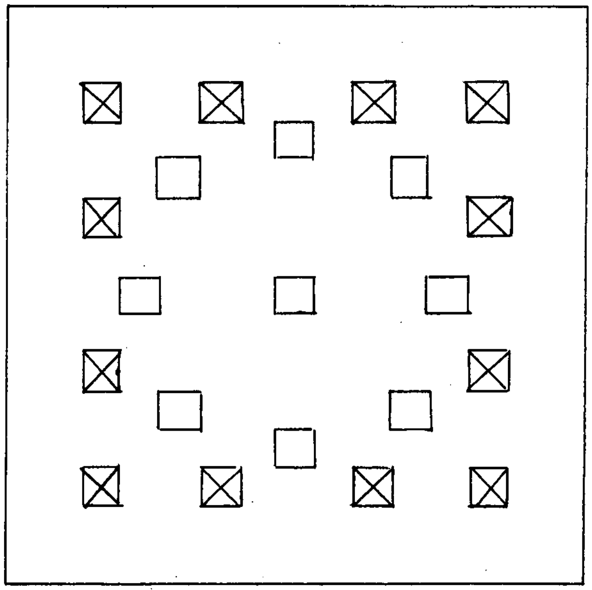


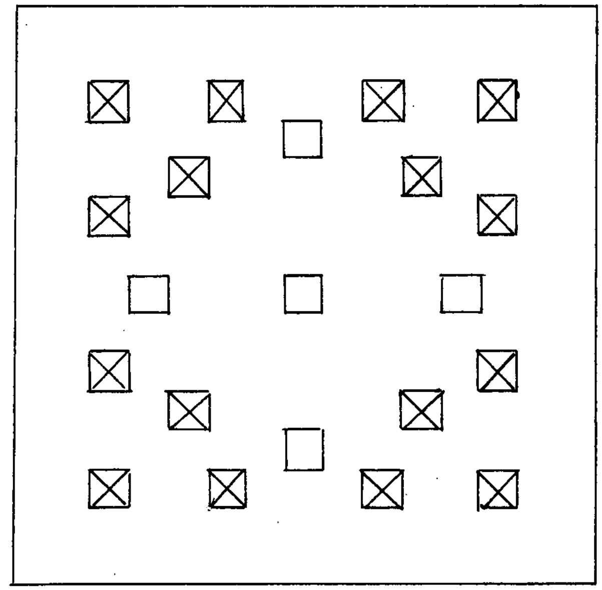
Figure 3-5

Distribution of Burnable Poison Rods -
Number of B. P. Rods per Assembly

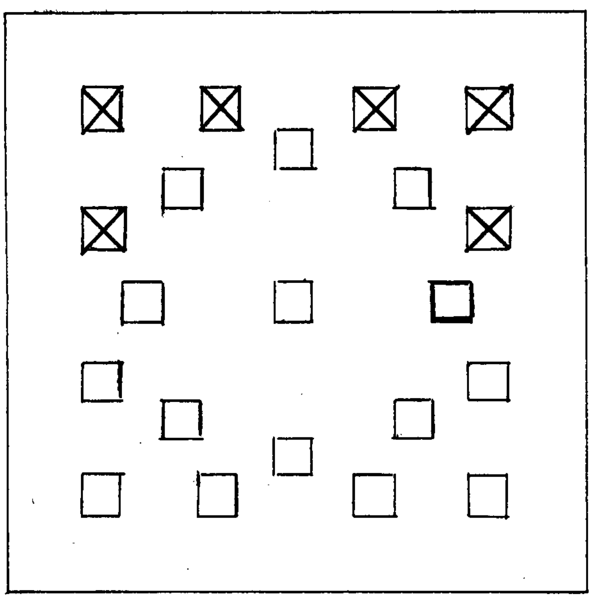
FIGURE 3-5



A. 12 Burnable Poison Rods



B. 16 Burnable Poison Rods



C. 6 Burnable Poison Rods

Note: x denotes burnable poison rods

Figure 3-6

Location of Burnable Poison Rods in Fuel Assemblies

1.244 x	F						
1.206	1.229						
1.216	1.176	1.177					
1.158	1.173	1.104	1.070				
1.155	1.111	1.084	0.962	1.150			
1.096	1.105	1.031	0.995	0.882	1.048		
1.046	1.100	0.968	0.949	0.867	0.548		
0.836	0.891	0.739	0.590				

Figure 3-7

Assemblywise Power Distribution
Burnup = 0 MWD/MT
Shim Boron = 1300 ppm
$F^{n}_{\Delta h}$ = 1.314

