Duke Power Company

OCONEE NUCLEAR STATION

UNITS 1, 2 AND 3

Final Safety Analysis Report

Volume 2

1763

LIST OF EFFECTIVE PAGES
TABLE OF CONTENTS
VOLUME 2

TABLE OF CONTENTS

TABLE OF CONTENTS - (Continued)

Rev. 9. 8/11/70

TABLE OF CONTENTS - (Continued

Rev. 10. 8/28/70
Rev. 18. 3/10/72

TABLE OF CONTENTS - (Continued)

APPENDIX 4A

APPENDIX 4B

APPENDIX 4C

Rev. 9. 8/11/70
Rev. 18. 3/10/72
Rev. 21. 7/26/72

TABLE OF CONTENTS - (Continued)

TABLE OF CONTENTS - (Continued)

TABLE OF CONTENTS - (Continued)

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 18 3/10/72
Rev. 19 5/5/72

TABLE OF CONTENTS - (Continued)

Rev. 9. 8/11/70
Rev. 18. 3/10/72
Rev. 21. 7/26/72

TABLE OF CONTENTS - (Continued)

TABLE OF CONTENTS - (Continued)

TABLE OF CONTENTS - (Continued)

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 18 3/10/72

TABLE OF CONTENTS - (Continued)

22. Replacement Pages for Units 1 and 2 Technical Specifications
Dockets 50-269, -270

Replacement Pages for Units 1, 2, and 3 Technical Specifications
Dockets 50-269, -270, -287

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 13. 9/17/70
Rev. 14. 12/14/70
Rev. 16. 7/30/71
Rev. 18. 3/10/72
Rev. 22. 8/25/72

FSAR SUPPLEMENTS (Located at rear of Volume 4)

LIST OF EFFECTIVE PAGES
FSAR SECTION 4

Reactor Coolant System

Reactor Coolant System

TABLE OF CONTENTS

CONTENTS (Cont'd)

CONTENTS (Cont'd)

Rev. 5. 5/25/70
Rev. 21. 7/26/72

CONTENTS (Cont'd)

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF FIGURES

(At Rear or Section)

Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 9. 8/11/70
Rev. 21. 7/26/72

Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protective system and engineered safety feature protective system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

30. 4.1.3 Using the incore instrumentation detector system, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten (10) effective full power days.

Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be calibrated (during steady state operating conditions) when indicated neutron power and core thermal power differ by more than 2 percent. During non-steady state operation, the nuclear flux channels amplifiers shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protective channels is required once every four weeks on a rotational or perfectly staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protective channels is as follows:

Channels A, B, C & D	Before startup if the reactor has been shutdown for greater than seven days
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protective system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every four weeks. The trip test checks all logic combinations and is to be performed on a rotational basis. Discovery of an unsafe failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the equipment and systems in a safe operational status.

Power Distribution Mapping

The incore instrumentation detector system will provide a means of assuring that axial and radial power peaks and the peak locations are being controlled by the provisions of the Technical Specifications within the limits employed in the safety analysis.

REFERENCE

FSAR, Section 7.1.2.3.4

designed to maintain their functional integrity during earthquake. Design is in accordance with the seismic design bases shown below. The loading combinations and corresponding design stress criteria for internals and pressure boundaries of vessels and piping are given in the section. A discussion of each of the cases of loading combinations follows:

4.1.2.5.1 Seismic Loads

<u>Case I - Design Loads Plus Design Earthquake Loads</u> - For this combination, the reactor must be capable of continued operation; therefore, all components excluding piping are designed to Section III of the ASME Code for Reactor Vessels. (1)
The primary piping is designed according to the requirements of USAS B31.1 and B31.7. The S_m values for all components, excluding bolting, are those specified in Table N-421 of the ASME code. The S_m value for bolts are those specified in Table N-422 of the ASME Code.

<u>Case II - Design Loads Plus Maximum Hypothetical Earthquake Loads</u> - In establishing stress levels for this case, a "no-loss-of-function" criterion applies, and higher stress values than in Case I can be allowed. The multiplying factor of (1.2) has been selected in order to increase the code-based stress limits and still insure that for the primary structural materials, i.e., 304 SST, 316 SST, SA302B, SA2102B, and SA106C, an acceptable margin of safety will always
3. exist. A more detailed discussion of the adequacy of these margins of safety is given in B&W Topical Report BAW-10008, Part 1, "Reactor Internals Stress & Deflection Due to LOCA and Maximum Hypothetical Earthquake". The S_m value for all components are those specified in Table N-421 of the ASME Code.

4.1.2.5.2 Loss-of-Coolant Loads

A loss-of-coolant accident coincident with a seismic disturbance has been analyzed to assure that no loss of function occurs. In this case, priamry attention is focused on the ability to initiate and maintain reactor shutdown and emergency core cooling. Two additional cases are considered as follows:

<u>Case III - Design Loads Plus Pipe Rupture Loads</u> - For this combination of loads, the stress limits for Case II are imposed for those components, systems, and equipment necessary for reactor shutdown and emergency core cooling.

<u>Case IV - Design Loads Plus Maximum Hypothetical Earthquake Loads Plus Pipe Rupture Loads</u> - Two thirds of the ultimate strength has been selected as the stress limit for the simultaneous occurrence of maximum hypothetical earthquake and reactor coolant pipe rupture. As in Case III, the primary concern is to maintain the ability to shut the reactor down and to cool the reactor core. This limit assures that a materials strength margin fo safety of 50 per cent will always exist.

The design allowable stress of Case IV loads is given in B&W Topical Report BAW-10008 for 304 stainless steel. This curve is used for all reactor vessel
1. internals including bolts. It is based on adjusting the ultimate strength curves published by U.S. Steel to minimum ultimate strength values by using the ratio of ultimate strength given by Table N-421 of Section III of the ASME code at room temperature to the room temperature strength given by U.S. Steel.

In Cases II, III, and IV, secondary stresses were neglected, since they are

self-limiting. Design stress limits in most cases are in the plastic region, and local yielding would occur. Thus, the conditions that caused the stresses
21. are assumed to have been satisfied. See B&W Topical Report BAW-10008, Part 1, for a more extensive discussion of the margin of safety, the effects of using elastic equations, and the use of limit design curves for reactor internals.

Stress Limits for
Seismic, Pipe Rupture and Combined Loads

Case	Loading Combination	Stress Limits
I	Design loads + design earthquake loads	$P_m \leq 1.0\ S_m$
		$P_L + P_b \leq 1.5\ S_m$
II	Design loads + maximum hypothetical earthquake loads	$P_m \leq 1.2\ S_m$
		$P_L + P_b \leq 1.2\ (1.5\ S_m) = 1.8\ S_m$
III	Design Loads + pipe rupture loads	$P_m \leq 1.2\ S_m$
		$P_L + P_b \leq 1.2\ (1.5\ S_m) = 1.8\ S_m$
IV	Design loads + maximum hypothetical earthquake loads + pipe rupture loads	$P_m \leq 2/3\ S_u$
	[illegible]	$P_L + P_b \leq 2/3\ S_u$

*where

P_L = Primary local membrane stress intensity

P_m = Primary general membrane stress intensity

P_b = Primary bending stress intensity

S_m = Allowable membrane stress intensity

S_u = Ultimate stress for unirradiated material at operating temperature

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*(1) All symbols have the same definition or connotation as those in ASME B&PV Code Section III, Nuclear Vessels.

(2) All components will be designed to insure against structural instabilities regardless of stress levels.

4.1.2.6 Service Lifetime

The design service lifetime for the major reactor coolant system components is 40 years. The number of cyclic system temperature and pressure changes (Table 4-8), is based on operation for this design lifetime.

4.1.2.7 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of the reactor coolant system surfaces. The design water quality is listed in Table 4-10. The reactor coolant chemistry is discussed in further detail in Section 9.2.

4.1.2.8 Vessel Radiation Exposure

The reactor vessel is the only reactor coolant system component exposed to a significant level of neutron irradiation and is therefore the only component subject to material radiation damage. The maximum exposure from fast neutrons ($E > 1.0$ MeV) has been computed to be less than 3.0×10^{19} neut/cm^2 over a 40-year life with an 80 per cent load factor. Reactor vessel irradiation calculations are described in 4.3.3.

4.1.3 CODES AND CLASSIFICATIONS

The codes listed in this section include the code addenda and case interpretations issued through Summer 1967 (June 30, 1967) unless noted otherwise. Quality control and quality assurance programs relating to the fabrication and erection of system components are summarized in 4.3.11.

4.1.3.1 Vessels

The design, fabrication, inspection and testing of the reactor vessel and closure head, steam generator (both reactor coolant side and secondary side) and pressurizer is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels.

4.1.3.2 Piping

The design, fabrication, inspection and testing of the reactor coolant piping including the pressurizer surge line and spray line is in accordance with USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, dated February, 1968, and as corrected for Errata under date of June, 1968. The feedwater header and the auxiliary feedwater header for the steam generator meet the requirements of the Code for Pressure Piping, Power Piping USAS B31.1.0 - 1967.

16. #### 4.1.3.3 Reactor Coolant Pumps (Reference Supplement 6 Revisions for Oconee 1)

The reactor coolant pump casings are designed, fabricated, inspected and tested to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels, but are not code stamped.

The pump is designed such that the pressure boundary casing is similar in shape to a simple pressure vessel. Internal to the casing is a separate diffuser section which provides the correct flow passages to develop the hydraulic characteristics of the pump. This design will permit the initiation of the analysis by separating the vessel into shell elements of simple geometry (such as rings, cylinders, etc.) of which the structural behavior is known. The pressure, mechanical and thermal loads acting on the structure are applied to the shell elements with a system of forces required to maintain the static equilibrium of each element.

The following computer programs have been utilized by Mechanics Research Institute in performing the code calculations on this casing for the pump vendor:

a. SEAL SHELL 2

SEAL SHELL 2 is a digital computer program prepared by Westinghouse Electric Corporation for the U.S. Atomic Energy Commission. The program determines the stresses, loads, and deformations for a shell of revolution with applied axisymmetric end loads (axial, radial, and moment), pressure (internal or external), and thermal gradients.

b. MAX STRESS

MAX STRESS is a digital computer program prepared by Mechanics Research Institute to maximize the stresses in the nozzle due to pressure, external bending and axial loads. The first phase of the program calculates the torsional moment, bending moment, bending moment angle, axial load, normal shear load, and normal shear angle for each possible combination of loads.

Phase 2 of the program calculates the maximum principal stress and twice the maximum shear stress at the inside fiber, midwall fiber, and outside fiber for each of the possible combinations of loadings. The maximums are then printed along with the maximum axial stress, maximum hoop stress (VQ/It), and maximum midwall axial stress.

The program also has the option of maximizing the stress at any section around the circumference of the nozzle. The stress was generally maximized within $\pm$ 10° of the section analyzed.

c. THAN

THAN or THERMAL ANALYZER solves three-dimensional transient heat flow problems, producing the temperature history of a physical system of any arbitrary geometry. This is accomplished through the concept of lumped parameters, and the problem is expressed as an electrical analog of the heat transfer problem. The program utilizes finite difference techniques for problem solution. Steady-state problems are also solved by the program and depend only on the boundary conditions and lumped parameter characteristics. From these data supplied by the THAN program, temperature time histories (in the case of the bowl) are determined.

An analysis in accordance with paragraph N-415.1 of the ASME Code was performed to determine if the pump casing required a fatigue analysis for the number of design cycles specified. This analysis showed that the pump casing bowl met all the requirements of paragraph N-415.1. Thus a fatigue analysis was not required. However, a fatigue analysis was performed on the pump casing cover in which the worst possible stress combination was considered at the two most critical points in the cover. It was found from this analysis, with this very conservative approach, that the maximum cumulative usage factor is only 0.125 for the design cycles specified for this plant.

A summary of the code allowables is listed in Table 4-20 and shown pictorally on Figures 4-7a and 4-7b. The reinforcement area is as defined in paragraph N-454 of the ASME Code Section III. The stress analysis performed on the bowl and the attached nozzles showed that the stresses are within the allowable limits. Note that a factor of two was applied to the nozzle loading due to seismic reactions and when these were combined with the dead weight and thermal expansion reactions, the stress levels fell within a realistic allowable of the stress intensities shown in Table 4-20. A summary of maximum calculated stresses is given in Table 4-21.

The casing cover analysis indicates that the thermal stresses and pressure stresses on the cover are within the Section III code allowables.

There are no deviations from the applicable ASME Code requirements in the design and fabrication of the pump casings other than code stamping.

4.1.3.4 Relief Valves

The pressurizer code safety valves and the electromatic relief valve comply with Article 9, Section III, of the ASME Boiler and Pressure Vessel Code.

4.1.3.5 Attachments to Loop

Nozzles on the reactor coolant piping comply with Paragraph 4.1.3.2 above, and nozzles on the vessels comply with Paragraph 4.1.3.1 above.

4.1.3.6 Welding

Welding qualifications are in accordance with the ASME Boiler and Pressure Vessel Code, Section III and Section IX, as applicable.

4.2 SYSTEM DESCRIPTION AND OPERATION

4.2.1 GENERAL

4.2.1.1 System

The reactor coolant system consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer and interconnecting piping. The system is arranged in two heat transport loops, each with two reactor coolant pumps and one steam generator. The reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes transferring heat to the steam and water on the shell side of the steam generator. In each loop, the reactor coolant is returned to the reactor through two lines, each containing a reactor coolant pump, to the reactor vessel. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector, and a solvent for the soluble poison (boron in the form of boric acid). The reactor coolant system schematic is shown in Figure 4-1.

4.2.1.2 System Protection

a. Missiles

Engineered safety features and associated systems are protected from missiles which might result from a loss of coolant accident. Protection is provided by concrete shielding and/or segregation of redundant components.

The reactor vessel is surrounded by a concrete primary shield wall and the heat transport loops are surrounded by a concrete secondary shield wall. These shielding walls provide missile protection for the reactor building liner plate and equipment located outside the secondary shielding.

Removable concrete slabs over the reactor vessel area and the concrete deck over the area outside of the secondary shield wall also provide shielding and missile protection.

b. Seismic

The reactor coolant system is analyzed for maximum hypothetical earthquake to determine that resultant stresses do not jeopardize the safe shutdown of the reactor coolant system and removal of decay heat.

4.2.1.3 System Arrangement

The arrangement of the reactor coolant system is shown in Figures 4-2 and 4-3. Figures in Section 1 depict the system arrangement in relation to shielding walls, the reactor building and other equipment in the building.

4.2.2 MAJOR COMPONENTS

4.2.2.1 Reactor Vessel

The reactor vessel consists of a cylindrical shell, a cylindrical support skirt, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. The reactor closure head is a spherically dished head welded to a matching ring flange. The reactor vessel general arrangement is shown in Figure 4-4. The general arrangement of the reactor vessel with internals is shown in Figures 3-46 and 3-47. Reactor vessel design data is listed in Table 4-3.

The number and size of reactor vessel nozzles are also shown in Table 4-3. All coolant inlet, coolant outlet, core flooding, and control rod drive nozzles are located above the level of the top of the core. The reactor vessel is vented through the control rod drives.

All major reactor vessel nozzles are installed with full penetration welds. All control rod drive and incore instrument nozzles are installed with partial penetration welds. The gasket leakage tap is installed in each reactor vessel
4. flange with a partial penetration weld. In addition, the Oconee #1 closure head contains eight (8) instrumentation nozzles installed by partial penetration welds outside of the region of the control rod drive nozzles.

The reactor closure head flange and the reactor vessel flange are joined by sixty 6-1/2 in. diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. Test taps are provided in the annulus between the two O-rings to afford a means to leak test the vessel closure seal after refueling. To insure uniform loading of the closure seal, the studs are hydraulically tensioned.

The reactor vessels and closure heads are constructed of a combination of formed plates and forgings. The rin forging in the reactor vessel shells, other than
4. closure flanges, for Unit #1, #2, and #3 are identified in Figures 4-4 (Oconee I), 4-4a (Oconee II) and 4-4b (Oconee III).

The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control rod assemblies, axial power shaping rod assemblies, surveillance specimens and holder tubes, and incore instrumentation. Guide lugs welded to the inside of the reactor vessel wall limit reactor internals and core to a vertical drop of one-half in. or less, and prevent rotation of the core and internals about the vertical axis in the unlikely event of a major core barrel or core support shielf failure. The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods throughout their full stroke. The internals and the core are supported from the reactor vessel flange. The control rod drive mechanisms are supported by the nozzles in the reactor vessel head.

Surveillance specimens, made from appropriately selected specimens of reactor vessel steel, are located between the reactor vessel wall and the thermal shields. These specimens are located so as to afford the desired fast neutron exposure lead time with respect to the vessel wall, and will be examined at appropriate intervals to evaluate reactor vessel material NDTT changes.

4.2.2.2 Steam Generator

16. The steam generator general arrangement is shown in Figure 4-5. Principal design data are tabulated in Table 4-4. (Refer to Supplement 9 Revisions for Oconee 3)

The once-through steam generator supplies superheated steam and provides a barrier to prevent fission products and activated corrosion products from entering the steam system.

The steam generator is a vertical, straight tube, tube and shell heat exchanger which produces superheated steam at constant pressure over the power range. Reactor coolant flows downward through the tubes and transfers heat to generate steam on the shell side. The high pressure (reactor coolant pressure) parts of the unit are the hemispherical heads, the tube sheets and the tubes between the tube sheets. Tube support plates maintain the tubes in a uniform pattern along their length. The unit is supported by a skirt attached to the bottom head.

The shell, the outside of the tubes, and the tube sheets form the boundaries of the steam producing section of the vessel. Within the shell, the tube bundle is surrounded by a cylindrical baffle. There are openings in the baffle at the feedwater inlet nozzle elevation to provide a path for steam to afford contact feedwater heating. The upper part of the annulus formed by the baffle plate and the shell is the superheat steam outlet, while the lower part is the feedwater inlet heating zone.

Vent, drain, and instrumentation nozzles and inspection handholes are provided on the shell side of the unit. The reactor coolant side has manway openings in both the top and bottom heads, and a drain nozzle on the bottom head. Venting of the reactor coolant side of the unit is accomplished by a vent connection on the reactor coolant inlet pipe to each unit.

Feedwater is supplied to the steam generator through an auxiliary feedwater ring located at the top of the steam generator to assure natural circulation of the reactor coolant following the unlikely event of the loss of all reactor coolant circulating pumps.

Four heat transfer regions exist in the steam generator as feedwater is converted to superheated steam. Starting with the feedwater inlet these are:

a. Feedwater Heating

Feedwater is heated to saturation temperature by direct contact heat exchange. The feedwater entering the unit is sprayed into the downcomer annulus formed by the shell and the cylindrical baffle around the tube bundle. Steam is drawn by aspiration into the downcomer and heats the feedwater to saturation temperature.

The saturated water level in the downcomer provides a static head to balance the static head in the nucleate boiling section, and the required head to overcome pressure drop in the circuit formed by the downcomer, the boiling sections and the bypass steam flow to the feedwater heating region. The downcomer water level varies with steam flow from 15 - 100% load. A constant minimum level is held below 15% load.

b. Nucleate Boiling

The saturated water enters the tube bundle just above the lower tube sheet and the steam-water mixture flows upward on the outside of the tubes countercurrent to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until DNB is reached, and then film boiling and superheating occurs.

c. Film Boiling

Dry saturates steam is produced in the film boiling region of the tube bundle.

d. Superheated Steam

Saturated steam is raised to final temperature in the superheater region. The amount of surface available for superheat varies inversely with load. As load decreases the superheat section gains surface from the nucleate and film boiling regions. Mass inventory in the steam generator increases with load as the length of the heat transfer regions vary. Changes in temperature, pressure and load conditions cause an adjustment in the length of the individual heat transfer regions and result in a change in the inventory requirements. If the inventory is greater than that required, the pressure increases. Inventory is controlled automatically as a function of load by the feedwater controls in the integrated control system.