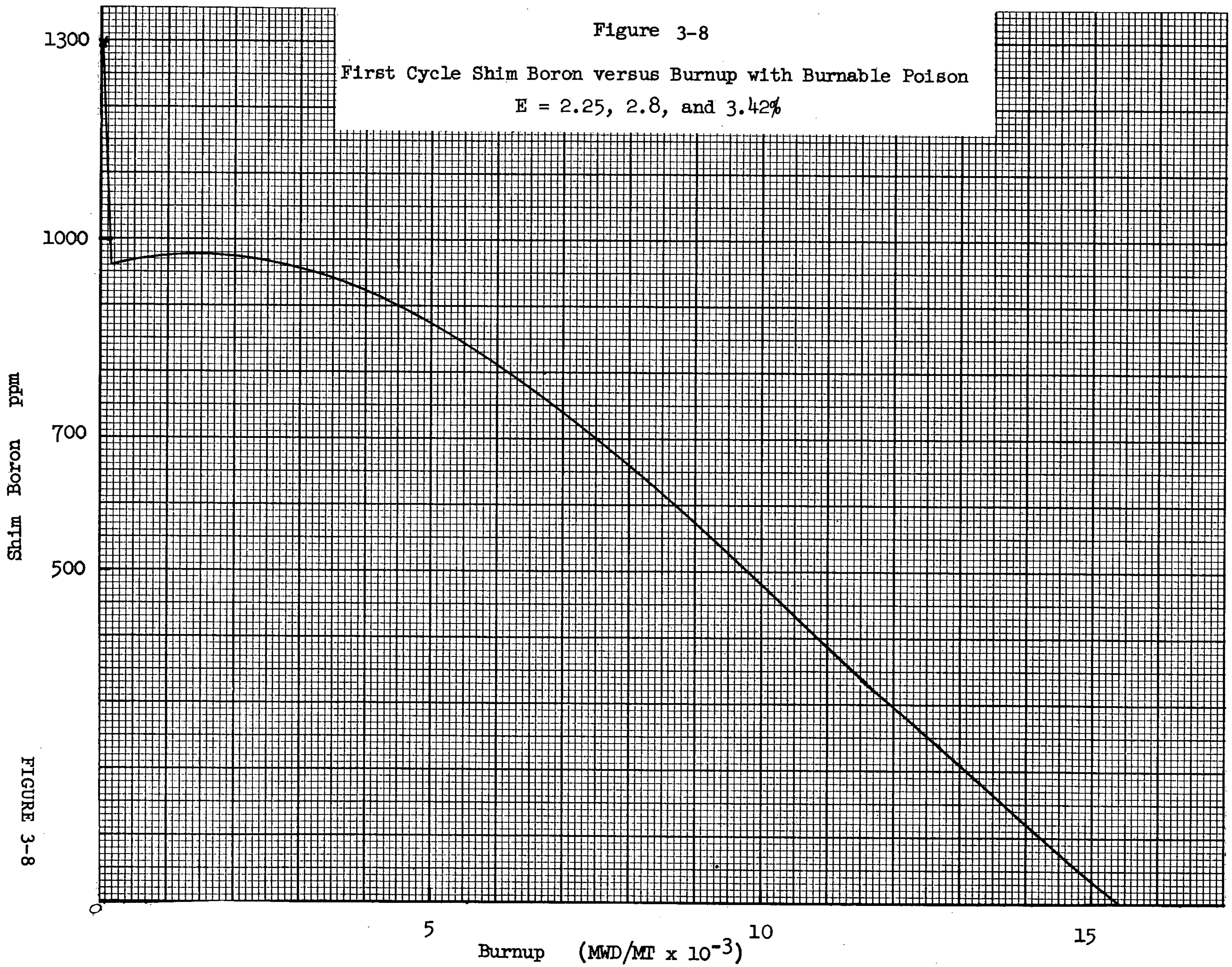


Figure 3-8
First Cycle Shim Boron versus Burnup with Burnable Poison
E = 2.25, 2.8, and 3.42%
1300
1000
700
500
0
Shim Boron ppm
5
10
15
Burnup (MWD/MT x 10-3)