Steam generator feedwater quality requirements are shown in Table 4-11.

Electroslag welding is utilized on longitudinal seams of the 7-inch shell courses of the steam generator as shown in Figure 4-5a. The techniques used in the electroslag welding for the Oconee steam generators are identical to those used in the electroslag welding program reported as Appendix F of Dockets No. 50-237 and 50-249 (Dresden Units 2 and 3). The procedures used were appropriately modified to reflect the difference in materials of the components being welded.

Each weld is subjected to radiographic inspection, ultrasonic inspection and the finished surfaces of the weld is magnafluxed. In addition, each plate is ordered with excess width so that test specimens may be removed after
5. heat treatment. Physical property test specimens including tensile and impact specimens of the base material heat affected zone and weld metal is obtained from this excess material in accordance with Section III of the ASME Code. Radiographic, ultrasonic, and magnetic particle inspection is performed in accordance with Section III of the ASME Code and as required by Code Case 1355 which permits such welds for Class A vessels.

Physical tests are performed per Section N-511 of Section III of the ASME Code. For example:

a. All weld metal tensile specimens from each heat of weld wire, batch of flux and for each combination of heat of wire and batch of flux used is obtained and tested after heat treatment.

b. Charpy impact test specimens representing weld metal and heat affected base material for every heat of wire, batch of flux and combination of heat of wire and batch of flux used is tested.

c. Charpy V-notch impact specimens and tensile specimens are tested for 15 percent of all production welds. Included in this 15 percent are the tests required by (a) and (b) above.

All electroslag welds are made in the vertical position. Two men, one on
5. the inside and one on the outside of the vessel, are used to check the
progress of the weld and to insure that the prescribed welding procedure is
being followed. The weld is started in a U-shaped starting fixture about
six inches deep attached to the bottom of the joint. The weld stabilizes in
this starting tab which is later cut off and discarded. The weld once started
is not stopped until the total seam is completed.

The weld receives a heat treatment which consists of a water quench from 1625°F and a temper of 1150°F followed by an air cool. This post-weld heat treatment refines the grain of the weld and the base material heat affected zone such that it is virtually indistinguishable from the unaffected base material. The microstructure is the same through the weld.

4.2.2.3 Pressurizer

The pressurizer general arrangement is shown in Figure 4-6 and principal design data are tabulated in Table 4-5.

The electrically heated pressurizer establishes and maintains the reactor coolant system pressure within prescribed limits, and provides a steam surge chamber and a water reserve to accommodate reactor coolant density changes during operation.

The pressurizer is a vertical cylindrical vessel with a bottom surge line penetration connected to the reactor coolant piping at the reactor outlet. The pressurizer contains removable electric heaters in its lower section and a water spray nozzle in its upper section to maintain reactor coolant system pressure within desired limits. The pressurizer vessel is protected from thermal effects by a thermal sleeve in the surge line and by an internal diffuser located above the surge pipe entrance to the vessel.

During outsurges, as reactor coolant system pressure decreases, some of the pressurizer water flashes to steam, thus assisting in maintaining the existing pressure. Heaters are then actuated to restore the normal operating pressure. During insurges, as system pressure increases, water from the reactor vessel inlet piping is sprayed into the steam space to condense steam and reduce pressure. Spray flow and heaters are controlled by the pressure controller. The pressurizer water level is controlled by the level controller.

Since all sources of heat in the system; core, pressurizer heaters, and reactor coolant pumps, are interconnected by the reactor coolant piping with no intervening isolation valves, relief protection is provided on the pressurizer. Overpressure protection consists of two code safety valves and one electromatic relief valve.

To eliminate abnormal buildup or dilution of boric acid within the pressurizer, and to minimize cooldown of the coolant in the spray and surge lines, a bypass flow is provided around the pressurizer spray control valve. This continuously circulates approximately 1 gpm of reactor coolant from the heat transport loop. A sampling connection to the liquid volume of the pressurizer is provided for auditing boric acid concentration. A steam space sampling line provides capability for monitoring of or venting accumulated gases.

During cooldown and after the decay heat system is placed in service, the pressurizer can be cooled by circulating water through a connection from the discharge of the low pressure injection pump to the pressurizer spray line.

Electroslag welding is utilized in the fabrication of the pressurizer, only in the longitudinal seams of the shell courses. A total of three individual
5. electroslag welds are made in the fabrication of each pressurizer. The electroslag welding process and quality control is the same as described in paragraph 4.2.2.2.

16. 4.2.2.4 Reactor Coolant Piping (Reference Supplement 6 Revisions for Oconee 1)

The general arrangement of the reactor coolant piping is shown in Figures 4-2 and 4-3. Principal design data are tabulated in Table 4-6.

The major piping components in this system are the 28-inch ID cold leg piping from the steam generator to the reactor vessel and the 36-inch ID hot leg piping from the reactor vessel to the steam generator. Also included in this system are the 10-inch surge line and the 2-1/2-inch spray line to the pressurizer. The system piping also incorporates the auxiliary system connections necessary for operation. In addition to drains, vents, pressure taps, injection and temperature element connections, there is a flow meter section in each 36-inch line to the steam generators to provide a means of determining the flow in each loop.

9. The 28-inch and 36-inch piping is carbon steel clad with austenitic stainless steel. Short sections of 28-inch stainless steel transition piping are provided between the pump casing and the 28-inch carbon steel lines. (Reference Supplement 6 revisions for Oconee 1.) Stainless steel or Inconel safe-ends are provided for field welding the nozzle connections to smaller piping. The piping safe-ends are designed so that there will not be any furnace sensitized stainless steel in the pressure boundary material. This is accomplished either by installing stainless steel safe-ends after stress relief or using Inconel. Smaller piping, including the pressurizer surge and spray lines, is austenitic stainless steel. All piping connections in the reactor coolant system are butt-welded except for the flanged connections on the pressurizer for the relief valves.

Thermal sleeves are installed where required to limit the thermal stresses developed because of rapid changes in fluid temperatures. They are provided in the following nozzles: the four high pressure injection nozzles on the reactor inlet pipes; the two core flooding low pressure injection nozzles on the reactor vessel; and the surge line nozzle and spray line nozzle on the pressurizer.

Rev. 5. 5/25/70

(Carryover)

4.2.2.5 Reactor Coolant Pumps

(Reference Supplement 6 Revisions for Oconee 1)

The reactor coolant pumps are single suction, single stage, vertical, radially balanced, constant speed centrifugal pumps. This type of pump employs mechanical seals to prevent reactor coolant fluid leakage to the atmosphere. A view of the pump is shown in Figure 4-7 and the principal design parameters are listed in Table 4-7. The reactor coolant pump performance characteristics are shown in Figure 4-8.

The pump casing design utilizes a quad-volute inner case permanently welded to a pressure containing outer case. The configuration of the pressure containing outer case is kept simple so that the casing quality will meet the required radiographic level and the stresses can be analyzed to meet the requirements of the design specification. The quad-volute inner casing consists of four volute passages spaced 90° apart which receive the discharge from the pump impeller and guide it efficiently into the outer casing where it flows to the discharge nozzle through a passage having a constantly increasing cross-sectional area. The pump casing is welded into the piping system and the pump internals can be removed for inspection or maintenance without removing the casing from the piping.

The pump cover and stuffing box is a unit containing a thermal barrier, recirculation impellers, shaft, journal bearing, and mechanical face-type seals. The pump shaft is coupled to the motor with a spacer coupling which
16. will permit removal and replacement of the seals without removing the motor. The pump cover has a cooling jacket to remove the heat which passes through the thermal barrier. This jacket has a capacity large enough to remove all heat which is transmitted to the cover. However, additional cooling capacity is provided, in case injection cooling water is lost. A recirculation impeller on the shaft immediately above the journal bearing circulates water in the bearing chamber to a heat exchanger and returns it to the chamber. The pump may be operated with loss of either injection water or cooling water.

The shaft seal system consists of face-type mechanical seals operating in tandem. Injection water, at a pressure above the pump suction pressure, is injected into the pump bearing chamber. A small portion of the injection water flows into the pump through a restriction bushing. The major portion flows through cooling slots in the O.D. of the bearing steel. The shaft seal system is made up of two mechanical seals operating in tandem, wherein about one-half of the system pressure is expanded in each seal. Each seal is capable of operation at the full system pressure. The fluid which leaks past the face-type mechanical seals passes into a seal leakage chamber and then out to the quench tank. A low pressure mechanical seal at the top of the seal leakage chamber prevents the escape of fluid to the atmosphere.

Electroslag welding is used to make the seven-inch thick circumferential butt weld which welds together the upper and lower halves of the pump casing. This weld is performed in accordance with ASME Code Case 1355-2 which permits electroslag welding of Class A pressure vessels. The casings are cast and welded by ESCO, who is the leading supplier of RC pump casings for the industry.

Electroslag welding is a welding process wherein coalescence is produced by heat generated in a conductive molten slag which melts the filler metal and the surfaces of the work to be welded. The weld pool is shielded by this slag and moves along the full cross section of the joint as the welding progresses. The conductive slag is maintained molten by its resistance to the flow of electric current passing between the electrode and the work. Water-cooled, non-fusing metal shoes are used to contain the molten metal on both sides of the weld. The welding is performed in a vertical position with the start and finish performed on run-out tabs affixed to the casting. These run-out tabs are later cut off and discarded. The only variables contained in the method of welding are the wide range of amperage (480-720) and voltage (44-52) needed
5. to control the molten pool of metal.

The weld is examined 100 percent using liquid penetrant and radiographic examination methods in accordance with Section III of the ASME Code. Ultrasonic inspection is not performed because the pump casing material, austenitic stainless steel, precludes achieving meaningful inspection results.

The pump casing receives two heat treatment cycles. The first is a solution annealing treatment where the pump casing halves are furnace heated to 1900 F,
6. held for a specified time, and water quenched. The second heat treatment is a stabilizing treatment in which the welded pump casing is heated to 725 F and air cooled.

4.2.2.6 Reactor Coolant Pump Motors

The reactor coolant pump motors are large, vertical, squirrel cage, induction machines. The motors have flywheels to increase the rotational-inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost. The flywheel is mounted on the upper end of the rotor, below the upper radial bearing and inside the motor frame. An anti-reverse device is included in the flywheel assembly to eliminate reverse rotation when there is back flow. Prevention of back rotation also reduces motor starting time.

The motors are enclosed with water-to-air heat exchangers so as to provide a closed circuit air flow through the motor. Radial bearings are floating pad type, and the thrust bearing is a double-acting Kingsbury type designed to carry the full thrust of the pump. A high pressure oil system with separate pumps is provided with each motor to jack and float the rotating assembly before starting. Once started, the motor provides its own oil circulation.

Instrumentation is provided to monitor motor cooling, bearing temperature, winding temperature, winding differential current, and speed.

In evaluating the design of the reactor coolant pump motor as it relates to the safety of the reactor coolant system, many items have been considered, namely: the overspeed of the motor; flywheel and shaft integrity; bearing design and system monitoring; seismic effects; and quality control and documentation.

An analysis of these considerations are given as follows as an indication of the safety and reliability that is integral with the motors:

a. Overspeed Considerations

The reactor coolant pump motors normally receive their electrical power from the nuclear generating unit through the unit's auxiliary electric system. On load rejection, the generating unit is designed to separate from the transmission network and remain in a standby operating condition carrying its own auxiliaries.

4. Figure 4-12 shows the turbine speed response following load rejection with the steam control valves wide open (VWO). On load rejection with VWO, the speed of the turbine-generator will increase under the control of the normal speed governing control system. The maximum speed attainable under the normal speed governing control system is less than 106% with the unit auxiliaries connected. This governing system is comprised of three independent control activities, namely: the speed control unit, power unbalance relay and the fast acting intercept valves all of which function to limit overspeed to below 106%.

As indicated in Figure 4.12 there are additional safety devices backing up the speed governing system, namely:

1. Mechanical overspeed trip which operates at 110% turbine-generator speed.
2. Generator overfrequency relay trip which is an electrical trip that operates at 111% turbine-generator speed.
3. Electrical back-up overspeed relay trip which operates at 112% turbine-generator speed.

In addition, each individual reactor coolant pump motor control circuit includes an overfrequency relay which trips the motor at 115% motor (or turbine-generator) speed. Therefore, it is evident that the reactor coolant pump motors' speed will be limited to less than 115%.

b. Flywheel Design Consideration:

For conservatism, the design of the flywheel on the reactor coolant pump motor is based on a design speed of 125%. The primary stress at the flywheel bore radius, with a speed of 125%, is 20,000 psi which is less than 50% of the 50,000 psi minimum yield strength of the flywheel material.
9. This, therefore, yields a centrifugal stress design safety margin of 250% at 125% speed.

The Duke Power Company specification on the motor calls for 500 motor starts in forty years; the flywheels have been designed for 10,000 starts yielding a safety factor of 20. However, calculation based on the material used in the flywheel results in 400,000 cycles required for crack initiation which results in a flywheel fatigue design safety factor of 800.

c. Flywheel Material, Fabrication, Test and Inspection

1. Material - The flywheel is manufactured from vacuum degassed ASTM 533 steel.
2. Fabrication and Test -
   A. Flywheel blanks are flame cut from a plate with enough surplus material to allow for the removal of the flame affected metal.
   B. At least three charpy tests are made on each plate parallel and normal to the rolling direction to determine that the blank meets specifications.
   C. A complete 100% volumetric ultrasonic test is made on the blank and tension and bend tests are also made prior to shipment of a blank to Westinghouse Electric Company.
   D. Following the machining of the flywheel at the Westinghouse plant, a complete 100% volumetric ultrasonic test is conducted on the flywheel and a liquid penetrant test is conducted on the bore.
   E. After the flywheel is installed and the motor is completely assembled, a 125% overspeed test for one minute is conducted on the assembled unit.
   F. Following the overspeed test, a periphery sonic test is conducted on the flywheel through access holes in the motor frame.
   G. To assure the original integrity of each flywheel during operation, the following inservice inspections will be performed:

At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an in-place, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed, if the interval
35. measured from the previous such inspections is greater than 6 2/3 years. 35.
Results of the examination will be evaluated by the original acceptance criteria and compared with the original examination data to assure the absence of unacceptable defects.

d. Shaft Design and Integrity

The shear stress on the shaft in the vicinity of the flywheel is 5520 psi with short circuit torque on the motor. The minimum strength of the shaft material is 23,000 psi which results in a safety factor of 4 under the maximum torque condition. Because of the conservatism used in the design of the shaft, it is concluded that shaft failure is not credible.

e. Bearing Design and Failure Analysis

The motor pump assembly is supported by a Kingsbury type thrust bearing which consists of a runner and upper and lower thrust plates. The history of the Kingsbury type bearing design indicates that the device is highly reliable and has a non-locking failure mode.

Provided on the motor are a number of devices to warn the operator of bearing trouble and these devices are each independent in their operation. The thrust bearing monitoring devices are as follow:

1. Two (2) thermocouples located diametrically opposite to each other in the upper thrust plates.
2. Two (2) thermocouples located diametrically opposite to each other in the lower thrust plates.
3. One (1) thermocouple in the upper oil pot.
4. Oil pot level alarm device.
5. Vibration device.

These devices are arranged to provide alarm indications to the control room operator. If a thrust bearing fails and the motor is shut down, the result would be melting of the bearing babbitt and, finally, automatic tripout of the motor on overload. Since seizure of the bearing will not result from a bearing failure, it is concluded that missles will not be produced.

f. Seismic Effects

The pump motor units are being analyzed against the combination effects of mechanical and seismic loads including the gyroscopic effects of the flywheel to verify that the stress limits will not be exceeded and the pump motor unit will operate through the maximum hypothetical earthquake.

g. Documentation and Quality Assurance

The Duke Power Company and the motor supplier, Westinghouse Electric Corp., have a rigid quality assurance program directed at assuring the integrity of the reactor coolant pump motors.

A quality assurance folder is developed by Duke Power Company on each motor and the folder includes the following:

1. Specifications and addendum
2. Description of the manufacturer's quality control organization and engineering order handling.
3. Copies of all inspection reports relating to the appropriate motor.
4. Samples of quality control drawings.

5. Copies of all test reports including flywheel material vendor test reports; Westinghouse motor test reports; bearing assembly reports; shaft tests; sonic test reports on the machined flywheel prior to assembly on the motor and following the 125% speed test; and certification on the motor test report that the overspeed test was conducted on the assembled motor.
6. Copies of the Duke Form QA-2 which is the manufacturer's certification to Duke Power Company Design Engineering that the motors were manufactured per specification and the Duke Power quality assurance program.
7. Copies of Duke Form QA-1 which the indication to the field quality control engineer that the motor described thereon was manufactured to the specification and the Duke quality assurance program.
8. Copies of Duke Form QC-31 which is the field receiving report on the motor.

A copy of each quality assurance folder is sent to the field quality control engineer and a copy is kept in the Design Engineering Dept. file.

4.2.2.7 Reactor Coolant Equipment Insulation

The reactor coolant system components are insulated with metal reflective type insulation. The insulation is supported by rings welded to weld pads on the components during field installation of the insulation. The weld pads to which the holding rings are attached are added to the components prior to final stress relief of the component.

The insulation units are removable and are designed for ease of removal and installation in such areas as field welds, nozzles, and bolted closures. The insulation units permit free drainage of any condensate or moisture from within the insulation unit.

4.2.3 SYSTEM PARAMETERS

4.2.3.1 Flow

The reactor coolant system is designed on the basis of 176,000 gpm flow rate in each heat transport loop.

4.2.3.2 Temperatures

Reactor coolant system temperatures as a function of power are shown in Figure 4-9. The system is controlled to a constant average temperature throughout the power range from 15% to 100% full power. The average system temperature is decreased between 15% and 0% of full power to the saturation temperature at 900 psia.

4.2.3.3 Heatup

All reactor coolant system components are designed for a continuous heatup rate of 100 F/hr.

4.2.3.4 Cooldown

All reactor coolant system components are structurally designed for a continuous cooldown rate of 100 F/h. System cooldown to 250 F is accomplished by use of the steam generators and by bypassing steam to the condenser with the turbine bypass system. The low pressure injection system provides the heat removal for system cooldown below 250 F.

4.2.3.5 Volume Control

4.2.3.5.1 Letdown

The only coolant removed from the reactor coolant system is that which is letdown to the high pressure injection system. The letdown flow rate is set at the desired rate by the operator positioning the letdown control valve and/or opening the stop valve for the letdown orifice.

4.2.3.5.2 Makeup

To maintain a constant pressurizer water level, total makeup to the reactor coolant system must equal that which is letdown from the system. Total makeup consists of the seal injection water through the reactor coolant pump shaft seals and makeup returned to the system through the reactor coolant volume control valve (high pressure injection system). The pressurizer level controller provides automatic control of the valve to maintain the desired pressurizer water level. Reactor coolant volume changes during plant load changes exceed the capability of the reactor coolant volume control valve, and thus result in variations in pressurizer level. The level is returned to normal as the system returns to steady state conditions.