FIGURE 3-8

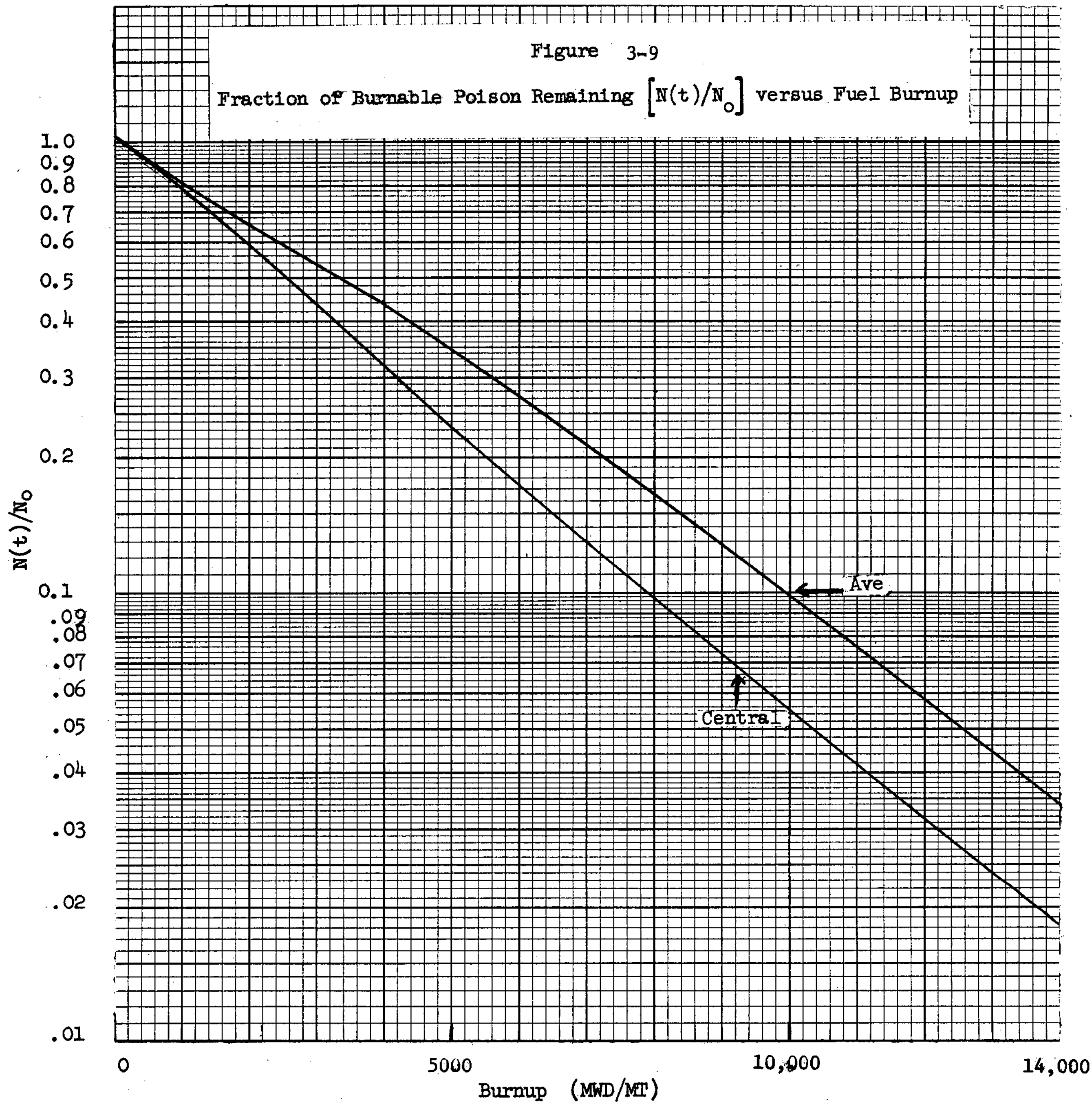


Figure 3-9
Fraction of Burnable Poison Remaining [N(t)/N$_o$] versus Fuel Burnup
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
.09
.08
.07
.06
.05
.04
.03
.02
.01
N(t)/No
Ave
Central
0
5000
10,000
14,000
Burnup (MWD/MT)

FIGURE 3-9

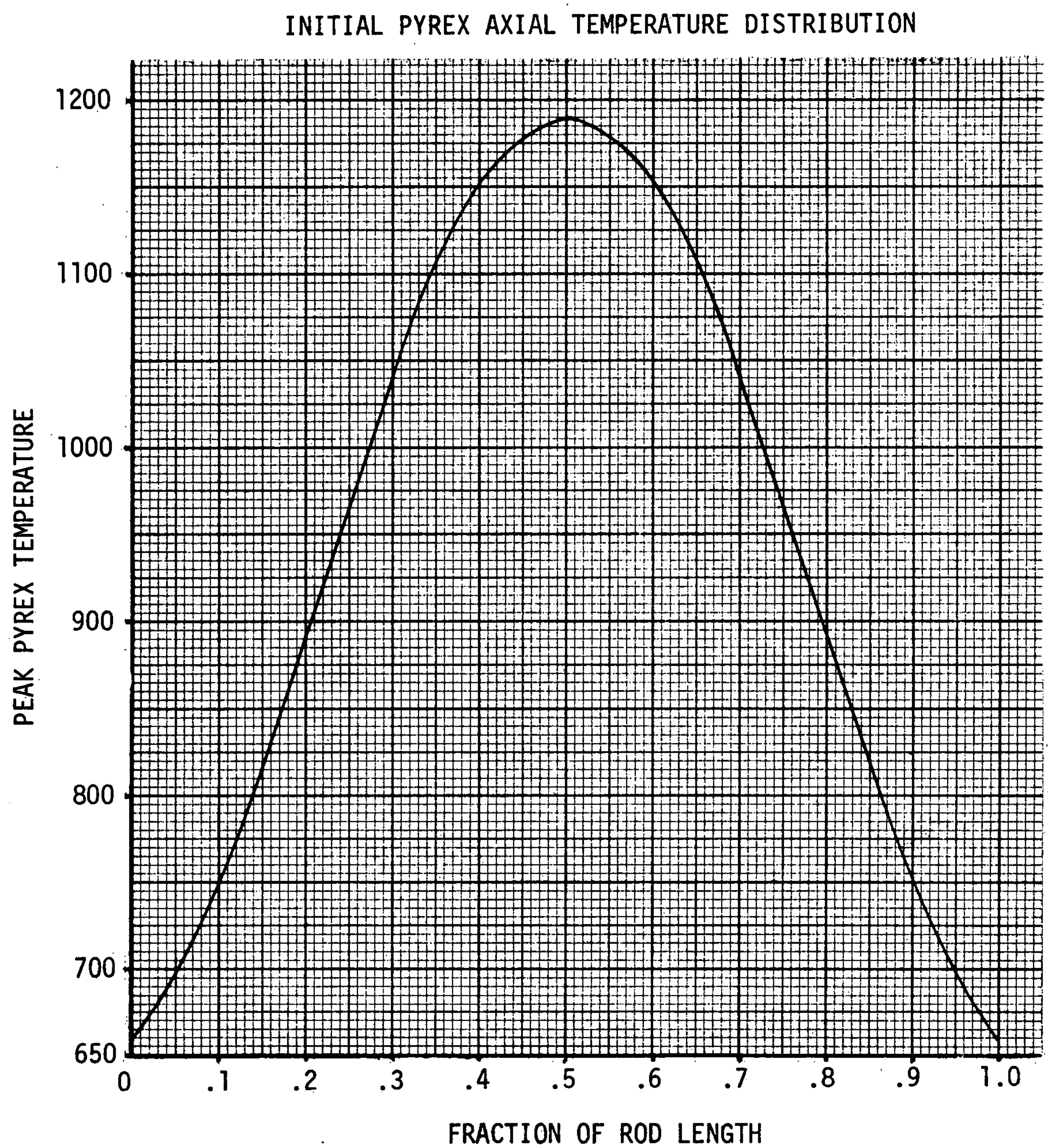

INITIAL PYREX AXIAL TEMPERATURE DISTRIBUTION
PEAK PYREX TEMPERATURE
1200
1100
1000
900
800
700
650
0
.1
.2
.3
.4
.5
.6
.7
.8
.9
1.0
FRACTION OF ROD LENGTH

FIGURE 3-10

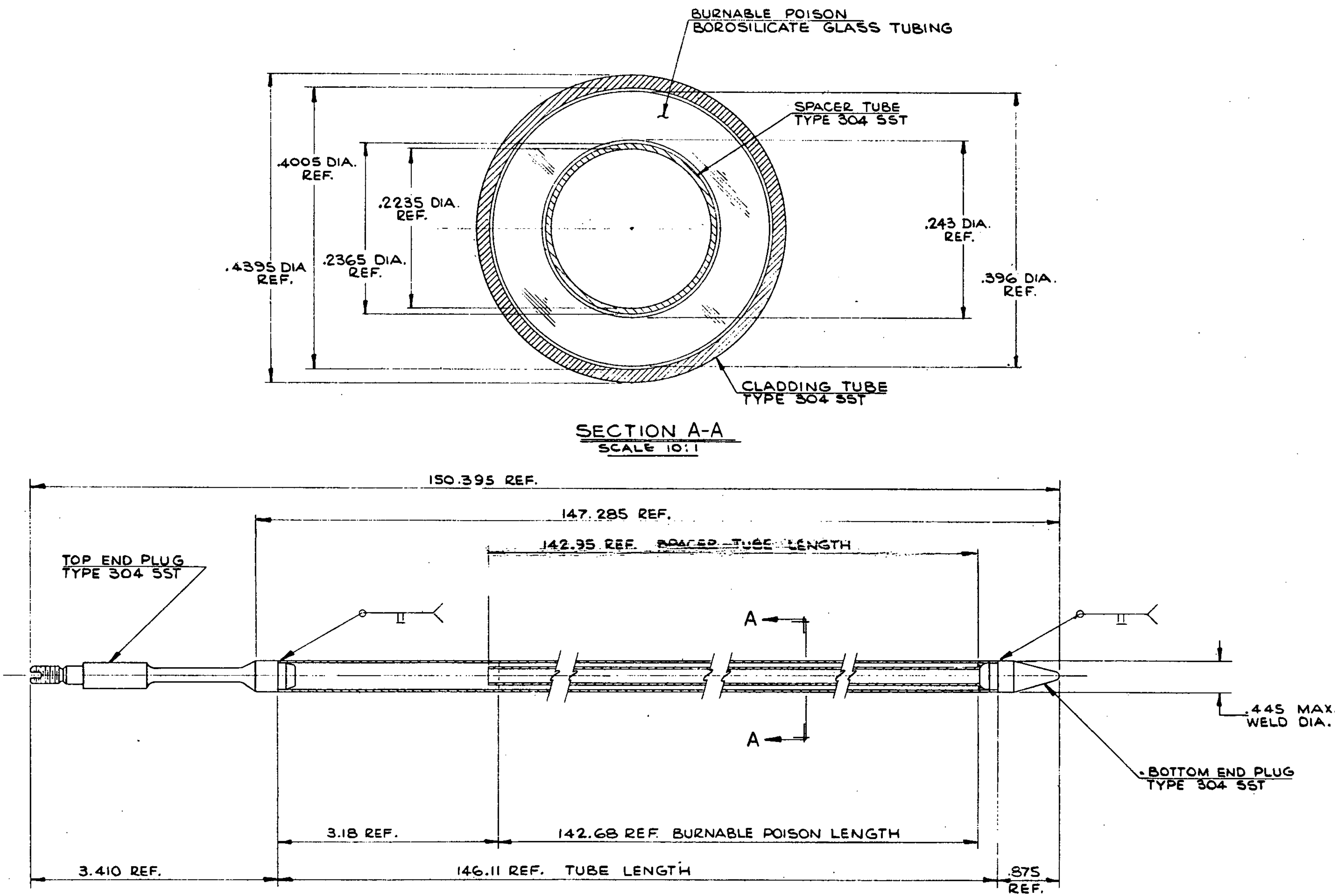

BURNABLE POISON
BOROSILICATE GLASS TUBING
SPACER TUBE
TYPE 304 SST
.4005 DIA.
REF.
.2235 DIA.
REF.
.2365 DIA.
REF.
.4395 DIA
REF.
.243 DIA.
REF.
.396 DIA.
REF.
CLADDING TUBE
TYPE 304 SST
SECTION A-A
SCALE 10:1
150.395 REF.
147.285 REF.
142.95 REF. SPACER TUBE LENGTH
TOP END PLUG
TYPE 304 SST
A
A
.445 MAX.
WELD DIA.
BOTTOM END PLUG
TYPE 304 SST
3.18 REF.
142.68 REF. BURNABLE POISON LENGTH
3.410 REF.
146.11 REF. TUBE LENGTH
.875
REF.