4.2.3.6 Chemical Control

Control of the reactor coolant chemistry is a function of the chemical addition and sampling system. Sampling lines from the letdown line of the high pressure injection system provide samples of the reactor coolant for chemical analysis. All chemical addition is made from the chemical addition and sampling system to the high pressure injection system. See Section 9.2 for detailed information concerning the chemical addition and sampling system and the high pressure injection system.

4.2.3.6.1 Boron

Boron in the form of boric acid is used as a soluble poison in the reactor coolant. Concentrated boric acid is stored in the chemical addition and sampling system and is transported to the reactor coolant system in the same manner as described above for chemical addition. A second source of stored concentrated boric acid is that which is reclaimed from the coolant bled to the coolant storage system. This reclaimed boric acid is stored in the concentrated boric acid storage tank and is pumped to the high pressure injection system which transports it to the reactor coolant system. All bleed and feed operations for changing the boric acid concentration of the reactor coolant are made between the high pressure injection system and the coolant storage system. Section 9.2 contains detailed information concerning these two systems.

4.2.3.6.2 pH

The pH of the reactor coolant is controlled to minimize corrosion of the reactor coolant system surfaces which minimizes coolant activity and radiation levels of the components.

4.2.3.6.3 Water Quality

The reactor coolant water chemistry specifications, listed in Table 4-10, have been selected to provide the necessary boron content for reactivity control and to minimize corrosion of reactor coolant system surfaces. The solids content of the reactor coolant is maintained below the design level by minimizing corrosion through chemistry control and by continuous purification of the letdown stream of reactor coolant in the letdown filter and purification demineralizer of the high pressure injection system. Excess hydrogen is maintained in the reactor coolant to chemically combine with the oxygen produced by radiolysis of the water.

4.2.3.7 Flow Measurement

Reactor coolant flow rate is measured for each heat transport loop by a flow tube welded into the reactor outlet pipe. The power/flow monitor of the reactor protective system utilizes this flow measurement to prevent reactor power from exceeding a permissible level for the measured flow. This is discussed in further detail in Section 7.3.2.

4.2.3.8 Leak Detection

The entire reactor coolant system is located within the secondary shielding and is inaccessible during reactor operation. Any leakage drains to the reactor building normal sump. Any coolant leakage to the atmosphere will be in the form of fluid and vapor. The fluid will drain to the sump and the vapor will be condensed in the reactor building coolers and also reach the sump via a drain line from the cooler.

All power operated valves containing reactor coolant, located in the reactor building, have two sets of stem packing, with a leakoff connection between the packing sets. For the reactor coolant pump, any leakage past the primary mechanical seal is piped to the quench tank and any leakage past the backup seal is piped to the sump.

Locating the actual point of reactor coolant system leakage can most readily be accomplished when the reactor is shutdown, thereby allowing personnel access inside the secondary shielding. Location of leaks can then be accomplished by visual observation of escaping steam or water, or of the presence of boric acid crystals which would be deposited near the leak by evaporation of the leaking coolant.

Leakage of reactor coolant into the reactor building during reactor operation will be detected by one or a combination of the following methods.

<u>Sump and Tank Levels</u>

All leakage, both reactor coolant and cooling water is collected in the reactor building sump. The sump water level is indicated and annunciated at high level in the control room. Changes in sump water level are an indication of total leakage.

Measurement of the letdown storage tank coolant level provides a direct indication of reactor coolant leakage. Since the pressurizer level is maintained constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the letdown storage tank resulting in a decrease in tank level. Both the pressurizer and letdown storage tank coolant levels are recorded in the control room. A comparison of these two recordings over a time period yields the total reactor coolant leakage rate.

<u>Radioactivity</u>

Changes in the reactor coolant leakage rate in the reactor building may cause changes in the control room indication of the reactor building atmosphere particulate and gas radioactivities.

4.2.3.9 Vents and Drains

Vent and drain lines are shown on the system diagram, Figure 4-1. They are located at the high and low points of the system and provide the means for draining, filling, and venting the heat transport loops and pressurizer. The reactor vessel cannot be drained below the top of the reactor outlet nozzle using these drain lines. Each vent and drain line contains two manual valves in series. Vent lines are routed to a header connected to the quench tank gas space and drain lines are routed to a header connected to the suction of the component drain pump.

4.2.4 PRESSURE CONTROL AND PROTECTION

Normal reactor coolant system pressure control is by the pressurizer steam cushion in conjunction with the pressurizer spray, electromatic relief valve and heaters. The system is protected against overpressure by reactor protective system circuits such as the high pressure trip and by pressurizer relief valves located on the top head of the pressurizer. The schematic arrangement of the relief valves is shown in Figure 4-1. Since all sources of heat in the system, i.e., core, reactor coolant pumps, and pressurizer heaters, are interconnected by the reactor coolant piping with no intervening isolation valves, all relief valves are located on the pressurizer. Reactor coolant system pressure settings and relief valve capacities are listed in Table 4-1.

4.2.4.1 Pressurizer Code Safety Valves

Two pressurizer code safety valves are mounted on individual nozzles on the top head of the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston. The valve inlet and outlet is flanged to facilitate removal for maintenance or set point testing.

4.2.4.2 Pressurizer Electromatic Relief Valve

The pressurizer electromatic relief valve is mounted on a separate nozzle on the top head of the pressurizer. The main valve operation is controlled by the opening or closing of a pilot valve which causes unbalanced forces to exist on the main valve disc. The pilot valve is opened or closed by a solenoid in response to the pressure set points. Flanged inlet and outlet connections provide ease of removal for maintenance purposes.

4.2.4.3 Pressurizer Spray

The pressurizer spray line originates at the discharge of a reactor coolant pump in the same heat transport loop that contains the pressurizer. Pressurizer spray flow is controlled by an electric motor operated valve using on-off control in response to the opening and closing pressure set points. An electric motor operated valve in series with the spray line is to provide for remote spray line isolation.

4.2.4.4 Pressurizer Heaters

The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressure during reactor coolant system heatup from the cooled down condition, and restore system pressure following transients. The heaters are grouped in four banks and are controlled by the pressure controller. The first bank utilizes proportional control and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at the set point. On-off control is used for the remaining three banks. A low level interlock prevents the heaters from being energized with the heaters uncovered.

4.2.4.5 Relief Valve Effluent

Effluent from the pressurizer electromatic-relief and code safety valves discharges into the quench tank which condenses and collects the relief valve effluent. This is shown schematically in Figure 4-1. After the quench tank receives relief valve effluent, the tank contents are cooled to normal temperature by the component drain pump and quench tank cooler of the coolant storage system. The tank fluid is circulated from the tank through the cooler and returned to the tank by spraying into the tank vapor space. The quench tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the pressurizer electromatic relief valve. The quench tank can be remotely vented to the gaseous waste disposal system (Section 9.9).

4.2.4.6 Cooldown

Reduction of pressure during reactor coolant system cooldown is accomplished by the pressurizer spray provided by the reactor coolant pump. Below a system temperature of approximately 250° F, the low pressure injection system is used for system heat removal and the steam generators and reactor coolant pumps are removed from service. During this period, spray flow is provided by a branch line from one low pressure injection line to the pressurizer spray line for further pressure reduction or complete depressurization of the reactor coolant system.

4.2.4.7 Sampling

A sample line from the pressurizer steam space to the chemical addition and sampling systems permits detection of non-condensible gases in the steam space. This sample line also permits a bleeding operation from the vapor space to the letdown line of the high pressure injection system to transport accumulated non-condensible gases in the pressurizer to the letdown storage tank.

4.2.5 INTERCONNECTED SYSTEMS

4.2.5.1 Low Pressure Injection

The low pressure injection system provides the capability for cooling the reactor coolant system below about 250 F during plant cooldown. During this mode of operation, coolant is drawn from the reactor coolant system through a nozzle on the reactor outlet pipe, circulated through the low pressure injection coolers by the low pressure injection pumps and then injected back into the reactor coolant system through two nozzles on the reactor vessel into the inlet side of the core. The heat received by this system is rejected to the low pressure service water system. Components in these two systems are redundant for reliability purposes. These two systems are described in Section 9.

The low pressure injection system also performs an emergency injection function for a loss of coolant accident and provides long term emergency core cooling; this is described in Section 6.

4.2.5.2 High Pressure Injection

The high pressure injection system controls the reactor coolant system coolant inventory, provides the seal water for the reactor coolant pumps, and recirculates reactor coolant system letdown for water quality maintenance and reactor coolant boric acid concentration control. Letdown of reactor coolant is through a nozzle on the outlet coolant pipe from one steam generator. The discharge of the high pressure injection pumps connects to a nozzle on each of the reactor inlet pipes downstream of the reactor coolant pumps. The reactor coolant which is letdown is returned to the reactor coolant system through the nozzles in a different heat transport loop from the heat transport loop containing the letdown line. Components are redundant for reliability purposes (Section 9.1).

The high pressure injection system utilizes four injection nozzles in carrying out the high pressure emergency injection function after a loss of coolant accident. This is described in Section 6.1.2.1.1.

4.2.5.3 Core Flooding System

The core flooding system floods the core in the event of a loss of coolant accident. Connection to the reactor vessel is through the two nozzles described above for low pressure injection. The low pressure injection and core flooding lines tie together and connect to the same nozzle on the reactor vessel. The core flooding system is described in Section 6.1.2.1.3.

4.2.5.4 Secondary System

The principal decay heat removal system interconnected with the reactor coolant system is the steam and power conversion system. The reactor coolant system is dependent upon the steam and power conversion system for decay heat removal at normal operating conditions and for all reactor coolant operating temperatures above 250 F. The system is discussed in detail in Section 10.2.

The turbine bypass system routes steam to the condensers when the turbine has tripped or is shutdown and also during large plant load reduction transients when steam generation exceeds the demand. Overpressure protection for the secondary side of the steam generators is provided by the turbine bypass system and by safety valves mounted on the main steam lines outside of the reactor building. The auxiliary feedwater system will supply water to the steam generators in the event that the main feedwater system is inoperative. The physical layout of the reactor coolant system provides natural circulation of the reactor coolant to ensure adequate core cooling following a loss of all reactor coolant pumps.

4.2.6 COMPONENT FOUNDATIONS AND SUPPORTS

The supports for all major components listed in this section are analyzed in
4. detail to insure adequate structural integrity for their intended function during normal operating, seismic, and accident conditions. Following calculation of sources of loading; stresses and motions at significant locations are computed and compared to applicable criteria. Details of this analysis are given in Appendix 4B.

4.2.6.1 Reactor Vessel

The reactor vessel is bolted to a reinforced concrete foundation designed to support and position the vessel and to withstand the forces imposed on it by a combination of loads including the weight of vessel and internals, thermal expansion of the piping, design basis earthquake, and dynamic load following reactor trip.

The foundation, in addition, restrains the vessel during the combined forces imposed by the circumferential rupture of a 36-inch reactor outlet line and a simultaneous maximum hypothetical earthquake.

The vessel foundation further is designed to provide accessibility for the installation and later inspection of incore instrumentation, piping, and nozzles; to contain ductwork and vent space for cooling air to remove heat losses from the vessel insulation; and to provide a sump and drainage line for leak detection.

4.2.6.2 Pressurizer

The pressurizer is supported on a structural steel foundation by eight (8) lugs welded to the side of the vessel.

The foundation and supports are designed to withstand the loads imposed by thermal expansion of the pressurizer, the weight of the pressurizer including its contents and attached piping, relief valve reaction forces, and forces imposed by the design basis earthquake. In addition, the foundation and supports will restrain the vessel during the combined forces imposed by the circumferential rupture of the 10-inch surge line coupled with the maximum hypothetical earthquake.

The foundation is also designed to permit accessibility to pressurizer surfaces for inspection.

4.2.6.3 Steam Generator

The steam generator foundation is designed to support and position the generator. The foundation is designed to accept the loads imposed by the generators and feedwater piping filled with water, the attached reactor coolant piping also filled with water, and steam lines under the design basis earthquake and the maximum hypothetical earthquake.

The foundation is also designed to restrain the steam generator under the combined forces due to a circumferential rupture of a 28-inch coolant line and a simultaneous maximum hypothetical earthquake.

Forces imposed on the generator by the rupture of a 36-inch coolant line are transferred to the shielding walls by a support structure located near the top of the generator.

4.2.6.4 Piping

The reactor coolant piping, inlet and outlet lines, are supported by the reactor vessel and steam generator nozzles. The piping will withstand the forces imposed on it by the design basis earthquake and the maximum hypothetical earthquake.

4.2.6.5 Pump and Motor

The reactor coolant pump casing, internals, and motor weight are supported by the 28 inch coolant lines and constant load hangers attached to the motor. In the cold condition, the coolant piping will support the coolant pump and motor without the hangers. The hangers are designed to withstand the forces imposed on them by the design basis earthquake and the maximum hypothetical earthquake.

4.2.6.6 LOCA Restraints

Each steam generator has a support located opposite the upper tube sheet and transfers forces from the generator into the shield walls in the event of a circumferential rupture of the 36 inch line.

Each 28 inch reactor coolant inlet line and 36 inch reactor coolant outlet line has a restraint located outside of and bolted to the primary shield to limit pipe motion in the event of a circumferential rupture of the piping inside the primary shield.

5. 4.2.7 MISSILE PROTECTION AND PIPE WHIP PROTECTION (Ref. 4.3.2)

The major components including reactor vessel, reactor coolant piping, reactor coolant pumps, steam generators, and the pressurizer are located within three shielded cubicles. Each of two cubicles contain one steam generator, two coolant pumps, and associated piping. One of the cubicles also contains the pressurizer. The reactor vessel is located within the third cubicle or primary shield. The reactor vessel head and control rod drives extend into the fuel transfer canal.

Penetrations in the generators, piping, and the pressurizer are located such that missiles which may be generated, such as valves, valve bonnets, valve stems, or reactor coolant temperature sensors will not escape the cubicles or possess sufficient energy to damage the reactor building liner plate.

Openings are provided in the lower shield walls to provide vent area. To assure that no missile will impact on the reactor building liner plate, concrete shielding is provided for the liner plate area opposite the openings. The shielding extends beyond the openings so that any missile will impact on the shields.

Pipe lines carrying high pressure injection water are routed outside the shield walls entering only when connecting to the loop. Missiles which may be generated in one cubicle cannot rupture high pressure injection lines for the other loop. Low pressure injection lines and core flooding lines are routed outside of the shield walls, behind missile shield walls, and through the primary shield where they enter the reactor vessel. They are, thus, protected from missiles which might be generated in either cubicle.

A concrete missile shield is located above the control rod drives to stop a control rod drive should it become a missile. The shield is removed during refueling.

In addition, items that could become missiles are oriented so they impinge on concrete surfaces.

Analysis of the missile penetration is based on the methods described in Nav. Docks P-51, Design of Protective Structures by Amirikan (Bureau of Yards and Docks, August 1950).

The penetration formulae are:

$$D = k\ ApV^1 \qquad V^1 = \log_{10}\left(1 + \frac{V^2}{215000}\right)$$

$$K = \frac{D^1}{D} = \left[1 + e^{-4\ (a^1-2)}\right]; \quad a^1 = \frac{T}{D}$$

where:

D = Penetration in a slab of infinite thickness (ft.)
D^1 = Penetration is a slab of thickness "T" (ft.)
T = Thickness of slab (ft.)
5. Ap = Sectional pressure, obtained by dividing the weight of missile by its cross sectional area (psf)
V = Velocity of missile (fps)
k = Material's coefficient, in our case, $k=2.30 \times 10^{-3}$ for reinforced concrete

Formulae for determining energy loss due to drag:

$$\frac{Ti}{Tc} = \frac{1}{1 + \frac{2Tc}{WL}} \quad ; \qquad L = \frac{2W}{SAC_d}$$

where:

A = Average area
Cd = Drag coefficient ($Cd \approx 1.0$ in our case)
Ti = Kinetic energy on impact
Tc = Kinetic energy after leaving casing
W = Weight in lbs.
S = Air density = 0.074 $\#/ft^3$

The study of postulated ruptures made by the turbine generator supplier concludes that the missile having the highest combination of weight, size and energy effects is the last stage wheel. The properties of this missile are shown in Table 5-2, Section 5.1.3.2.

In addition to the penetration calculation, the overall structural strength of the removable concrete slabs, its supports and anchors were analyzed based on the research paper "Impact Effect of Fragments Striking Structural Elements" by R. A. Willimson and R. R. Alvy.

The following three missiles were used to design the removable concrete slabs:

Description	Wt. Lbs.	Imp. Area In^2	Velocity FPS	Kin. Energy Ft-lbs.
C.R. Drive Assembly	1500	64.0	254	1.49×10^6
CRD Vent Cap w/valve	55	13.4	546	0.12×10^6
CRD Motor and Clutch Assem.	750	47.0	483	1.35×10^6

The properties of other missiles postulated by the NSSS supplier are given in Table 4-22 to 4-28 (pages 4-63 to 4-71).

4.3 SYSTEM DESIGN EVALUATION

4.3.1 DESIGN MARGIN

The reactor coolant system is designed structurally for 2,500 psig and 650 F. The system will normally operate at 2,155 psig and 604 F.

In the event of a complete loss of power to all reactor coolant pumps, reactor coolant flow, coastdown and subsequent natural circulation flow is more than adequate for core cooling and decay heat removal as shown by the analysis in Section 14.1

The number of transient cycles specified in Table 4-8 for the fatigue analysis is conservative. Twelve heatup and cooldown cycles per year are specified, where the system may not be required to complete more than one or two cycles per year in actual operation. A heatup rate of 100 F/h was used in the analysis of Transients 1 and 2 in Table 4-8.

4.3.2 MATERIAL SELECTION

Each of the materials used in the reactor coolant system has been selected for the expected environment and service conditions. The major component materials are listed in Table 4-9. All reactor coolant system materials normally exposed to the coolant are corrosion-resistant materials consisting of 304 or 316 stainless steel, Inconel, 17-4PH(H1100), Zircaloy, or weld deposits with corrosion-resistant properties equivalent to or better than those of 304 SS. These materials were chosen for specific uses at various locations within the system because of their compatibility with the reactor coolant. There are no novel material applications in the reactor coolant system.

To assure long steam generator tube lifetime, feedwater quality entering the steam generator is maintained within the specification given in Table 4-11 in order to prevent deposits and corrosion inside the steam generator. These required feedwater specifications have been successfully used in comparable once-through non-nuclear steam generators.

The selection of materials and the manufacturing sequence for the reactor coolant system components, is arranged to insure that no pressure boundary material is furnace-sensitized stainless steel. Safe ends are provided on those carbon steel nozzles of the system vessels which connect to stainless steel piping. All dissimilar metal welds, with the exception of Inconel to Stainless Steel pipe welds, will be made in the manufacturer's shops.

Piping systems designed to resist seismic forces have been restrained by steel supports capable of withstanding these seismic forces. The restraints also act as pipe stops restraining the lines against whipping. In systems, where it was necessary to use hydraulic snubbers to resist seismic forces, the mechanical action associated with the snubbers makes it possible to consider them as re-
5. straints against pipe whipping. When a seismic acceleration equal to or greater than two (2) feet/sec./sec. acts on the system, a differential pressure is generated on the ends of the snubber valve piston which is spring centered. This differential pressure is sufficient to cause the piston to shift and close the by-pass ports. With the by-pass ports closed, the snubber acts as a rigid

structural member, thus limiting any further movement of the pipe at the point of attachment.