Figure 3-11

MEMORANDUM

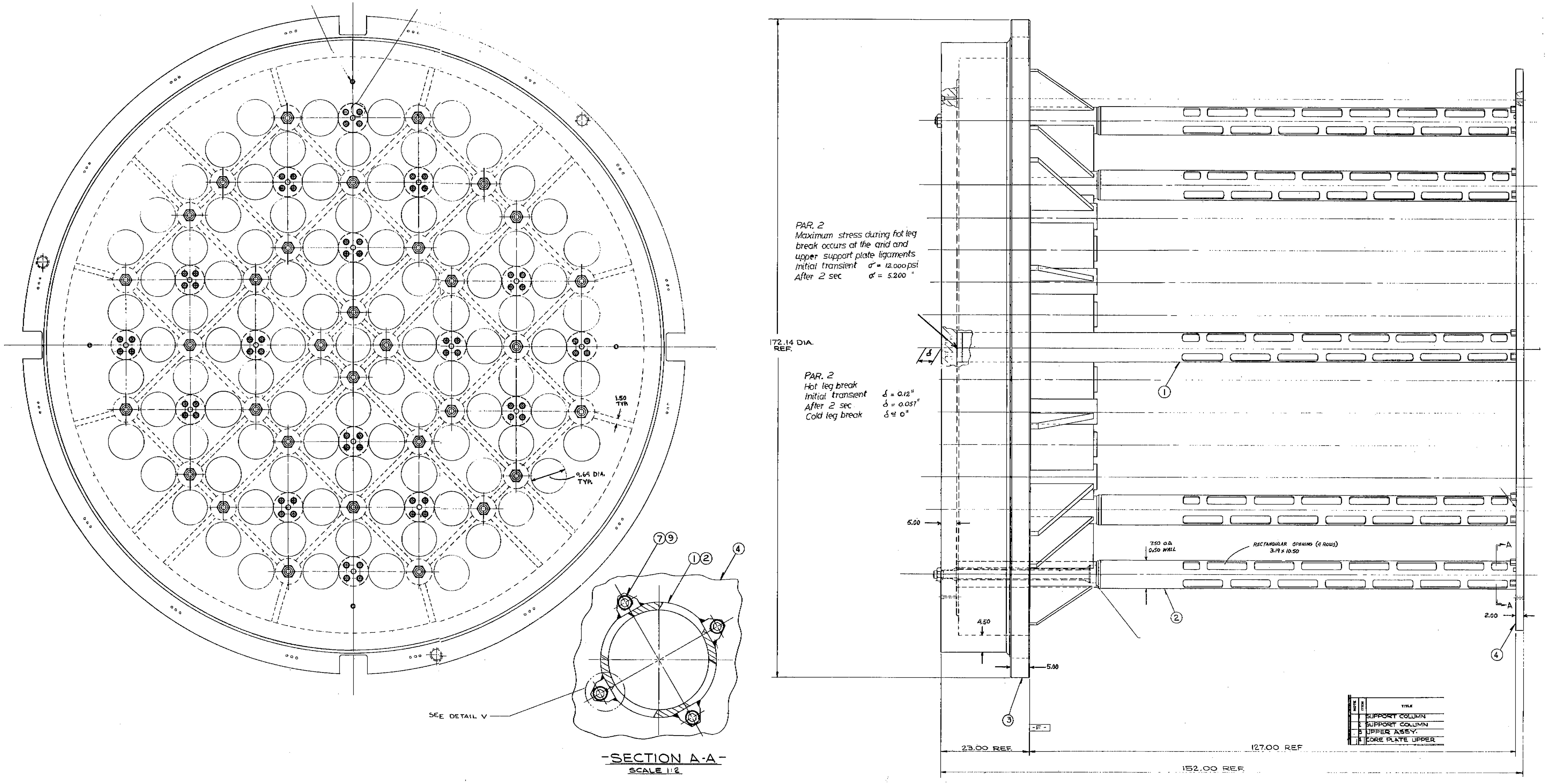
MEMORANDUM

431 77 501 7

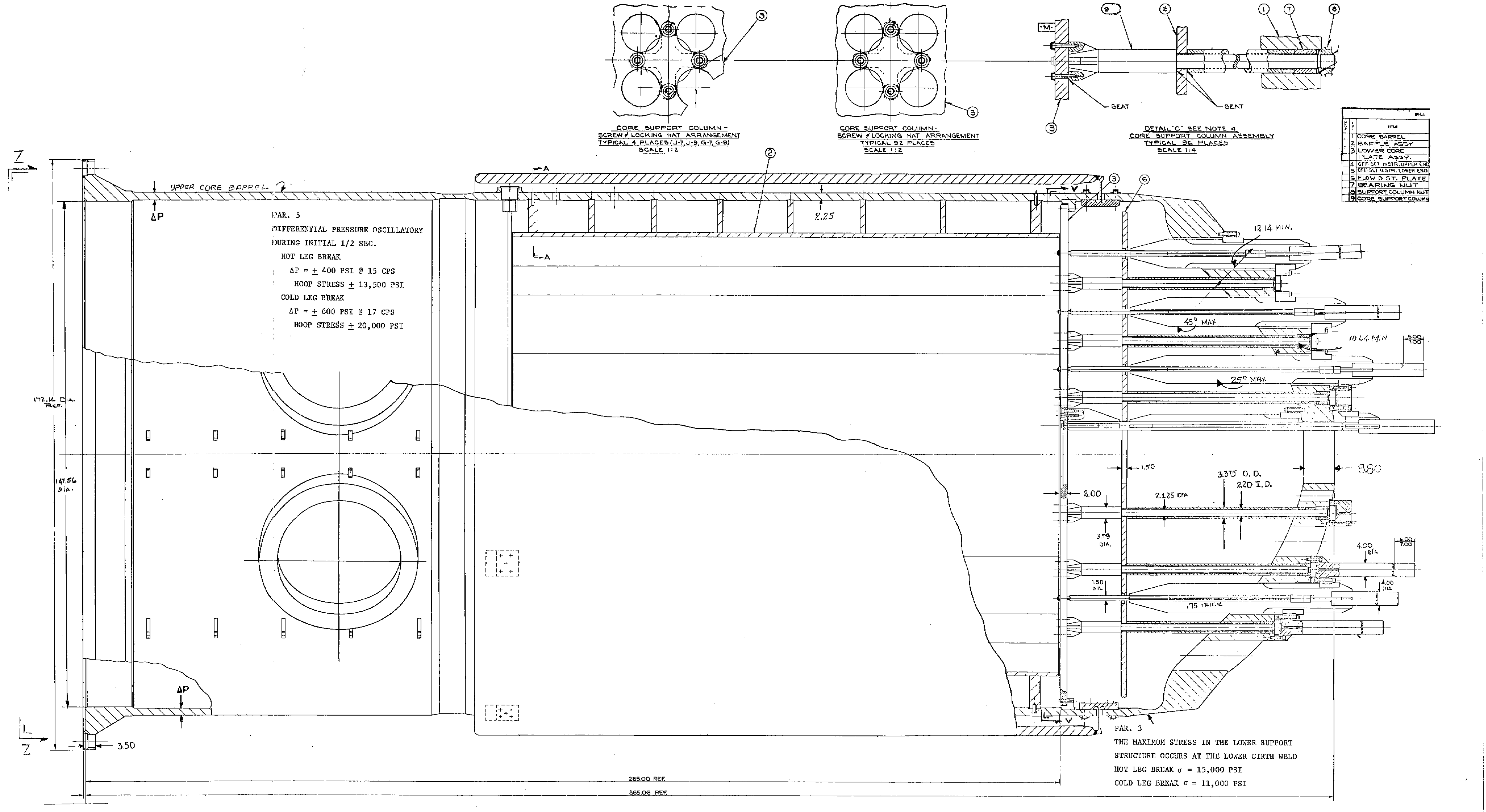
ITEM	TITLE
1	TOP NOZZLE ASSY.
2	SKELETON ASSY.
3	FUEL ROD ASSY.
4	ADAPTER PLATE
5	.2503/.2501 DIA. x .219 SST DOWEL PIN
6	CONTROL ROD ASSY.

TERA
APERTURE
CARD

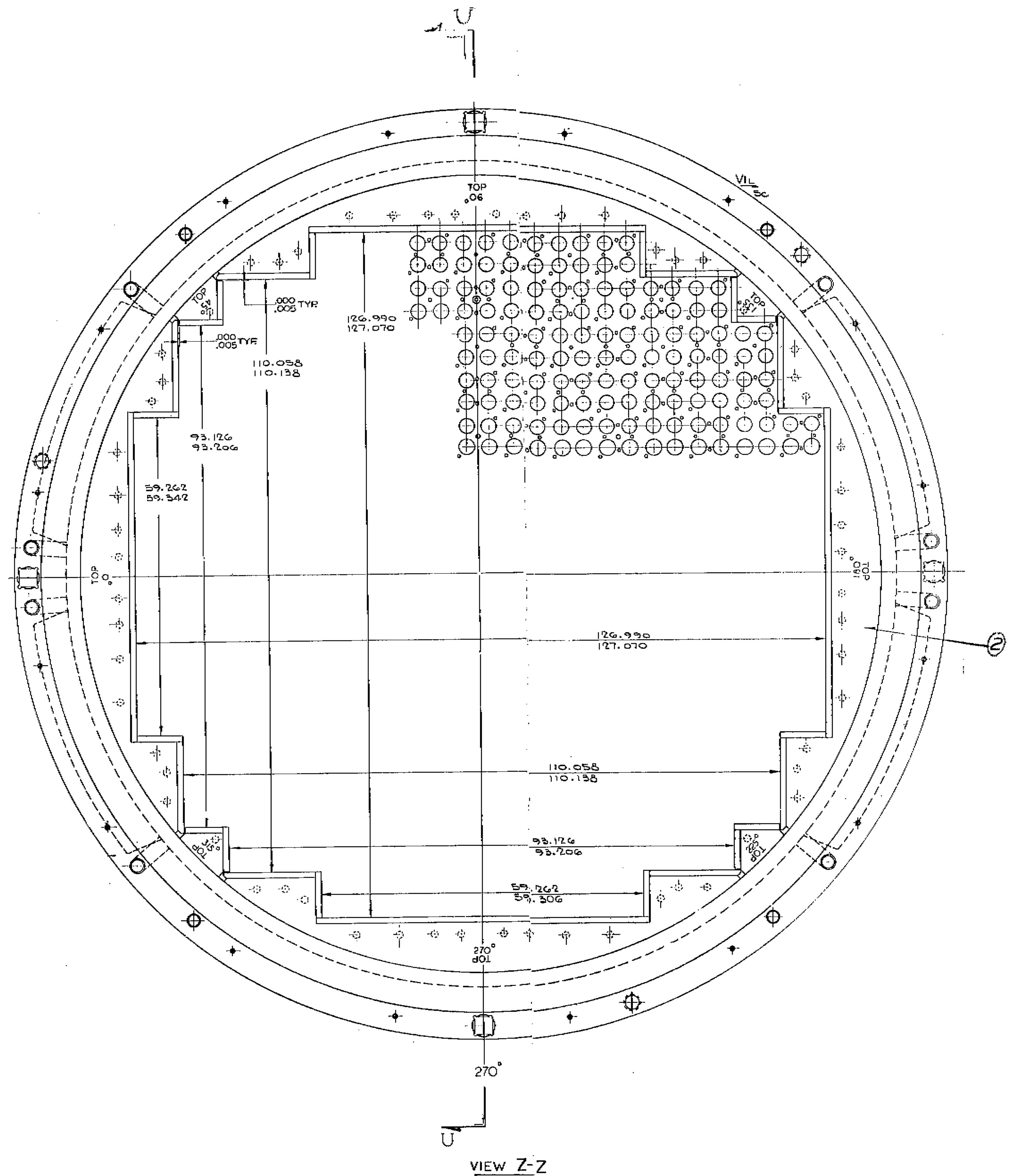
FUEL ASSEMBLY
FIG. 2-4



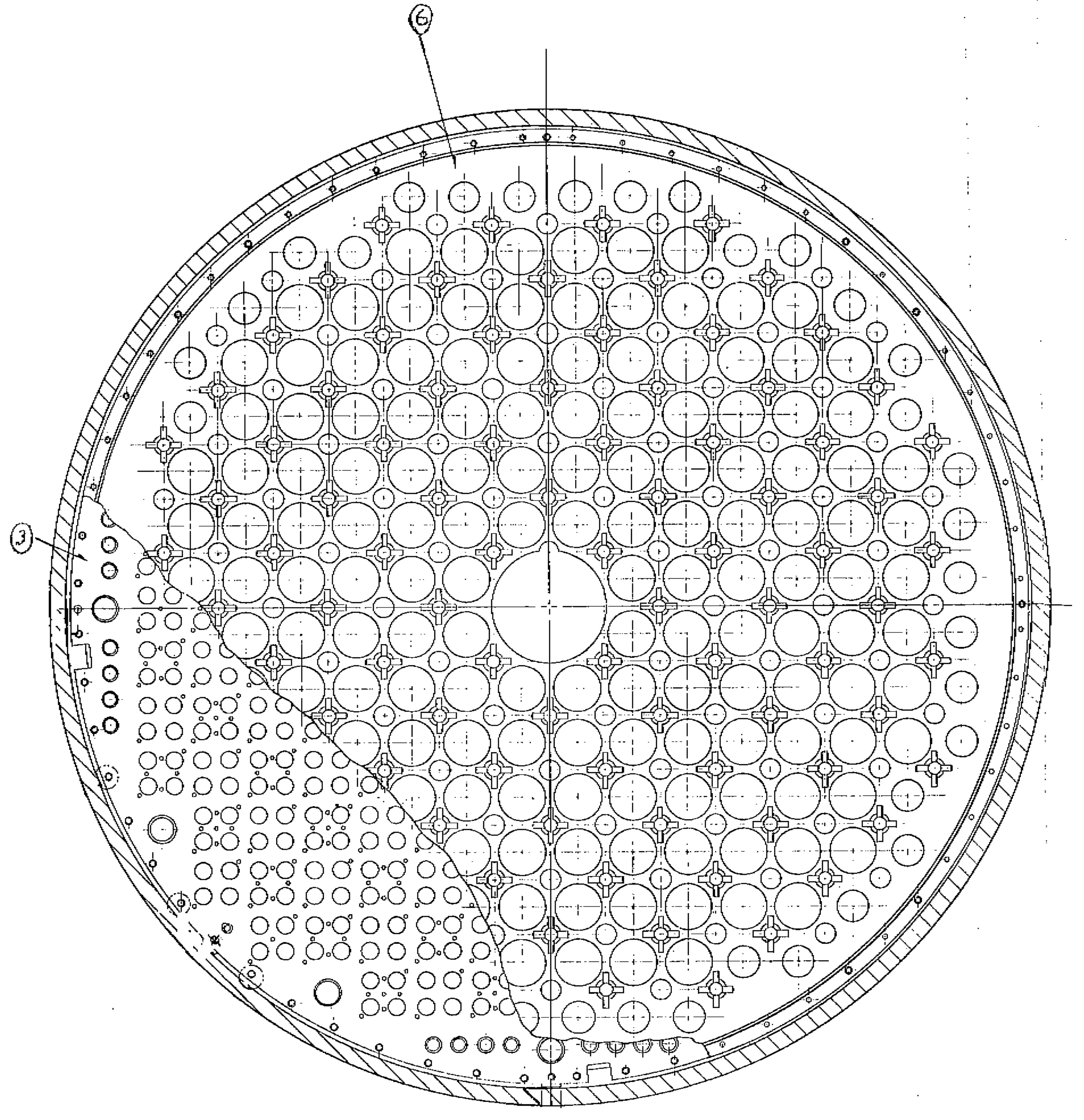
UPPER CORE SUPPORT STRUCTURE
FIG. 2-3