The basic design criteria for pipe whip protection is as follows:

(a) All penetrations are designed to maintain containment integrity for any loss of coolant accident combination of containment pressures and temperatures.

(b) All penetrations are designed to withstand line rupture forces and moments generated by their own rupture as based on their respective design pressures and temperatures.

(c) All primary penetrations, and all secondary penetrations that would be damaged by a primary break, are designed to maintain containment integrity.

(d) All secondary lines whose break could damage a primary line and also breach containment are designed to maintain containment integrity.

4.3.3 REACTOR VESSEL

Stress Evaluation

A stress evaluation of the reactor vessel has been performed in accordance with Section III os the ASME Code. The evaluation shows that stress levels are within the Code limits. Table 4-14 lists the reactor vessel steady-state stresses at various load points. The results of the transient analysis and the determination of the fatigue usage factor at the same load points are listed in Table 4-15. As specified in the ASME Code, Section III, Paragraph 415.2(d)(6), the cumulative fatigue usage factor is less than 1.0 for the design cycles listed in Table 4-8. Figure 4-4c illustrates the points of stress analysis for the stresses listed in Table 4-14 and the fatigue usage factors listed in Table 4-15. These stress summaries demonstrate that all of the requirements for stress limits and fatigue required by ASME Section III for all of the operational requirements imposed by the design specifications have been met. The values tabulated in these summaries are the maximum value obtained in each region. The imposed transients are based on description of the realistic behavior that might be expected for this plant. Transients such as loss of flow and load that cause temperature and pressure variations are included in the reactor vessel specification and Table 4-8. Their effect on accumulated usage factor is included in the stress analysis and in the values reported in summary Table 4-15. These transients are not the major contributors to the largest usage factor of 0.38 for the stud bolts as given in Table 4-15. Table 4-16 presents a summary of the major reactor vessel
1. material physical properties including the results of impact tests. Table 4-17 lists the chemical analysis results for the same material.

Nil-Ductility Transition Temperature (NDTT)

The reactor vessel plate material opposite the core was purchased to mill practices which improve material toughness and result in a lower range of NDTT values for heavy sections. The raw material was purchased to be capable of meeting Charpy V-notch values of 30 ft-lb or greater at a temperature of 10°F. The material was tested during vessel fabrication after forming to show conformity to specified requirements or to determine the actual temperature at which the specified 30 ft-lb Charpy V-notch value was met.

The unirradiated or initial NDTT of pressure vessel base plate material is presently measured by the Charpy V-notch impact test (Type A) given in ASTM E23. 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 or SA-533B steel, the ASME Section III Table N-332 specifies an
1. energy value of 30 ft-lb. A curve of the temperature versus the energy absorbed in breaking the specimen is plotted. To obtain this curve at least two specimens are tested at a minimum of five different temperatures. Available data indicate NDTT differences as great as 40°F between curves plotted through the minimum and average values respectively. The intersection of the energy versus temperature curve with the 30 ft-lb ordinate is designated as the NDTT. The determination of NDTT from the average curve is considered representative of

the material and is consistent with procedures specified in ASTM E23. The material for these tests is treated by the methods outlined in ASME III Paragraph N-313. The test coupons are taken at a distance of T/4 (1/4 of the plate thickness) from the quenched surfaces and at a distance of T from the quenched edges. These tests are performed by the material supplier or B&W, in accordance with ASME Section III, Paragraphs N-313 and N-330.

The basic determination of vessel operation from cold startup and shutdown to full pressure and temperature operation is performed in accordance with a "Fracture Analysis Diagram" as published by Pellini and Puzak.[2] At temperatures below the Design Transition Temperature (DTT), which is equal to NDTT + 60°F, the pressure vessels will be operated so that the stress levels will be restricted to a value that will prevent brittle failure. These levels are:

Rev. 4. 4/20/70
(Carry-Over)

a. Below the temperature of DTT minus 200 F, a maximum stress of 10 per cent yield strength.

b. From the temperature of DTT minus 200 F to DTT, a maximum stress which will increase from 10 to 20 per cent yield strength.

c. At the temperature of DTT, a maximum stress of 20 per cent yield strength.

With the stresses held within the above limits (a through c above), brittle fracture will not occur. This statement is based on data reported by Robertson(3) and Kihara and Masubichi(4) in published literature. These stress values are interpreted in terms of operating temperatures and pressures, and it can be shown that stress limits can be controlled by imposing operating procedure limits through control of pressure and temperature during heatup and cooldown.(1,3) This procedure assures that the stress levels do not exceed those specified in a through c above.

Flux and nvt at Reactor Vessel Wall

The design value for the fast neutron flux greater than 1.0 MeV at the inner surface of the reactor vessel is 3.0×10^{10} n/cm^2-s at a rated power of 2,568 MWt. The corresponding calculated maximum fast neutron flux at the vessel wall is 2.2×10^{10} n/cm^2-s. This calculated value includes a lifetime average axial peaking factor of 1.3 and an azimuthal peaking factor of 1.29. For 40 years at 80% load this corresponds to an nvt of 2.2×10^{19} n/cm^2 (maximum) for the vessel wall.

The attenuation of the neutron flux from the core to the reactor vessel is computed using the NRN(5) program. This is a one-dimensional multigroup removal-diffusion program in slab, cylindrical, or spherical geometry. This code uses the method in which the uncollided and strongly forward-scattered neutrons (removal groups) are computed by integration of an energy dependent attenuation kernel over the source volume. Scattering of neutrons out of the removal groups forms a source term for the multigroup diffusion calculations. Neutron slowing-down is handled by elastic and non-elastic scattering matrices for both the removal and diffusion groups.

Neutron fluxes at the vessel wall are computed with the core represented as a slab source equal in thickness to the equivalent core diameter. A lifetime average power distribution through the thickness of the core was determined from calculated power profiles over several core cycles and at various times during each cycle. The neutron energy spectrum was represented by 26 energy groups with 14 of these groups covering the range above 1.0 MeV.

Local flux peaking on the vessel wall due to fuel assemblies extending beyond the equivalent core diameter (azimuthal peaking) is determined with PDQ 5,(6) a two-dimensional diffusion program. The lifetime-average axial flux peak at the vessel is the same as that in the two outer rows of fuel assemblies.

Calculations with the NRN code were compared with data from various experiments including measurements on the R2-0 reactor at Studsvik Research Center,(7) on the LIDO pool reactor at Harwell,(8) and on a shielding mockup of the reactor vessel and internals at the B&W Critical Experiment Laboratory (CEL).[9] At the R2-0 reactor, measurements were made through about 3 feet of water with threshold detectors which included ^{115}In (n,n^{1}) 115mIn (1.5 MeV), ^{32}S (n,p) ^{32}P (3 MeV), and ^{27}Al (n,α), ^{24}Na (6 MeV). Energies shown are threshold energies for the reaction. In the LIDO pool, thermal flux measurements were made through laminations of iron and water over a penetration distance of about 4 feet. In the experiment at B&W's CEL, sulphur foil data was taken at points covering the distance between the core and the reactor vessel.

In all cases, and over the entire penetration distances, the calculations were either in agreement with the data, or predicted higher flux and activation levels. It is thus concluded that the NRN code provides a conservative method for the calculation of vessel nvt.

Expected NDTT Shift

As a result of fast neutron bombardment of vessel metal in the region surrounding the core, the reactor vessel material ductility will change. The effect is an increase in the Nil-Ductility Transition Temperature (NDTT). For the 40 year exposure (Section 4.1.2.8), the predicted NDTT shift, shown in Figure 4-10, is 250 F. This is based on the "Maximum Curve for 550 F Data" shown in Figure 4-11. The "Trend Curve for 550 F Data," as shown in Figure 4-11, represents irradiated material test results and was compiled from the reference documents listed in Table 4-13.

The NDTT shift is factored into the plant startup and shutdown procedures so that full operating pressure is not attained until the reactor vessel temperature is about DTT. The total stress in the vessel wall due to both pressure and the associated heatup and cooldown transient is restricted to 5,000-10,000 psi, which is below the threshold of concern for safe operation. An adjusted 100 F/h heatup rate can be maintained throughout life. An adjusted rate is one in which the pressure is held constant to maintain stresses at the desired low level while temperatures are at a level below DTT. A 100F/h temperature increase is maintained until DTT is passed and pressure can be raised to a new
4. higher level. These operating restrictions are based on the NRL generalized fracture analysis diagram which is a semi-empirical method of material selection and approximate analysis to prevent brittle fracture. This diagram plots failure stress (normalized to yield) as a function of temperature referenced to the NDT temperature for a family of finite flaw sizes. The parametric crack size curves were determined partially by fracture mechanics and partially by plotting actual failure data. The assumed flaw for this analysis was slightly greater than 24 inches.

The maximum stress in the reactor vessel wall caused by internal heat generation from gamma radiation occurs at the core midplane region during full power operation. The value of stress is 3160 psi (tension). There are no structural discontinuities in this region to cause stress concentration.

The reactor vessel design provides for vessel material surveillance specimens which will permit an evaluation of the actual neutron exposure-induced shift for material Nil-Ductility Transition Temperature.

Test coupons of welds, heat-affected zones, and base material for the material used in the reactor vessel, are incorporated in the reactor vessel surveillance program, as described in 4.4.6. The Reactor Vessel Material Surveillance Program is described in further detail in a Babcock & Wilcox company topical report, BAW-10006; "Reactor Vessel Material Surveillance Program."

Fracture Mode Evaluation

An analysis has been made to demonstrate that the reactor vessel can accommodate without failure the rapid temperature change associated with the postulated operation of the emergency core cooling system (ECCS) at end of vessel design life. A summary of the evaluation follows:

The state of stress in the reactor vessel during the loss-of-coolant accident was evaluated for an initial vessel temperature of 603 F. The inside of the vessel wall is rapidly subjected to 90 F injection water of the maximum flow rate obtainable. The results of this analysis show that the integrity of the vessel is not violated.

The assumed modes of failure are ductile yielding and brittle fracture, which includes the nil-ductility approach and the fracture mechanics approach. The modes of failure are considered separately as follows:

a. Ductile Yielding

The criterion for this mode of failure is that there shall be no gross yielding across the vessel wall using the minimum specified yield strength in the ASME Code, Section III. The analysis considered the maximum combined thermal and pressure stresses through the vessel wall thickness as a function of time during the safety injection. Comparison of calculated stresses to the material yield stress indicated that local yielding may occur in the inner 8.0 per cent of the vessel wall thickness.

b. Brittle Fracture

Because the reactor vessel wall in the core region is subjected to neutron flux resulting in embrittlement of the steel, this area was analyzed from both a nil-ductility approach and a fracture mechanics approach. The results of the two methods of analysis compare favorably and show that pressure vessel integrity is maintained.

The criterion used in the nil-ductility approach is that a crack cannot propagate beyond any point where the applied stress is below the threshold stress for crack initiation (5-8 ksi) or when the stress is compressive. This approach involves making the very conservative assumption that all of the vessel material could propagate a crack by a low energy absorption or cleavage mode. End-of-life vessel conditions were assumed. The crack arrest temperature through the thickness of the wall was developed on a stress-temperature coordinate system. The actual quench-induced, stress-temperature condition through the thickness of the wall at several times during the quench was developed and plotted. The maximum depth at which the material in the vessel wall would be in tension or at which the stress in the material would be in excess of the threshold stress for crack initiation (5-8 ksi) was determined by comparison of the plots. The comparison showed that a crack could propagate only through the inner 35 per cent of the wall thickness if a crack initiation threshold of 5-8 ksi is applicable.

The foregoing method of analysis is essentially a stress analysis approach which assumes the worst conceivable material properties and a flaw size large enough to initiate a crack. Actually, the outer 83 per cent of the vessel wall is at a temperature above the DTT (NDTT + 60 F) when credit is taken for the neutron shielding, and for the original DTT profile through the wall thickness. The analysis is conservative in that it does not deny that cracks can be initiated, and in that it assumed a crack from 1 to 2-ft long to exist in the vessel wall at the time of the accident. Therefore, it can be concluded that, if a crack were present in the worst location and orientation (such as a circumferentially oriented crack on the inside of the vessel wall), it could not propagate through the vessel wall.

A fracture mechanics analysis was conducted which assumed a continuous surface flaw to exist on the inside surface of the vessel wall. The criterion used for the analysis is that a crack cannot propagate when the stress intensity at the tip of the crack is below the critical crack stress intensity factor (K_{IC}).

Babcock & Wilcox Topical Report No. 10018; "Analysis of the Structural Integrity of the Reactor Vessel Subjected to a Thermal Shock Induced by the Operation of the Emergency Core Cooling System," provides the details of the analysis. This report includes an evaluation considering the Irwin fracture mechanics method and performs a sensitivity analysis of the effect of varying the conservatism of several major parameters on the result.

Closure

The reactor closure head is bolted to a ring flange on the reactor vessel. The vessel closure seal is formed by two concentric metal O-ring seals with provisions for leak-off between the O-rings. Reactor closure head leakage will be negligible from the annulus between the metallic O-ring seals during vessel steady-state and virtually all transient operating conditions. Only in the event of a rapid transient operation, such as an emergency cooldown, would there be some leakage past the inner-most O-ring seal. A stress analysis on a similar vessel design indicates this leak rate would be approximately 10 cc/min and no leakage would occur past the outer O-ring seal.

The reactor closure head is attached to the reactor vessel with sixty 6-1/2 in. diameter studs. The studs have a minimum yield strength of 130,000 psi. The studs, when tightened for operating conditions, will have a tensile stress of approximately 30,000 psi. An evaluation of stud failures shows that:

a. 10 adjacent studs can fail before a leak occurs.

b. 25 adjacent studs can fail before the remaining studs reach yield strength.

c. 26 adjacent studs can fail before the remaining studs reach the ultimate tensile strength.

d. 43 symmetrically located studs can fail before the remaining studs reach yield strength.

The fatigue evaluation results of the studs is included in Table 4-15.

Control Rod Drive Service Structure

The control rod drive service structure is designed to support the control rod drives to assure no loss of function in the event of a combined loss of coolant accident and maximum hypothetical earthquake. Requirements for rigidity, imposed on the structure to avoid adversely affecting the natural frequency of vibration of the vessel and internals, as well as space requirements for service routing, result in stress levels considerably lower than design limits. The structure is more than adequate to perform its required function.

4.3.4 STEAM GENERATORS

4.3.4.1 Research and Development

A. Test Program Design Criteria

In August of 1964, B&W began design and construction of facilities to test full scale sections of the Once Through Steam Generator. Since that time, three different test models of the Once Through Steam Generator have been tested. The design criteria for the test steam generators were as follows:

1. To provide a test steam generator for investigation of the operational characteristics of the steam generator such as heat transfer, pressure drop, control characteristics (including measurements necessary for control), and stability.

2. To provide a test steam generator for investigation of manufacturing procedures, fouling characteristics, and cleaning procedures.

3. To provide a test steam generator which could be non-destructively examined and analyzed with respect to vibration, corrosion, and unit integrity.

B. Test Program Design Bases

The design bases for the test steam generators were:

1. To duplicate tube length, tube thickness, tube diameters of the full size
6. steam generator.

2. To duplicate important dynamic characteristics such as secondary flow area per tube, downcomer annulus area and feedwater spray velocity.

3. To operate the test units under temperature, pressure, and control conditions of the full size units.

C. General Test Objectives

The general objectives of the model tests include:

1. heat transfer tests
2. pressure drop tests
3. stability tests
4. fouling and cleaning tests
5. mechanical design tests including vibration, and structural tests.

D. Test Results

In April, 1971, B&W submitted a topical report, BAW-10027, Once Through Steam
21. Generator Research and Development Report. General results and evaluation of the model tests including the following were reported in BAW-10027:

1. The steady state and transient operation tests have confirmed the analytically predicted performance characteristics of the steam generator and have provided the data for the control system.

2. Feedwater spray nozzle tests have demonstrated that the design will satisfactorily heat the feedwater.

3. Tube leak simulation tests have demonstrated that a leak in one tube will not propagate by causing a failure in adjacent tubes.

4. Mechanical tests have demonstrated that the tubes can withstand, without failure, the mechanical loads they may experience either during normal operation or accident conditions.

5. Vibration testing demonstrated that the unit contained no undesirable resonance characteristics.

6. Tests to simulate a steam line failure or reactor coolant system failure have demonstrated the integrity of the steam generator under conditions of rapid depressurization and large temperature differentials between the tubes and the shell of the unit.

7. Secondary side fouling tests demonstrated that fouling will be detected by increased pressure drop in the downcomer. Feedwater nozzle flooding causes
6· the downcomer water temperature to fall below saturation temperature. Feedwater nozzle flooding is prevented in high downcomer level limits which restrict and/or limit feedwater flow. Cleaning of the secondary side of the steam generator is required when the high downcomer level limit is activated at full power. If the operator chooses, cleaning may be postponed indefinitely by reducing the power level to the point at which the high downcomer level limit is not actuated.

21. 8. Additional information concerning steam generator research and development, design programs, and evaluations are contained in BAW-10027 as follows:

   a. Objectives and evaluations of all model steam generator tests.
   b. Extrapolation of model tests to full size performance.
   c. Verification test program to be conducted at Oconee I.
   d. Cleaning processes to be used.
   e. Computer programs used in the design of the steam generator and transient analysis.

4.3.4.2 Stress Evaluation

A. General

Because the steam generator is of a straight tube-straight shell design and because of a minor difference in the coefficient of thermal expansion between Inconel and carbon steel, there exists a structural limitation on the mean temperature difference between the tubes and the shell. During normal operation of the steam generator, the tube mean temperature should not be more than 32 F higher than the shell mean temperature. The maximum calculated mean tube to shell ΔT at normal operating conditions poses no problems to the structural integrity of the reactor coolant boundary. The effect of loss of reactor coolant would impose tensile stresses on the tubes and cause slight yielding across the tubes. Such a condition would introduce a small permanent deformation in the tubes but would in no way violate the boundary integrity. The rupture of a secondary pipe would cause the tubes to become warmer than the shell and may cause tube deformation. Blowdown tests simulating secondary side blowdown on a 37-tube model boiler, show that although a slight buckling in the tubes occurred, there was no loss of reactor coolant.

Calculations confirm that the steam generator tube sheet will withstand the loading resulting from a loss-of-coolant accident. The basis for this analysis is a hypothetical rupture of a reactor coolant pipe resulting in a maximum design pressure differential from the secondary side of 1050 psi. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet).

The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 15,900 psi across the center ligaments which is well below the ASME Section III allowable limit of 40,000 psi at 650 F. Under the condition postulated, the stresses in the primary head show only the effect of its role as a structural restraint on the tube sheet. The stress intensity at the juncture of the spherical head with the tube sheet is 14,970 psi which is well below the allowable stress limit. It can therefore be concluded that no damage will occur to the tube sheet or the primary head as a result of this postulated accident.

In regard to tube integrity under loss of reactor coolant, actual pressure tests of 5/8 in. OD/0.034 inch wall Inconel tubing show collapse under an external pressure of 4,950 psi. This is a factor of safety of 4.7 against collapse under the 1,050 psig accidental application of external pressure to the tubes.

The rupture of a secondary pipe has been assumed to impose a maximum design pressure differential of 2,500 psi across the tubes and tube sheet from the primary side. The criterion for this accident permits no violation of the reactor coolant boundary (primary head, tube sheet, and tubes).

To meet this criterion, the stress limits delineated in the ASME Pressure Vessel Code, Section III, Paragraph N-714.2 for hydrotest limitations are applicable for the aforementioned abnormal operating circumstance. The referenced section states that the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed 90% of the material yield stress at the operating temperature; in addition, the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving a maximum stress, does not exceed 135% of the material yield stress at the operating temperature.

An examination of stresses under these conditions show that for the case of a 2,500 psig design pressure differential, the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 4-18.

Rev. 6. 6/22/70
(Carry-over)

The basic design criterion for the tubes assumes a pressure differential of 2,500 psig in accordance with Section III. Therefore, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

The superimposed effect of secondary side pressure loss and maximum hypothetical earthquake has been considered. For this condition, the criterion is that there be no violation of the primary to secondary boundary (tube and tube sheet). For the case of the tube sheet, the maximum hypothetical earthquake loading will contribute an equivalent static pressure loading over the tube sheet of less than 5 psi (for vertical shock).

The effect of fluid dynamic forces on the steam generator internals under secondary steam break accident conditions has been simulated in a 37-tube laboratory boiler. Results of the tests show that reactor coolant boundary integrity is maintained under the most severe mode of secondary blowdown.

Rev. 6. 6/22/70
(Carry-over)

The ratio of allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses for a design pressure differential of 2,500 psig are summarized in Table 4-19.

B. Stress Intensities and Cumulative Usage Factors

Table 4-29 lists the steam generator stress intensities at various load points due to design conditions as defined in the design specifications.

The results of the transient analysis and the determination of the fatigue usage factor at the same load points are listed in Table 4-29.

21. C. Additional Information

6 Additional information discussed in BAW-10027 includes:

1. Discussion of thermal fatigue due to fluctuation and shifting of the liquid-vapor interface on the tubes.
2. Stress distributions and effective elastic constants obtained under thermal inplane and transverse loadings, and analysis of tube to tube sheet complex.
3. Vibration Analyses.

4.3.5 RELIANCE ON INTERCONNECTED SYSTEMS

The principal heat removal system interconnected with the reactor coolant system is the steam and power conversion system. This system provides capability to remove reactor decay heat for the hypothetical case where all station power is lost. Under these conditions decay heat removal from the reactor core is provided by the natural circulation characteristics of the reactor coolant system. The turbine driven emergency feedwater pump supplies feedwater to the steam generators. Cooling water flow to the condenser is provided by the emergency discharge line which discharges to the tailrace of the Keowee Dam. The analysis for this unlikely condition of total loss of station electric power is presented in Section 14. Should the condenser not be available to receive the steam generated by decay heat, which is unlikely in view of emergency discharge line flow, the water stored in the feedwater system can be pumped to the steam generators and the resultant steam vented to atmosphere to provide required cooling.

4.3.6 SYSTEM INTEGRITY

The reactor protective system (Section 7) monitors parameters related to safe operation and trips the reactor to protect against reactor coolant system damage caused by high system pressure. The pressurizer code safety valves prevent reactor coolant system overpressure after a reactor trip as a result of reactor decay heat and/or any power mismatch between the reactor coolant system and the secondary system.

The integrity of the reactor coolant system is ensured by proper materials selection, fabrication quality control, design and operation. A summary of fabrication inspections for the components is given in Table 4-12. Components in the system are fabricated from materials initially having a low NDTT to eliminate the possibility of propagating type failures. Materials surveillance specimens inside the reactor vessel will provide a check on the predicted shift in NDTT. A complete stress analysis has been prepared for all design loadings specified in the design specification. The analysis shows that the reactor vessel, steam generators, pressurizer and pump casings comply with the allowable stress limits of Section III of the ASME Code and the requirements of the design specification. A similar analysis of the piping shows that it complies with the allowable stress limits of USAS B31.7.

As a further assurance of system integrity, the completed reactor coolant system will be hydrotested at 3,125 psig before initial operation.

The active components in the reactor coolant system which are classified as Class I mechanical components consist of the reactor coolant pumps with motors and the control rod drive mechanisms. These components are modeled in the seismic analysis of the reactor coolant system and the appropriate response spectra imposed at the support points of the major contributing components of the system. The resulting seismic displacements and accelerations of mass points are used to check bearing loadings and shaft deflections.

As a pump-motor shaft is designed to have a natural frequency at least 20 percent above the critical speed, the shaft is too stiff to respond to any of the lower seismic frequencies. The pump and motor bearings are designed to be capable of meeting the seismic design criteria.

The design specification for the control rod drives requires that the drives be capable of withstanding the seismic loadings within the stress limits for
5. Class I equipment.

The purchase specifications for the ECCS pumps and valves require that the units be capable of operating under the seismic loads predicted to exist at the building elevations where the units will be located. The equipment supplier has certified that the units, based on tests which exceeded the specification requirements on similar units, do adequately meet the purchase specification requirements for operation under seismic loads. The instrumentation transmitters are tested to demonstrate their suitability for the specified seismic conditions.

The center of gravity for this type of equipment is low and both the pump and the driver are rigidly connected to a structural baseplate which in turn is bolted to the building. This type of equipment is structurally quite rigid and in most instances will accommodate very high "g" loadings.

4.3.7 OVERPRESSURE PROTECTION

The reactor coolant system is protected against overpressure by the pressurizer code safety valves mounted on top of the pressurizer. The capacity of these valves is determined from considerations of: (1) the Reactor Protective System; (2) pressure drop (static and dynamic) between the point of highest pressure in the reactor coolant system and the pressurizer; and (3) accident or transient overpressure conditions.

The combined capacity of the pressurizer code safety valves is based on the hypothetical case of withdrawal of a regulating control rod assembly bank from a relatively low initial power. The accident is terminated by high pressure reactor trip with resulting turbine trip. This accident condition produces a power mismatch between the reactor coolant system and secondary system larger than that caused by a turbine trip without immediate reactor trip, or by a partial load rejection from full load.

4.3.8 SYSTEM INCIDENT POTENTIAL

Potential accidents and their effects and consequences as a result of component or control failures are analyzed and discussed in Section 14.

The pressurizer spray line contains an electric motor-operated backup valve which can be closed should the pressurizer spray valve malfunction and fail to close; this would prevent depressurization of the system to the saturation pressure of the reactor coolant. An electric motor-operated valve located between the pressurizer and the pressurizer electromatic relief valve can be closed to prevent pressurizer steam blowdown in the unlikely event the electromatic relief valve fails to reclose after being actuated. Because of the other protective features in the plant, it is unlikely that the code valves will ever lift during operation. In addition, it is extremely unlikely these valves would stick open, since there is adequate experience to indicate the reliability of code safety valves. The analysis in Section 14.2 indicates that one high pressure injection pump is sufficient to protect the core for an opening in the system considerably larger than one pressurizer code safety valve in the open position.

4.3.9 REDUNDANCY

Each heat transport loop of the reactor coolant system contains one steam generator and two reactor coolant pumps. Operation at reduced reactor power is possible with one or more pumps out of service. For added reliability, power to each pump is normally supplied by one of two electrically separated buses as shown in Figure 8-2. Each of the two pumps per loop is fed from separate buses.

Two core flooding nozzles are located on opposite sides of the reactor vessel to ensure core reflooding water in the event of a single nozzle failure. Reflooding water is available from either the core flooding tanks or the low pressure injection pumps. The high pressure injection lines are connected to the reactor coolant system on each of the four reactor coolant inlet pipes.

4.3.10 SAFETY LIMITS AND CONDITIONS

4.3.10.1 Maximum Pressure

The reactor coolant system serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the reactor building atmosphere. In the event of a fuel cladding failure, the reactor coolant system is the primary barrier against the release of fission products to the reactor building. The safety limit of 2,750 psig (110% of design pressure) has been established. This represents the maximum transient pressure allowable in the reactor coolant system under the ASME Code, Section III.

4.3.10.2 Maximum Reactor Coolant Activity

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the reactor coolant constitutes a hazard only if the amount of activity is excessive and it is released to the environment. The plant systems are designed for operation with activity in the reactor coolant systems resulting from 1 per cent defective fuel. Activity would be released to the environment if the reactor coolant containing gaseous activity were to leak to the steam side of the steam generator. Gaseous activity could then be released to the environment by the steam jet air ejector on the main condenser. In 10 CFR 20, maximum permissible concentrations (MPC's) for continuous exposure to gaseous activity have been established. These MPC's will be used as the basis for maximum release of activity to the environment which has unrestricted access. Section 11 presents analysis of allowable reactor coolant activities with a 1 gpm tube leak.

4.3.10.3 Leakage

Reactor coolant system leakage rate is determined by comparing instrument indications of reactor coolant average temperature, pressurizer water level and letdown storage tank water level over a time interval. All of these indications are recorded. The letdown storage tank capacity is 31 gallons per each inch of height and each graduatuion on the level recorder represents 2 inches of tank height.

Reactor coolant system leak detection is provided by monitoring the reactor building sump level and the letdown storage tank level. Since the pressurizer level controller maintains a constant pressurizer level, any reactor coolant system volume change due to leakage would manifest itself as a Reactor Building sump level change and/or a corresponding letdown storage tank level change. Considering the most adverse initial conditions of a low level in the Reactor Building sump and a high level in the letdown storage tank, a 1 gpm leak from the reactor coolant system would initiate a Reactor Building sump
4. high level alarm indication in the control room within 5 hours and a letdown storage tank low level alarm indication in the control room within 16 hours. A 3 gpm leak would be detected in 1/3 the time given above for detection of a 1 gpm leak. Normally, with the Reactor Building sump level and the letdown storage tank level between their high alarm and the low alarm respectively, these detection times would be reduced.

If the leak allows primary coolant into the containment atmosphere, additional leak detection is provided by the Reactor Building gaseous monitoring system and the Reactor Building area monitoring system. The sensitivity and time for detection of a coolant system leak by any of the radioactivity monitoring systems depends upon reactor coolant activity and the location of the leak. Alarm indication for each sample point in these systems is in the control room.

If the leak is in a steam generator, the leak will be detected by a decrease in the level of the letdown storage tank as described above and also by main steam line and condenser air ejector off gas radiation monitors. The sensitivity of the radiation monitors for leak detection depends upon the activity of the primary coolant system.

Class I fluid systems other than the reactor coolant system pressure boundary will be monitored for leakage by monitoring the various storage and/or surge tanks for the applicable systems. The radiation monitoring system for the station will aid in leak detection of systems containing radioactive fluids. In addition to the above, routine Operator and/or Health Physics radiation surveillance will detect leakage in both radioactive and non-radioactive systems.

Natural circulation can be maintained in the reactor coolant system for decay heat removal following a complete loss of station power even if the system has been operating with an equipment leak. The natural circulation path will be maintained solid with water until the pressurizer has emptied, which is 6,000 gallons of coolant. A 30 gpm leakage rate in conjunction with a complete loss of station power and subsequent cooldown of the reactor coolant system by the turbine bypass system (set at 1,040 psia) and steam driven emergency feedwater pump would require a minimum of 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. Sixty minutes is ample time to restore electrical power to the plant and makeup flow to the reactor coolant system.

4.3.10.4 System Minimum Operational Components

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. One steam generator is required to be operable prior to criticality as the steam generator is the means for normal decay
9. heat removal at temperatures above 250 F. (Reference Supplement 6 revisions for Oconee 1.)

A reactor coolant pump or low pressure injection pump is required to be in operation prior to reducing boron concentration by dilution with make-up water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor.

4.3.10.5 Combined Heatup, Cooldown, and Pressure Limitations

The stress level of the material in a reactor vessel, or any other component of the coolant system is a combination of stresses caused by internal pressures and temperature gradients. The maximum steady-state stress resulting from gamma heating in the vessel is a relatively low value, and no problems are anticipated from thermal stresses in the reactor vessel wall. The initial Nil-Ductility Transition Temperature (NDTT) for all reactor vessel material is based on Charpy V-notch tests. The Design Transition Temperature (DTT) is defined as NDTT + 60 F. NDTT, and subsequently DTT, increases as a function of cumulative neutron exposure. To prevent brittle fracture during operation, stress levels will be restricted to the following levels: (1) at the temperature of DTT, the maximum stress permissible is 20 per cent of the yield strength; and (2) this stress limit tapers off to 10 per cent of design yield strength for DTT - 200 F, and remains at 10 per cent below DTT - 200 F. Curves in the plant operating manual define the operating limitation of pressure versus temperature to maintain stresses within the above levels. The predicted DTT shift due to irradiation exposure will be monitored by the surveillance program testing. The operating limit curves will be revised if required by the results of the surveillance program testing.

4.3.11 QUALITY ASSURANCE

Assurance that the reactor coolant system will meet its design bases insofar as the integrity of the pressure boundary is concerned, is obtained by analysis, inspection, and testing.

4.3.11.1 Stress Analyses

Detailed stress analyses of the individual reactor coolant system components including the vessel, piping, pumps, steam generators, and pressurizer have been performed for the Design Bases.

Dynamic analyses have been performed on the complete system treating each steam generator and associated coolant piping as an independent system to include the effect of the design bases earthquake or the maximum hypothetical earthquake in the piping stresses and nozzle stresses.

Independent thermal and dynamic analyses have been performed to insure that piping connecting to the reactor coolant system is of the proper schedule and that it does not impose forces on the nozzles greater than allowable. Small nozzles are conservatively designed and utilize ASA schedule 160.

The reactor coolant pump casing has been completely analyzed including a dynamic analysis separately from the loop to insure that the stresses throughout the casing are below the allowable for all design conditions.

Stress analysis reports required by codes for the several components have been prepared by the manufacturer and reviewed for adequacy by a separate organization.

4.3.11.2 Shop Inspection

Inspection and non-destructive testing of materials prior to and during manufacturing in accordance with applicable codes and additional requirements imposed by the manufacturer have been carried out for all of the reactor coolant system components and piping. The extent of these inspections and testing is listed in Table 4-12 for each of the components in the system. Shop testing culminates with a hydrostatic test of each component followed by magnetic particle inspection of the component external surface. (Piping will be hydrostatically tested in the field and will undergo the final inspection described in 4.3.11.3)

Components are cleaned, packaged to prevent contamination, and shipped over a pre-selected route to the site. For materials purchased or manufactured outside of B&W, the results of the material inspection and testing program have been observed or audited by B&W and audited by the applicant. In addition there is an independent audit by B&W's Nuclear Power Generation Department Quality Assurance Section.

4.3.11.3 Field Inspection

Field welding of reactor coolant piping and piping connecting to nozzles is performed using procedures which will result in weld quality equal to that obtained in shop welding. Non-destructive testing of the welds is identical to that performed on similar welds in the shop and is shown in Table 4-12. Accessible shop and field welds and weld repairs in the reactor coolant piping are inspected by magnetic particle or liquid penetrant tests following the system hydrostatic test.

4.3.11.4 Testing

The reactor coolant system including the reactor coolant pump internals, reactor closure head, control rod drives, and associated piping out to the first stop valve undergoes a hydrostatic test following completion of assembly. The hydrostatic test is conducted at a temperature 60 F greater than the highest nil-ductility temperature. During the hydrostatic test, a careful examination is made of all pressure boundary surfaces including gasketed joints.

4.4 TESTS AND INSPECTIONS

4.4.1 GENERAL

This section discusses tests and inspections performed during and after the assembly of the individual components into a completed reactor coolant system. These tests and inspections are performed to demonstrate the functional capabilities of the components after assembly into a completed system, to inspect the quality of the system closure weldments, and to monitor system integrity during service.

4.4.2 CONSTRUCTION INSPECTION

The coolant piping for each loop is shipped to the field in six subassemblies. The loops are then assembled in the field. In order to accommodate the small fabricating and field installation tolerances, a number of the subassemblies are fabricated with excess length. Thus, the final fitting of the coolant piping is accomplished in the field. The ends with excess length are field machined. All carbon steel-to-carbon steel field welds are back-clad with stainless steel following removal of the backing rings. Consumable inserts are used in stainless-to-stainless welds, such as surge line and some coolant pump welds. All welding is inspected in accordance with requirements of the applicable codes or better.

Welding of the auxiliary piping to reactor coolant system nozzles is done to the same standards as the main coolant piping. Consumable inserts are used in all cases.

Cleaning of reactor coolant piping and equipment is accomplished both before and after erection of various equipment. Piping and equipment nozzles will require cleaning in the area of the connecting weldments. Most of the piping and equipment are large enough for personnel entry and are cleaned by locally applying solvents and demineralized water and by wire brush to remove trapped
16. foreign particles. Where surfaces and equipment cannot be reached by personnel entry and have been cleaned in vendor shops to the required cleanliness for operation and appropriately protected to maintain cleanliness during handling, shipping, storage, and installation, further cleaning will not be performed. Appropriate checks to verify maintenance of required cleanliness will be performed prior to operation.

4.4.3 INSTALLATION TESTING

The Reactor Coolant System will be hydrostatically tested in accordance with USAS B31.7, Nuclear Power Piping Code. The test pressure will affect all parts of the Reactor Coolant System up to and including means of isolation from auxiliary systems, such as valves and blank flanges. The hydrostatic test will be performed at temperature above Design Transition Temperature.

The Reactor Coolant System relief valves will be inspected and shop-tested in accordance with Section III of the ASME code for Nuclear Vessels. The relief pressure setting will be made during the shop test.

4.4.4 FUNCTIONAL TESTING

Prior to initial fuel loading, the functional capabilities of the Reactor Coolant System components will be demonstrated at operating pressures and temperatures. Measurement of pressures, flows, and temperatures will be recorded for various system conditions. Operation of reactor coolant pumps, pressurizer heaters, pressurizer spray system, control rod drive mechanism, and other Reactor Coolant System equipment will be demonstrated. For descriptions of the various functional tests to be performed, refer to Section 13.1.

4.4.5 IN-SERVICE INSPECTION

Consideration has been given to the inspectability of the reactor coolant system in the design of components, in the equipment layout, and in the support structures to permit access for the purposes of inspection. Access for inspection is defined to be access for examination by direct or remote means during shutdown. See Appendix 4B for details of the In-Service Inspection Program.

4.4.6 MATERIAL IRRADIATION SURVEILLANCE

Surveillance specimens of the reactor vessel shell section material are installed in the reactor vessel in accordance with ASTM Specification E 185, Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors. Tensile specimens and Type A Charpy-V notch specimens which have been machined
7. from welds, heat-affected zones and base material will be contained in irradiation capsules in the reactor vessel. There are approximately three surveillance specimen holder tubes installed between the core and the inside wall of the reactor vessel shell. Refer to BAW-10006, "Reactor Vessel Material Surveillance Program," for a complete description of the surveillance program.

4.5 REFERENCES

(1) Porse, L., Reactor Vessel Design Considering Radiation Effects, ASME Paper No. 63-WA-100.

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(3) Robertson, T. S., Propagation of Brittle Fracture in Steel, Journal of Iron and Steel Institute, Volume 175, December 1953.

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(5) Hjarne, L., and Leimdorfer, M., "A Method for Predicting the Penetration and Slowing Down of Neutrons in Reactor Shields," Nuclear Science and Engineering 24, pp 165-174, 1966.

(6) Cadwell, *et al.*, The PDQ-5 and PDQ-6 Programs for the Solution of the Two-Dimensional Neutron Diffusion-Depletion Problem, WAPD-TM-477, January 1965.