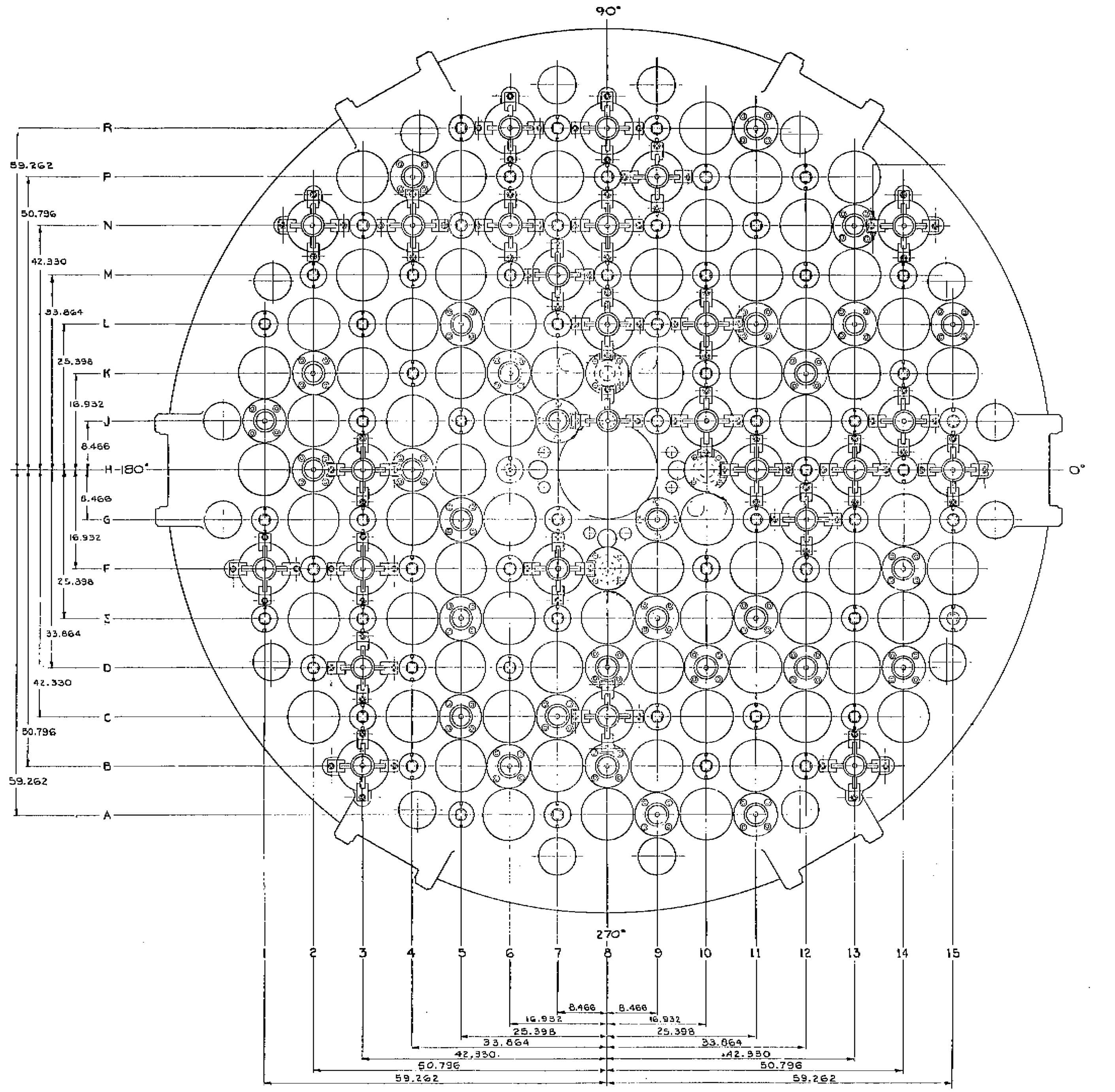
CORE BARREL ASSEMBLY, SHEET 1
FIG. 2-2



VIEW Z-Z



SECTION V-V



LOWER SUPPORT CASTING VIEW

CORE BARREL ASSEMBLY, SHEET 2
FIG. 2-2