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(8) Avery, A. F., The Prediction of Neutron Attenuation in Iron-Water Shields, AEEW-R 125, April 1962.

(9) Clark, R. H., and Baldwin, M. N., Physics Verification Program, Part II, BAW-3647-4, June 1967.

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	BAW-10006, Rev. 2	Reactor Vessel Material Surveillance Program
	BAW-10008, Part 1	Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Accident
21.	BAW-10018	Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock
	BAW-10027	Once-Through Steam Generator Research and Development (Nonproprietary version of BAW-10002, and BAW-10002, Sup. 1)
	BAW-10035	Fuel Assembly Stress and Deflection Due to Loss-of-Coolant Accident and Seismic Excitation (Nonproprietary version of BAW-10008, Part 2, Rev. 1)
23.	BAW-10051	Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations

Table 4-1
Reactor Coolant System Pressure Settings

		Pressure, psig	Capacity, lb/hr, total
	Design Pressure	2500	
	Pressurizer Code Safety Valves	2500	667,000
9.	High Pressure Trip	2355[a]	
	Pressurizer Electromatic Relief Valve		
	Open	2255[a]	107,000
	Close	2220[a]	
	High Pressure Alarm	2255[a]	
9.	Pressurizer Spray Valve		
	Open	2205[a]	
	Close	2155[a]	
	Operating Pressure[a]	2155	
	Low Pressure Alarm	2055	
9.	Low Low Pressure Alarm	1920[a]	
	Low Pressure Trip	1800[a]	
	Hydrotest Pressure	3125	

[a] At sensing nozzle on reactor outlet pipe.

Table 4-2
Reactor Coolant System Component Codes

	Component	Codes	Addendum
30.	Reactor Vessel	ASME III Class A	Summer 1967*
	Pressurizer	ASME III Class A	Summer 1967*
	Reactor Coolant System Piping	USAS B31.7	Errata through June 1968
	Feedwater Header	USAS B31.1	1967
	R.C. Pump Casings	ASME III Class A (not code stamped)	Summer 1967
	Safety and Relief Valves	ASME III Art. 9	Summer 1967
	Welding Qualifications	ASME III and IX	Summer 1967
	Steam Generator (primary and secondary sides)	ASME III Class A	Summer 1967*

30. *Welded joints tested in accordance with requirements of Article 7, Summer 1966 Addenda.

Table 4-3
Reactor Vessel Design Data

Design Pressure, psig	2500
Design Temperature, F	650
Coolant Operating Temperature, Inlet/Outlet, F	554/604
Hydrotest Pressure, psig	3125
Coolant Volume (Hot, Core and Internals in Place), ft^3	4058
Reactor Coolant Flow, lb/hr	131.32×10^6
Number of Reactor Closure Head Studs	60
Diameter of Reactor Closure Head Studs, in.	6-1/2
Vessel Dimensions	
Overall Height of Vessel and Closure Head, ft-in.[a]	40-8-3/4
Shell ID, in.	171
Flange ID, in.	165
Straight Shell Minimum Thickness, in.	8-7/16
Shell Cladding Miminum Thickness, in.	1/8
Shell Cladding Nominal Thickness, in.	3/16
Insulation Thickness, in.	3
Closure Head Minimum Thickness, in.	6-5/8
Lower Head Minimum Thickness, in.	5

Vessel Nozzles

Function	No.	ID, in.	Material
Coolant Inlet	4	28	Carbon Steel - SS Clad
Coolant Outlet	2	36	Carbon Steel - SS Clad
Core Flooding - LP Injection	2	14 Sch 40	Carbon Steel[b] - SS Clad
Control Rod Drive	61	2.76	Inconel[c]
Axial Power Shaping Rod Drive	8	2.76	Inconel[c]
Instrumentation	8	3/4 Sch 160	Inconel[c]
In-Core Instrumentation	52	3/4 Sch 160	Inconel

Dry Weight, lbs	
Vessel	646,000
Closure Head	158,300
Studs, Nuts, and Washers	39,500

[a]Instrument nozzle to CRD flange.

[b]With stainless steel safe end added after stress relief.

[c]With stainless steel flanges.

Table 4-4
Steam Generator Design Data

(Data per Steam Generator)

Steam Conditions at Full Load, Outlet Nozzles	
Steam Flow, lb/hr	5.6×10^6
Steam Temperature, F	570
Steam Pressure, psig	910
Feedwater Temperature, F	460
Reactor Coolant Flow, lb/hr	65.66×10^6
Reactor Coolant Side	
Design Pressure, psig	2500
Design Temperature, F	650
Hydrotest Pressure, psig	3125
Coolant Volume (Hot), ft^3	2030
Full Load Temperature Inlet/Outlet, F	604/554
Secondary Side	
Design Pressure, psig	1050
Design Temperature, F	600
Hydrotest Pressure, psig	1315
Net Volume, ft^3	3412
Mass of Steam and Water at Full Load, lb	55,000
Energy Content of Steam and Water at Full Load, Btu	32.0×10^6
Dimensions	
Tubes, OD/min Wall, in.	0.625/0.034
Overall Height (Including Skirt), ft-in.	73-2-1/2
Shell OD, in.	151-1/8
Shell Minimum Thickness, in.	4.1875
Shell Minimum Thickness (at Tube Sheets and F.W. Connect), in.	6.625
Tube Sheet Thicknesses, in.	24
Dry Weight, lb	1,140,000
Tube length, ft-in.	52-1-3/8

Table 4-4 (Cont'd)

Nozzles - Reactor Coolant Side

Function	No.	ID, in.	Material
Inlet	1	36	Carbon Steel - SS Clad
Outlet	2	28	Carbon Steel - SS Clad
Drain	1	1 Sch 160	Inconel
Manways	2	16	Carbon Steel - SS Clad
Handholes	2	5	Carbon Steel - SS Clad

16. Nozzles - Secondary Side (Reference Supplement 9 for revisions to Oconee 3)

Function	No.	ID, in.	Material
Steam	2	24	Carbon Steel
Vent	1	1-1/2 Sch 80	Carbon Steel
Drains	6	1-1/2 Sch 80	Carbon Steel
Drain	2	1 Sch 80	Carbon Steel
Level Sensing	8	1 Sch 80	Carbon Steel
Temperature Well	3	3/8	Inconel
Manways	2	16	Carbon Steel
Feedwater Connect	32	3 Sch 80	Carbon Steel
Auxiliary Feedwater Connect	7	3 Sch 80	Carbon Steel
Handholes	15	5	Carbon Steel

Table 4-5

Pressurizer Design Data

Design/Operating Pressure, psig	2500/2166
Design/Operating Temperature, F	670/648
Steam Volume, ft^3	700
Water Volume, ft^3	800
16. 3. Hydrotest Pressure, psig (c)	3125
Electric Heater Capacity, kW	1638
Dimensions	
Overall Height, ft-in.	44-11-3/4
Shell OD, in.	96-3/8
Shell Minimum Thickness, in.	6.188
Dry Weight, lb	291,000

Nozzles

Function	No.	ID, in.	Material
Surge Line	1	10 Sch 140	Carbon Steel - SS Clad[a]
Spray Line	1	4 Sch 120	Carbon Steel - SS Clad[b]
Relief Valve	3	2-1/2	Carbon Steel - SS Clad[a]
Vent	1	1 Sch 160	Inconel
Sample	1	1 Sch 160	Carbon Steel - SS Clad[b]
Temperature Well	1	3/8	Inconel
Level Sensing	6	1 Sch 160	Carbon Steel - SS Clad[b]
Heater Bundle	3	19-1/8	Carbon Steel - SS Clad
Manway	1	16	Carbon Steel - SS Clad

(a) With stainless steel safe end added after stress relief.

(b) With Inconel safe end.

16. (c) Pressure retaining part (inlet bushing) of pressurizer relief valves shop hydrotested at 3750 psig.

Table 4-6
Reactor Coolant Piping Design Data

Reactor Inlet Piping	
Pipe ID, in.	28
Design Pressure/Temperature, psig/F	2500/650
Hydrotest Pressure, psig	3125
Minimum Thickness, in.	2-1/4
Coolant Volume (Hot - System Total), ft^3	1085
Dry Weight, System Total, lb	214,000
Reactor Outlet Piping	
Pipe ID, in.	36
Design Pressure/Temperature, psig/F	2500/650
Hydrotest Pressure, psig	3125
Minimum Thickness, in.	2-7/8
Coolant Volume (Hot - System Total), ft^3	979
Dry Weight, System Total, lb	200,000
Pressurizer Surge Piping	
Pipe Size, in.	10, Schd 140
Design Pressure/Temperature, psig/F	2500/670
Hydrotest Pressure, psig	3125
Coolant Volume, hot, ft^3	20
Dry Weight, lb	5000
Pressurizer Spray Piping	
Pipe Size, in.	2-1/2, Schd 160
Design Pressure/Temperature, psig/F	2500/650 & 670
Hydrotest Pressure, psig	3125
Coolant Volume, hot, ft^3	2
Dry Weight, lb	650

Nozzles:

	Function	No.	ID, in.	Material
	On Reactor Inlet Piping			
	High Pressure Injection	4	2-1/2 Sch 160	(a)
	Pressurizer Spray	1	2-1/2 Sch 160	Stainless Steel
	Drain/Letdown	1	2-1/2 Sch 160	(b)
16.	Drain	3	1-1/2 Sch 160	(c)
	Pressure Sensing	4	1 Sch 160	(c)
	Temperature Well	4	0.375	Inconel
	Temperature Sensing	4	0.613	Inconel

Table 4-6 (Cont'd)

Function	No.	ID, in.	Material
On Reactor Outlet Piping			
Decay Heat	1	12 Sch 140	(b)
Vent	2	1 Sch 160	(c)
Conn. on Flow Meters	4	1 Sch 160	(c)
Pressure Sensing	4	1 Sch 160	(c)
Temperature Well	2	3/8	Inconel
Temperature Sensing	6	0.613	Inconel
Surge Line	1	10 Sch 140	(b)
On Pressurizer Surge Piping			
Drain	1	1 Sch 160	Stainless Steel
16. On Pressurizer Spray Piping			
Auxiliary Spray	1	1-1/2 Sch 160	Stainless Steel
Spray Valve Bypass	2	1/2 Sch 160	Stainless Steel

(a) Carbon Steel - SS Clad - With Stainless Steel Safe End Added after Stress Relief

(b) Carbon Steel - SS Clad - with Inconel Safe End

(c) For Oconee 1 & 2:
Carbon Steel - SS Clad - with Inconel Safe End

For Oconee 3:
Solid Inconel

(Reference Supplement 6 Revision for Oconee 1.)

Table 4-7
<u>Reactor Coolant Pump - Design Data</u>

(Data per Pump)

	Design Pressure/Temperature, psig/F	2500/650
	Hydrotest Pressure, psig	3750
	RPM at Nameplate Rating	1190
	Developed Head, ft	396
	Capacity, gpm	88,000
16.	Seal Water Injection, gpm	8
	Seal Water Return, gpm	1
	Injection Water Temperature, F	125
	Cooling Water Temperature, F	105
9.	Pump Discharge Nozzle ID, in.	28
	Pump Suction Nozzle ID, in.	28
	Overall Height (Pump-Motor), ft-in.	29-4
	Dry Weight Without Motor, lb	108,300
	Coolant Volume, ft^3	98
	Pump-Motor Moment of Inertia, lb-ft^2	70,000
	Motor Data	
	Type	Squirrel Cage Induction Single Speed, Water Cooled
	Voltage	6600
	Phase	3
	Frequency, Hz	60
	Insulation Class	F
	Starting Current, amp	4350
	Power, HP (Nameplate)	9000

Table 4-8
Transient Cycles

	Transient Description	Design Cycles	ASME Category	Load* Case
3.	1. Heatup, 70 F to 557 F and Cooldown, 557 F to 70 F at 100 F/h	240	Normal	I
	2. Heatup, 532 F to 579 F (0 to 15% FP) and Cooldown, 579 F to 532 F (15 to 0%) at 5% of FP per Minute	1,440	Normal	I
	3. Plant Loading & Unloading (10% of FP per Minute Between 8 and 100% FP)	48,000	Normal	I
	4. Step Loading Increase and Decrease of 10% FP	8,000	Normal	I
1.	5. Step Load Reduction From FP to Auxiliary Load	310	Upset	I
	6. Reactor Trip From FP	400	Upset	I
	7. Rapid Depressurization (2200 psig to 300 psig in One Hour)	80	Emergency	II
	8. Rod Withdrawal Accidents	40	Upset	I
1.	9. Drop of One Control Rod	40	Upset	I
	10. Hydrotests at 3125 psig	20	Test	I

In addition to the system transients described above, the steam generator is also designed to withstand the effects of cyclic loads from the following transients:

	Transient Description	No. of Cycles		
1.	Loss of Feedwater Flow	80	Upset	I
	Loss of Station Power	40	Upset	I

Note: These transients are based on 40-year design life for equipment design purposes are are not intended to be actual transients or operating procedures.

*Note: For loading combinations and allowable stress limits see Pages 4-3 and 4-4.

Rev. 1. 9/15/69
Rev. 3. 3/16/70
Rev. 21. 7/26/72
(Entire Page Revised)

Table 4-9
Materials of Construction

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-533, Grade B, Class 1[a]
	Pressure Forgings	A-508-64, Class 2 (Code Case 1332-3)
	Cladding	18-8 Stainless Steel or Ni-Cr-Fe
	Studs, Nuts and Washers	A-540, Grade B23 (Code Case 1335-2)
	Thermal Shield and Internals	SA-240, Type 304
	Guide Lugs	Ni-Cr-Fe, SB-168 (Code Case 1336)
Steam Generator	Pressure Plate	SA-212, Grade B; SA-533, Grade B (Code Case 1339)
	Pressure Forgings	A-508-64, Classes 1 and 2 (Code Case 1332-3)
	Cladding for Heads	18-8 Stainless Steel
	Cladding for Tube Sheets	Ni-Cr-Fe
	Tubes	Ni-Cr-Fe, SB-163
	Studs - Reactor Coolant Side	SA-320, Grade L43
	Nuts - Reactor Coolant Side	SA-194, Grade 4
	Studs - Secondary Side	SA-193, B14
	Nuts - Secondary Side	SA-194-2H
Pressurizer	Shell, Heads, and External Plate	SA-212, Grade B
	Forgings	A-508-64, Class 1 (Code Case 1332-3)
	Cladding	18-8 Stainless Steel
	Studs and Nuts	SA-320, Grade L43
	Internal Plate	SA-240, Type 304
	Internal Piping	SA-312, Type 304

[a] This material is metallurgically identical to SA-302, Grade B, as modified by Code Case 1339.

Table 4-9 (Cont'd)

Component	Section	Materials
Reactor Coolant Piping	28 in. and 36 in.	SA-516, Grade 70 (Elbows) A-106, Grade C (Straights)
	Cladding	18-8 Stainless Steel
	10 in. Surge Line and 2-1/2 in. Spray Line	A-403, Grade WP 316 (Elbows) A-376, Type 316 (Straights)
	Piping Safe Ends	A-376, Type 316 and Ni-Cr-Fe, SB-166
Reactor Coolant Pumps (Reference Supplement 6 revision for Oconee 1.)	Castings	
	Casing	A-351, Grade CF8M
	Stuffing Box	A-351, Grade CF8M
	Forgings	
	Shaft	A-473, Type 316
9.	Bolting	
	Casing Studs	A-193, Grade B7
	Casing Nuts	A-194, Type 2H
Valves	Pressure Containing Parts	A-351, Grade CF8M A-182, F316 and F347

Table 4-10
Reactor Coolant Quality Specification

Total Solids, ppm, max	1.0 (including dissolved and undissolved but excluding LiOH and H_3BO_3)
Boric Acid, ppm	0 - 13,000
Lithium as ^7Li, ppm	0.5 - 2.0 (equivalent range as ^7LiOH is 1.455 to 5.82 ppm)
pH at 77 F	4.8 - 8.5 (equivalent pH at 600 F is 6.8 to 7.8)
Dissolved Oxygen as O_2	Not Applicable (with proper H_2 specification at critical condition, dissolved O_2 is assumed not to be present)
Chlorides as Cl^-, ppm, max	0.1
Hydrogen as H_2, std cc/liter H_2O	15 - 40
Fluorides as F^-, ppm, max	0.1

Table 4-11
Steam Generator Feedwater Quality Specification

Total Solids (dissolved and suspended), ppb, max	50
Dissolved Oxygen, ppb, max	7
Total Silica (as SiO_2), ppb, max	20
Total Iron (as Fe), ppb, max	10
Total Copper (as Cu), ppb, max	2
Nominal pH at 77 F (adjusted with ammonia)	9.3 - 9.5

Table 4-12
Fabrication Inspections

Component	RT	UT	PT	MT	ET
1. Reactor Vessel					
1.1 Forgings					
1.1.1 Flanges		X[a]		X	
1.1.2 Studs, Bar		X			
1.1.3 Studs After Final Machining				X	
1.1.4 Skirt Adaptor		X[a]			
1.1.5 Nozzle Shell Forgings		X		X	
1.1.6 Main Nozzle Forgings		X		X	
1.1.7 Dutchman Forging		X[a]		X	
1.1.8 CRD Mechanism Adaptor		X	X		
1.1.9 CRD Mechanism Housing		X	X		
1.2 Plates					
1.2.1 Head and Shell Plate		X[a]		X[+]	
1.2.2 Support Skirt		X[a]		X[+]	
1.3 Instrumentation Tubes		X	X		
1.4 Closure O-Rings		X	X		
1.5 Weldments					
1.5.1 Longitudinal and Circumferential Main Seams	X			X	
1.5.2 CRD Mechanism Adaptor to Shell	X		X		
1.5.3 CRD Mechanism Adaptor to Flange	X		X		
1.5.4 Main Nozzles	X			X	
1.5.5 Instrumentation Nozzle Connection			X		
1.5.6 Nozzle Safe-Ends, Weld Deposit		X	X		
1.5.7 Temporary Attachment After Removal				X	
1.5.8 All Accessible Welds After Hydrotest				X	
1.5.9 O-Ring Closure Weld	X		X		

Table 4-12 (Cont'd)

Component	RT	UT	PT	MT	ET
1.5.10 Cladding, Sealing Surfaces		X(b)(+)	X		
1.5.11 Cladding, All Other		X(c)(+)	X		
1.5.12 Insulation Support Lugs				X	
2. Steam Generator					
2.1 Tube Sheet					
2.1.1 Forging		X(a)		X	
2.1.2 Cladding		X(b)(+)	X		
2.2 Heads					
2.2.1 Plate		X(a)		X(+)	
2.2.2 Cladding		X(c)(+)	X		
2.3 Shell					
2.3.1 Plates		X(a)		X	
2.4 Tubes		X	X(e)(+)		X(+)
2.5 Nozzles (Forgings)		X		X	
2.6 Studs, Bar		X			
2.7 Studs After Final Machining				X	
2.8 Weldments					
2.8.1 Shell, Longitudinal as Deposited by Submerged Arc	X			X	
2.8.2 Shell, Longitudinal as Deposited by Electroslag	X	X		X	
2.8.3 Shell, Circumferential	X			X	
2.8.4 Cladding, Sealing Surfaces		X(b)(+)	X		
2.8.5 Cladding, All Other		X(c)(+)	X		
2.8.6 Nozzle to Shell	X			X	
2.8.7 Level Sensing and Drain Connections	X			X	
2.8.8 Instrument Connections			X		
1.&3. 2.8.9 Support Skirt		X(+)		X	
2.8.10 Tube-to-Tube Sheet (d)			X		

Rev. 1. 9/15/69
Rev. 3. 3/16/70

Table 4-12 (Cont'd)

Component	RT	UT	PT	MT	ET
2.8.11 Temporary Attachment After Removal				X	
2.8.12 After Hydrostatic Test (All Accessible Welds)				X	
2.8.13 Lifting Lugs				X	
2.8.14 Insulation Support Lug Pads				X	
3. Pressurizer					
3.1 Heads					
3.1.1 Plate		X(a)		X	
3.1.2 Cladding		X(c)(+)	X		
3.2 Shell					
3.2.1 Forging		X(a)		X	
3.2.2 Plate		X(a)		X(+)	
3.2.3 Cladding		X(c)(+)	X		
3.3 Heater Bundles					
3.3.1 Cover Plate		X		X	
3.3.2 Diaphragm and Spacer Plate		X	X		
3.3.3 Studs, Bar		X			
3.3.4 Studs and Nuts After Final Machining				X	
3.3.5 Heaters					
3.3.5.1 Tubing		X	X(+)		
3.3.5.2 Positioning of Heater Element in Tube	X				
3.4 Nozzle (Forgings)		X		X	
3.5 Weldments					
3.5.1 Shell, Longitudinal as Deposited by Submerged Arc	X			X	
3.5.2 Shell, Longitudinal as Deposited by Electroslag	X	X		X	

Table 4-12 (Cont'd)

Component	RT	UT	PT	MT	ET
3.5.3 Shell, Circumferential	X			X	
3.5.4 Cladding, Sealing Surfaces		X(b)(+)	X		
3.5.5 Cladding, All Other		X(c)(+)	X		
3.5.6 Nozzle to Shell	X			X	
3.5.7 Nozzle Safe-Ends (If Weld Deposit)		X	X		
3.5.8 Nozzle Safe-End (If Forging or Bar)	X		X		
3.5.9 Instrumentation and Vent Connections			X		
3.5.10 Support Brackets				X	
3.5.11 Heater Guide Tube Pad		X	X		
3.5.12 Temporary Attachment After Removal				X	
3.5.13 All Accessible Welds After Hydrotest				X	
3.5.14 Insulation Support Pads				X	
4. Piping					
4.1 Pipe					
4.1.1 Forgings		X(a)		X	
4.1.2 Cladding		X(c)(+)	X		
4.2 Bends					
4.2.1 Plate		X(a)		X(+)	
4.2.2 Cladding		X(c)(+)	X		
4.3 Nozzle Forgings		X		X	
4.4 Weldments					
4.4.1 Longitudinal	X			X	
4.4.2 Circumferential	X			X	
4.4.3 Cladding, Elbows		X(c)(+)	X		
4.4.4 Cladding, Straight		X(c)(+)	X		
4.4.5 Nozzles to Run Pipe	X			X	

Table 4-12 (Cont'd)

Component	RT	UT	PT	MT	ET
4.4.6 Thermowell Connections			X		
4.4.7 Insulation Support Lug Pads				X	
5. Reactor Coolant Pumps					
5.1 Castings	X		X		
5.2 Forgings		X	X		
5.3 Weldments					
5.3.1 Circumferential	X		X		
5.3.2 Piping Connections			X		
6. Valves					
6.1 Castings	X		X		
6.2 Forgings		X	X		

[a] 100% scanning for longitudinal wave technique and 100% shear wave technique.

[b] UT of clad defects and bond to base metal.

[c] UT of clad bond to base metal (spot check).

[d] Also gas leak test--B&W requirement.

[e] Over 12-inch length on each end.

[+] Additional B&W requirement.

Note: RT: Radiographic
UT: Ultrasonic
PT: Dye Penetrant
MT: Magnetic Particle
ET: Eddy Current

Table 4-13
References for Figure 4-10 -- Increase in Transition Temperature Due to Irradiation Effects for A302B Steel

Ref. No.	Reference	Material	Type	Temp., F	Neutron Exposure, n/cm^2 (>1 Mev)	NDTT, F
1	ASME Paper No. 63-WA-100 (Figure 1)	All	Steels	Max Curve for 550		Data
2	ASTM-STP 380, p 295	A302B	Plate	Trend Curve for 550		Data
3	NRL Report 6160, p 12	A302B	Plate	550	5×10^{18}	65
4	ASTM-STP 341, p 226	A302B	Plate	550	8×10^{18}	85[a]
5	ASTM-STP 341, p 226	A302B	Plate	550	8×10^{18}	100
6	ASTM-STP 341, p 226	A302B	Plate	550	1.5×10^{19}	130[a]
7	ASTM-STP 341, p 226	A302B	Plate	550	1.5×10^{19}	140
8	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/1-31-65	A302B	Plate	550	3×10^{19}	120
9	Quarterly Report on Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/1-31-65	A302B	Plate	550	3×10^{19}	135

[a] Transverse specimens.

Table 4-13 (Cont'd)

Ref. No.	Reference	Material	Type	Temp., F	Neutron Exposure, n/cm^2 (>1 Mev)	NDTT, F
10	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/ 1-31-65	A302B	Plate	550	3×10^{19}	140
11	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/ 1-31-65	A302B	Plate	550	3×10^{19}	170
12	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/ 1-31-65	A302B	Plate	550	3×10^{19}	205
13	Welding Research Supplement, Vol. 27, No. 10, Oct. 1962, p 465-S	A302B	Weld	550 to 575	5×10^{18}	70
14	Welding Research Supplement, Vol. 27, No. 10, Oct. 1962, p 465-S	A302B	Weld	500 to 575	5×10^{18}	50
15	Welding Research Supplement, Vol. 27, No. 10, Oct. 1962, p 465-S	A302B	Weld	500 to 575	5×10^{18}	37
16	Welding Research Supplement, Vol. 27, No. 10, Oct. 1962, p 465-S	A302B	Weld	500 to 575	5×10^{18}	25

Table 4-14
Summary of Primary Plus Secondary Stress Intensity for Components of the Reactor Vessel

Area	Stress Intensity, psi	Allowable Stress 3 S_m, psi (Operating Temperature)
Control Rod Housing	24,800	69,900
Head Flange	58,000	80,000
Vessel Flange	43,000	80,000
Closure Studs	89,400	107,400
Primary Nozzles - Inlet	24,000	80,000
Outlet	24,000	80,000
Bottom Head to Shell	23,300	80,000
Bottom Instrumentation	10,100	69,900
Nozzle Belt to Shell	32,300	80,000
Core Flooding Nozzle	23,660	80,000
Support Skirt	88,000	93,700

4. Locations or points of stress analysis are illustrated on Figure 4-4c.

Table 4-15
Summary of Cumulative Fatigue Usage Factors For Components of the Reactor Vessel

Item	Usage Factor[a]
Control Rod Housing	0.0
Head Flange	0.10
Vessel Flange	0.05
Stud Bolts	0.38
Primary Nozzles - Inlet	0.06
Outlet	0.06
Bottom Head to Shell	0.0
Bottom Instrumentation	0.0
Nozzle Belt to Shell	0.0
Core Flooding Nozzle	0.02
Support Skirt	0.14

[a] As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

4. Points of stress analysis are illustrated on Figure 4-4c.

Table 4-16
Reactor Vessel -- Physical Properties

	Item	Heat No.	Ultimate Strength (10^3 psi)	Yield Strength (10^3 psi)	Elong. in 2 in. (%)	Impact Test Temp. (°F)	Impact Values
	Closure Head Center Disc	C 2311-2	90.3	69.5	31.0	+10	44-38-43
	Bottom Head	A 0973-2	87.2	65.0	24.5	+10	35-30-47
	Upper Shell Course	C 2197-2	91.5	70.0	25.0	+10	39-45-26
	Middle Shell Course	C 3265-1	87.0	66.2	28.1	+10	34-64-27
16.	Middle Shell Course	C 3278-1	84.5	63.5	28.1	+10	35-29-53
	Bottom Shell Course	C 2800-1	85.0	60.5	29.0	+10	36-39-39
	Bottom Shell Course	C 2800-2	90.5	69.0	25.0	+20	32-33-49
	Core Flooding Nozzle	94894	98.0	74.0	21.5	+10	45-53-40
	Core Flooding Nozzle	94894	92.5	71.0	24.0	+10	37-50-45
	Inlet Nozzle	123S346VA1	90.0	67.5	25.0	+10	104-94-142
	Inlet Nozzle	123S346VA2	92.7	72.5	26.0	+10	104-121-106
	Inlet Nozzle	124S502VA1	97.2	76.0	25.0	+10	120-106-101
	Inlet Nozzle	124S502VA1	94.0	73.5	23.5	+10	110-85-77
	Outlet Nozzle	122S316VA2	90.0	67.0	26.0	+10	131-110-94
	Outlet Nozzle	122S316VA1	90.0	68.5	25.0	+10	92-86-82
	Upper Shell Flange	4P16373P1566	82.5	57.4	29.0	+10	49-41-71
	Head Transition Piece	122S347VAI	94.5	74.5	24.0	+10	92-70-70
	Closure Head Flange	125S535VAI	102.0	81.0	23.5	+10	59-47-70
	Closure Head Ring	99392D-2	96.5	75.5	26.0	+10	73-79-88
	Upper Nozzle Shell Course	ZV-2888	82.0	57.0	30.5	+34 avg	30 avg
	Lower Nozzle Shell Course	ZV-2861	85.0	63.5	29.0	+26 avg	30 avg

Table 4-17

Reactor Vessel -- Chemical Properties

Heat Number	Element C	Mn	P	S	Si	Ni	Mo	Co	V	Cr
C 2311-2	0.22	1.35	0.009	0.018	0.22	0.61	0.41	0.005	--	--
A 0973-2	.21	1.34	.011	.016	.18	.46	.47	.010	--	--
C 2197-2	.21	1.28	.008	.010	.17	.50	.46	.021	--	--
C 3765-1	.21	1.42	.015	.015	.23	.50	.49	.016	--	--
16. C 3278-1	.19	1.26	.010	.016	.23	.60	.47	.016	--	--
C 2800-1	.20	1.40	.012	.017	.20	.63	.50	.014	--	--
C 2800-2	.20	1.40	.012	.017	.20	.63	.50	.014	--	--
94894	.22	0.62	.006	.009	.23	.87	.60	.016	--	0.33
123S346VA1	.22	.61	.010	.010	.20	.69	.56	.01	0.01	.27
123S346VA2	.21	.62	.010	.008	.20	.69	.57	.01	.01	.28
124S502VA1	.22	.65	.010	.010	.22	.75	.59	.02	.01	.35
124S502VA2	.23	.68	.010	.014	.22	.78	.60	.02	.01	.31
122S316VA2	.20	.62	.010	.009	.28	.73	.57	.013	.01	.33
122S316VA1	.18	.58	.010	.014	.28	.68	.61	.015	.01	.32
4P1637 3P1566	.20	.72	.010	.012	.28	.74	.55	.011	.03	.34
122S347VA1	.20	.63	.010	.008	.25	.66	.55	.021	.02	.32
125S535VA1	.21	.63	.010	.011	.23	.72	.60	.010	< .02	.39
99392 D-2	.25	.72	.010	.025	.22	.78	.64	.010	--	.38
ZV-2888	.22	.74	.012	.010	.31	.71	.56	.007	.02	.36
ZV-2861	0.22	0.64	0.006	0.010	0.29	0.65	0.57	0.01	0.01	0.31

Table 4-18

Stresses Due to a Maximum Design Steam Generator Tube Sheet Pressure Differential of 2,500 psig at 650 F

Stress	Computed Value	Allowable Value
Primary Membrane	22,000 psi	37,200 psi (0.9 S_y)
Primary Membrane Plus Primary Bending	39,700 psi	55,900 psi (1.35 S_y)

Table 4-19

Ratio of Allowable Stresses to Computed Stresses for a Steam Generator Tube Sheet Pressure Differential of 2,500 psig

Component Part	Stress Ratio
Primary Head	4.02
Primary Head Tube Sheet Joint	4.02
Tubes	1.07
Tube Sheet	
Max Avg Ligament	1.02
Effective Ligament	1.70

Rev. 1. 9/15/69
Page No. Only

9.

Table 4-20. Pump Casings - Code Allowables
(Applies to Oconee Units 2 and 3)

Area		Governing Code III Para.	Condition	Allowable Stress or Stress Intensity
Nozzle Section (Outside reinforcement)	Extreme Fibers	N414.3	D	1.5 S_m = 25,050
			A+ (B+C)/2 + P	1.5 S_m = 25,050
		N417.7	A+ (B+C)/2 +D+P	3.0 S_m = 50,100
		N414.3	P	1.5 S_m = 25,050
		(1)	A+B+C+P 1.8	1.2x1.5 S_m = 30,060
		(1)	A+B+C+D+P 4.5Sm	1.2x3.0 S_m = 60,060
	℄ Fibers	N417.7	D	1.5 S_m = 25,050
			A+ (B+C)/2 +P	1.0 S_m = 16,700
			A+ (B+C)/2 +D+P	3.0 S_m = 50,100
		N414.1	P	1.0 S_m = 16,700
		(1)	A+B+C+P	1.2 S_m = 20,040
		(1)	A+B+C+D+P	1.2x3.0 S_m = 60,120
Nozzle Section (Inside Reinforcement)	Extreme Fibers	N414.3	A+ (B+C)/2 +D+P	1.5 S_m = 25,050
		(1)	A+B+C+D+P	1.2x1.5 S_m = 30,060
	℄ Fibers	N417.7	A+ (B+C)/2 +D+P	1.0 S_m = 16,700
		(1)	A+B+C+D+P	1.2 S_m = 20,040
Bowl Section	Ext. Fibers	N414.3	P	1.5 S_m = 25,050
	℄ Fibers	N414.1	P	1.0 S_m = 16,700
Cover	℄ Fibers	N712.1	Hydrostatic Press.	0.9 Y.S.= 27,000
	Ext. Fibers		Hydrostatic Press.	1.35 Y.S. = 40,500
	All Fibers	N412(m)(1)	Operating (thermal only)	2.0 Y.S. = Temp Depend.
		N414.4	Operating (thermal & press)	3.0 S'_m = 60,000
	℄ Fibers	N414.1	Operating (Pressure Only)	1.0 S'_m = 20,000
	Ext. Fibers	N414.3		1.5 S'_m = 30,000

Notes:
1. Reactor Coolant Piping reactions on pump
2. A = Dead Load Reactions
   B = Vertical seismic reactions (MHE)
   C = Horizontal seismic reactions (MHE)
   D = Thermal expansion reactions
   P = 2500 psia (operating design pressure)
3. Hydrostatic pressure = 3750 psi
4. S_m=16,700 psi for A351 CF8M, at 650°F

Rev. 5. 5/25/70
(New Page)
Rev. 9. 8/11/70

9. (Applies to Oconee Units 2 and 3)

Table 4-21. Summary of Maximum Stresses - Casing

Location	Fiber	Calculated Stress Intensity	Allowable Stress Intensity
Discharge Nozzle	℄	15,880	16,700
	Extreme	25,000	25,050
Suction Nozzle	℄	15,960	16,700
	Extreme	21,290	25,050
Upper Bowl Section	℄	4,814	16,700
	Extreme	23,200	25,050
Cylindrical Bowl Section	℄	9,535	16,700
	Extreme	11,067	25,050
Lower Bowl Section	℄	9,432	16,700
	Extreme	10,917	25,050

NOTE: Nozzle stress intensity based upon loading due to pressure, piping flexibility, piping dead load and operational basis earthquake loading as determined by a system dynamic analysis.

Table 4-22
Summary of Missile Equations

Missile Category	Principle	Symbolic Form of Equation	Solution
I	Stored strain energy equals kinetic energy	$\frac{\sigma\varepsilon v}{2} = \frac{mV_o^2}{2}$ $\frac{\sigma^2 v}{2E} = \frac{mV_o^2}{2g}$	$V_o = \sigma\sqrt{\frac{E_g}{\rho}}$
II	Work done is converted to kinetic energy	$F\ell = \frac{mV_o^2}{2}$ $= PA_o\ell$	$V_o = \sqrt{\frac{2PA_o\ell}{m}}$
III	Newton's second law	$F = ma$ $a = \dot{V} = \ddot{X} = \frac{F}{m}$ $\dot{V} = \left[\frac{\rho_f A_o V_f}{m}\right]\frac{A_m}{A_j}\left(V_f - V\right)$	$(1-\frac{V}{V_f}) - \ln(1-\frac{V}{V_f}) = K_1 - \frac{K_2}{r_o + X\tan\beta}$ $K_1 = (1-\frac{V_o}{V_f}) - \ln(1-\frac{V_o}{V_f}) + \frac{K_2}{r_o}$ $K_2 = \frac{\rho_f A_o A_m}{m\pi\tan\beta}$

NOTE: Either graphical techniques or numerical methods must be used to obtain the solution to category III.

Table 4-23

List of Symbols

σ	=	ultimate tensile stress, (lb/ft^2)
ε	=	strain = $\sigma/_E$, (in./in.)
E	=	modulus of elasticity, (lb/ft^2)
v	=	volume, (ft^3)
m	=	mass of the missile, ($lb\text{-}sec^2/ft$)
V	=	velocity of missile, (ft/sec)
g	=	gravity constant, (ft/sec^2)
F	=	force on the missile, (lb)
ℓ	=	stroke length, (ft)
P	=	system pressure, (lb/ft^2)
A_o	=	missile area under pressure, throat area, (ft^2)
P_f	=	density of fluid, ($\#/ft^3$)
V_f	=	jet velocity, (ft/sec)
A_m	=	projected area of missile, (ft^2)
A_j	=	jet area, (ft^2)
β	=	angle of jet expansion, (°from normal)
X	=	distance missile travels, (ft)
V_o	=	initial velocity of missile, (ft/sec)
r_o	=	radius of throat (ft)
K_2	=	constant

Table 4-24

Properties of Missiles - Reactor Vessel & Control Rod Drive

Missile Class		Description	Weight (lbs.)	Impact Area in^2	Velocity (ft/sec)	Kinetic Energy Ft-lbs
I	1.	Closure head nut	80	38	97	11,680
	2.	Closure stud w/nut	660	71	97	96,400
	3.	1" Valve bonnet stud	0.5	0.6	73.5	42
	4.	C.R. nozzle flange bolt & nut	3.0	3.1	97	438
II	1.	CRD closure cap	8.0	7.0	215	5,742
III	1.	C.R. drive assembly	1000	64.0	90	125,777

Table 4-25

Properties of Missiles - Steam Generator

Missile Class		Description	Weight (lbs.)	Impact Area in^2	Velocity (ft/sec)	Kinetic Energy Ft-lbs
I	1.	1½" Vent valve bonnet stud	2.0	.8	73.5	167
	2.	Feedwater inlet flange bolt	0.3	.6	67.5	21
	3.	16" I.D. manway stud, tube side	8.0	2.1	67.5	566
	4.	5" Inspection opening cover stud	1.5	1.2	73.5	125
	5.	1" Valve bonnet stud	0.5	.6	73.5	42
II	1.	1½" Vent valve stem & wheel	5.0	.45	44.5	154
	2.	Sample line 1" valve stem & wheel	4.0	.3	35.8	80
	3.	Sample line 1" EMO valve stem and wheel	4.0	.3	35.8	80
III	1.	16" I.D. manway cover, tube side	955	615	515	1,950,000
	2.	16" I.D. manway cover, shell side	478	615	777	2,230,000
	3.	5" I.D. inspection cover, tube side	80	150	515	160,000
	4.	5" I.D. inspection cover, shell side	40	150	852	220,000
	5.	1½" Vent valve bonnet and assembly	24	38	371	51,180
	6.	Sample line 1" valve bonnet & assy.	30	27	243	27,460
	7.	Sample line, 1" EMO bonnet & assy.	115	27	138	34,250

Table 4-26

Properties of Missiles - Pressurizer

Missile Class		Description	Weight (lbs.)	Impact Area in^2	Velocity (ft/sec)	Kinetic Energy Ft-lbs
I	1.	4" Valve bonnet stud	3.0	1.8	73.5	250
	2.	5" Valve bonnet stud	3.0	2.4	73.5	250
	3.	16" Mayway cover stud	7.5	3.1	67.5	530
	4.	Heater bundle stud	25.0	7.0	73.5	2100
	5.	3/4" Valve stem stud	0.8	.45	73.5	67
II	1.	Spray line 4" EMO valve stem	9	1.0	135.0	2560
	2.	Sample line 3/4" valve stem	4	.3	72.7	330
	3.	Sample line 3/4" EMO valve stem	4	.3	72.7	330
III	1.	16" I.D. manway cover	250	615	375	546,000
	2.	Heater bundle assembly	2500	850	375	5,400,000
	3.	Spray line 4" EMO valve bonnet and assembly	325	150	521	1,370,000
	4.	$2\frac{1}{2}$" x 6" Relief valve bonnet and assembly	175	65	232	146,000
	5.	Sample line 3/4" valve bonnet and assembly	20	21	364	41,150
	6.	Sample line 3/4" EMO valve bonnet and assembly	115	21	258	118,400

Table 4-27

Properties of Missiles - Quench Tanks & Instruments

QUENCH TANKS

Missile Class		Description	Weight (lbs.)	Impact Area in^2	Velocity ft/sec	Kinetic Energy Ft-lbs
I	1.	1½" Drain valve bonnet stud	0.6	.2	73.5	50
	2.	4" Valve bonnet stud	2.0	.3	73.5	167
II	1.	1½" EMO drain valve stem	5.0	.45	11.0	9
	2.	4" EMO valve stem	9.0	1.0	21.5	65
III	1.	1½" EMO drain valve & op. assy.	220	20	73.5	18,450
	2.	1½" Drain valve bonnet & assy.	20	20	73.5	1,670
	3.	4" EMO valve bonnet & op. assy.	355	65	73.5	29,780

INSTRUMENTS

Missile Class		Description	Weight (lbs.)	Impact Area in^2	Velocity ft/sec	Kinetic Energy Ft-lbs
III	1.	RTE	1.0	.2	208	670
	2.	RTE & Plug	2.0	4.0	448	6230

Table 4-28

Properties of Missiles - System Piping

Missile Class	Description	Weight (lbs.)	Impact Area in^2	Velocity Ft/sec	Kinetic Energy Ft-lbs
	Core Flooding Line				
I	14" C.V. bonnet stud	2.0	1.7	73.5	167
I	14" Valve bonnet stud	3.5	4.0	67.5	248
II	14" C.V. check pivot stud	10.0	1.75	249	9650
II	14" P.O. valve stem	98.0	5.0	143	31,100
III	14" C.V. bonnet & assembly	525.0	125	448	1,640,000
III	14" P.O. valve bonnet and assy.	1900.0	650	558	9,180,000
	L.P. Injection Line				
I	12" C.V. bonnet stud	2.0	1.7	73.5	167
II	12" C.V. check pivot stud	10	1.75	249	9,650
III	12" C.V. bonnet and assy.	450	95	558	2,170,000
	R.V. Outlet Line to L.P. System				
I	10" Valve bonnet stud	2.5	1.7	73.5	177
I	Relief valve bonnet stud	0.5	.3	73.5	42
I	Relief valve stem assy.	40	12.5	35.3	768
II	10" EMO valve stem	50	3.1	130	13,200
III	10" EMO valve bonnet & assy.	1270	415	558	6,140,000
	R.V. Inlet Line from H.P. System				
I	4" C.V. bonnet stud	1.0	.8	73.5	83.5
II	4" C.V. check pivot stud	3.0	.8	158	1170
III	4" C.V. bonnet and assy.	30	19	558	145,000
	S.G. Outlet Line to Pump Inlet				
I	1" Drain valve bonnet stud	0.8	.6	73.5	67
II	1" Drain valve stem assy.	4.0	.3	84	438
III	1" Drain valve & bonnet assy.	30.0	27	448	84,380

Table 4-28 (Cont'd.)
Properties of Missiles - System Piping

Missile Class	Description	Weight (lbs.)	Impact Area in^2	Velocity Ft/sec	Kinetic Energy Ft-lbs
	Pressurizer to C.A. System Line				
I	3/4" Valve bonnet stud	1.0	.45	73.5	83
II	3/4" Valve stem	4	.3	73	330
II	3/4" EMO valve stem	4	.3	73	330
III	3/4" Valve bonnet and assy.	20	21	425	56,250
III	3/4" EMO valve bonnet and assy.	115	21	280	140,000
	Primary Pump Seal Water Return to H.P. System Line				
I	3" EMO valve bonnet stud	1.0	1.0	73.5	83.5
II	3" EMO valve stem	25.0	.3	125.7	6150
III	3" EMO valve bonnet and assy.	285.0	85	507	1,137,000
	Letdown Cooler Inlet & Outlet Lines				
I	1½" EMO valve bonnet stud	2.0	.8	73.5	167
II	1½" EMO valve stem	1.0	1.0	153.2	1830
III	1½" EMO valve bonnet and assy.	250.0	38	320	397,000
	Primary Pump Seal Water Inlet and Outlet Lines				
I	3" Inlet C.V. bonnet stud	1.0	.8	73.5	83.5
I	3" Outlet valve bonnet stud	2.0	1.0	73.5	167
II	3" C.V. check pivot stud	3.0	.8	158.4	1170
II	3" Outlet valve stem	25.0	2.4	125.7	6150
III	3" Inlet C.V. bonnet and assy.	25.0	85	558	120,800
III	3" Outlet valve bonnet and assy.	65.0	85	523	276,000

Rev. 5. 5/25/70
(New Page)

Table 4-28 (Cont'd.)

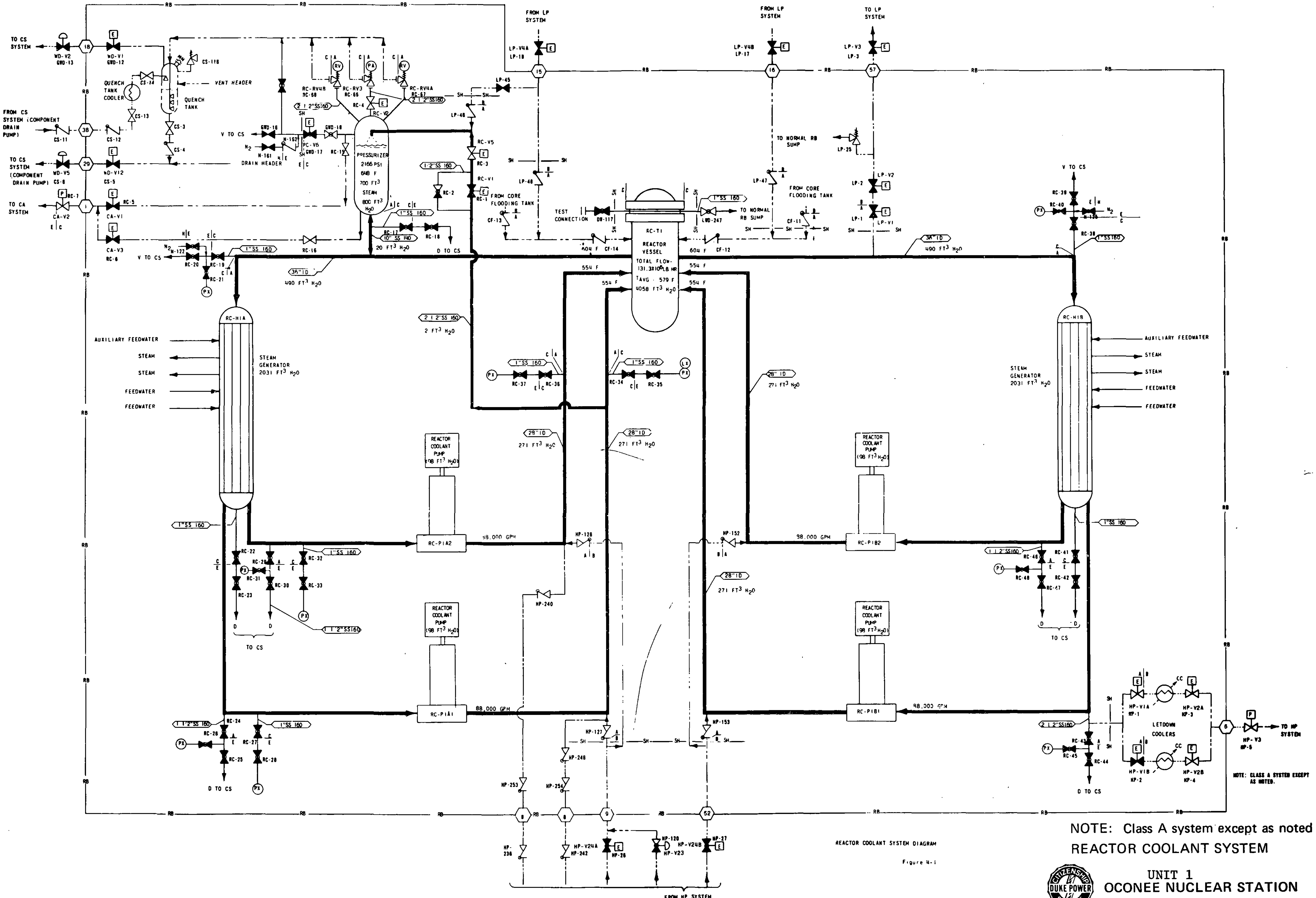
Properties of Missiles - System Piping

Missile Class	Description	Weight (lbs.)	Impact Area in^2	Velocity Ft/sec	Kinetic Energy Ft-lbs
	Primary Pump Vent & Drain Lines				
I	$1\frac{1}{2}$" Vent & drain valve bonnet stud	2.0	.8	73.5	167
II	$1\frac{1}{2}$" Vent & drain valve stem	5.0	1.0	153.2	1830
III	$1\frac{1}{2}$" Vent & drain valve bonnet and assy.	55.0	38	435.0	161,600

Table 4-29

Steam Generator Stress Intensities and Usage Factors

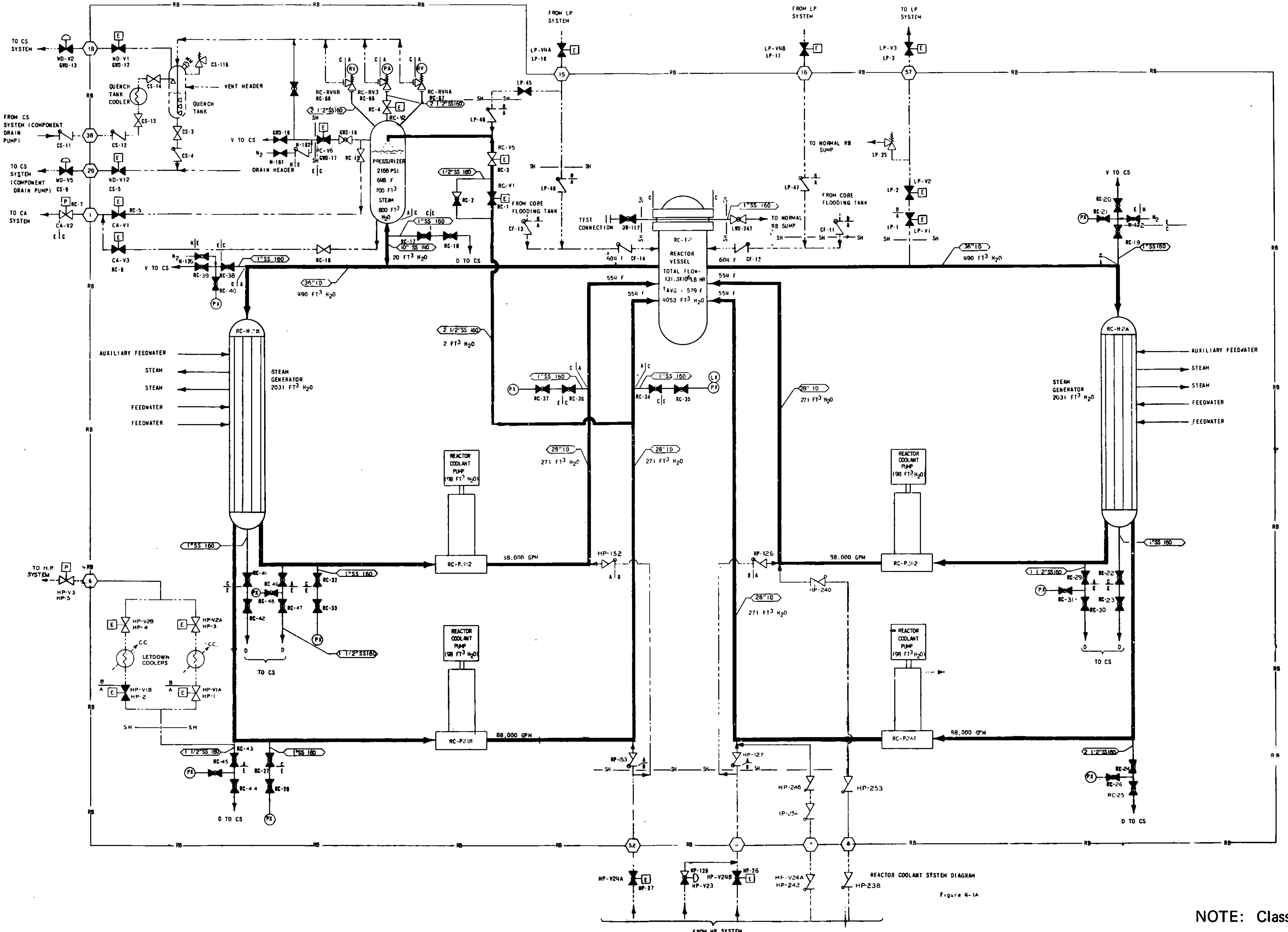
	Pri. + Sec. Stress Intensity Range	Usage Factor
Upper & Lower Tube Sheet	35.0 Ksi	0.13
Primary Inlet Nozzle	18.0 Ksi	0.01
Primary Outlet Nozzle	24.0 Ksi	0.01
Steam Outlet Nozzle	27.0 Ksi	0.0
Auxiliary Feedwater Nozzle	44.5 Ksi	0.0
OTSG Shell	25.5 Ksi	0.0
Feedwater Nozzle	50.7 Ksi	0.56



NOTE: Class A system except as noted

REACTOR COOLANT SYSTEM

UNIT 1
OCONEE NUCLEAR STATION
Figure 4 - 1
Rev. 16 7/30/71
Rev. 21, 7/21/72
Rev. 35, 9/30/74



NOTE: Class A system except as noted

REACTOR COOLANT SYSTEM

OCONEE NUCLEAR STATION

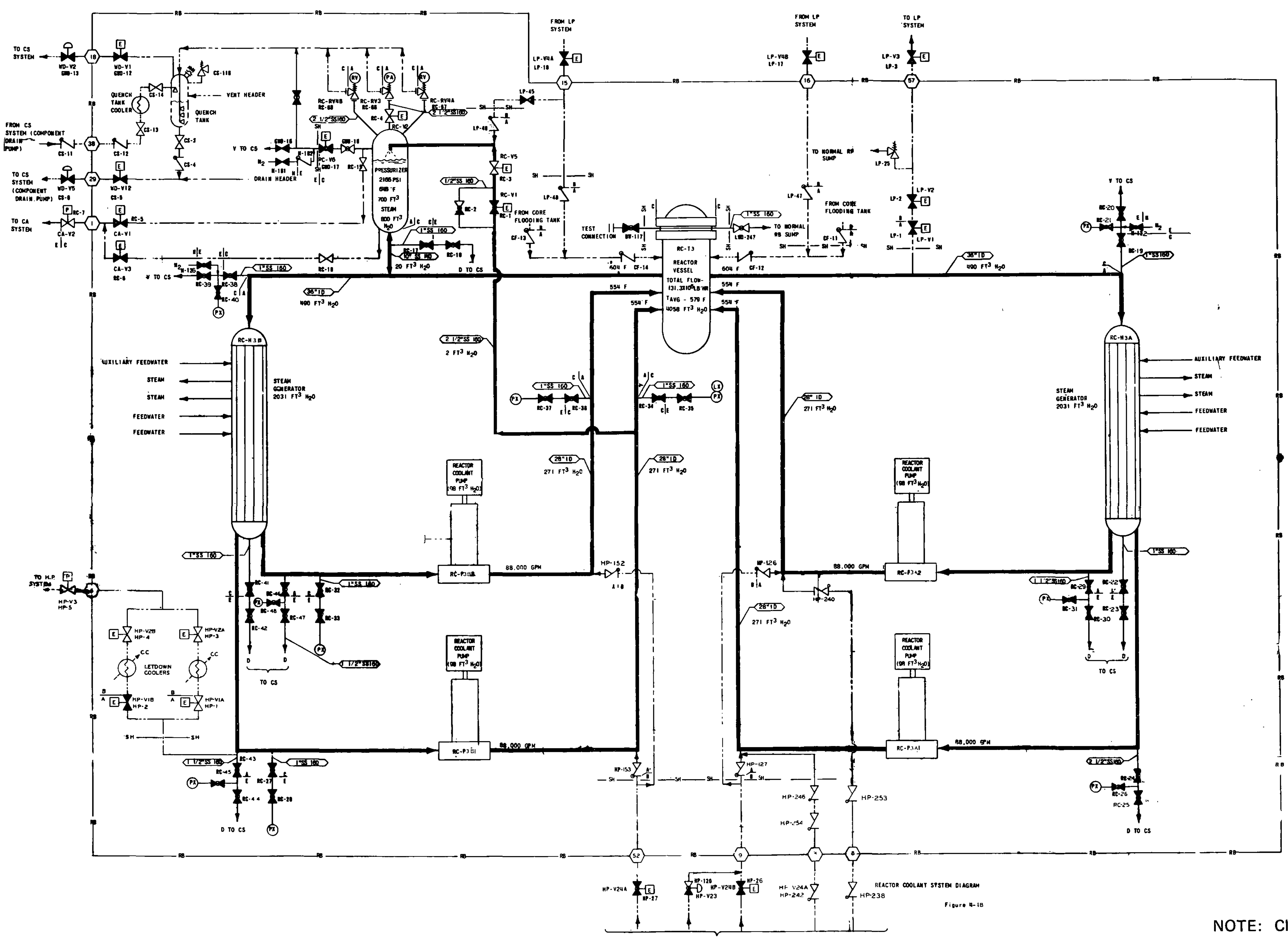
UNIT 2

Figure 4 - 1A

Rev. 16 7/30/71

Rev. 21 7/26/72

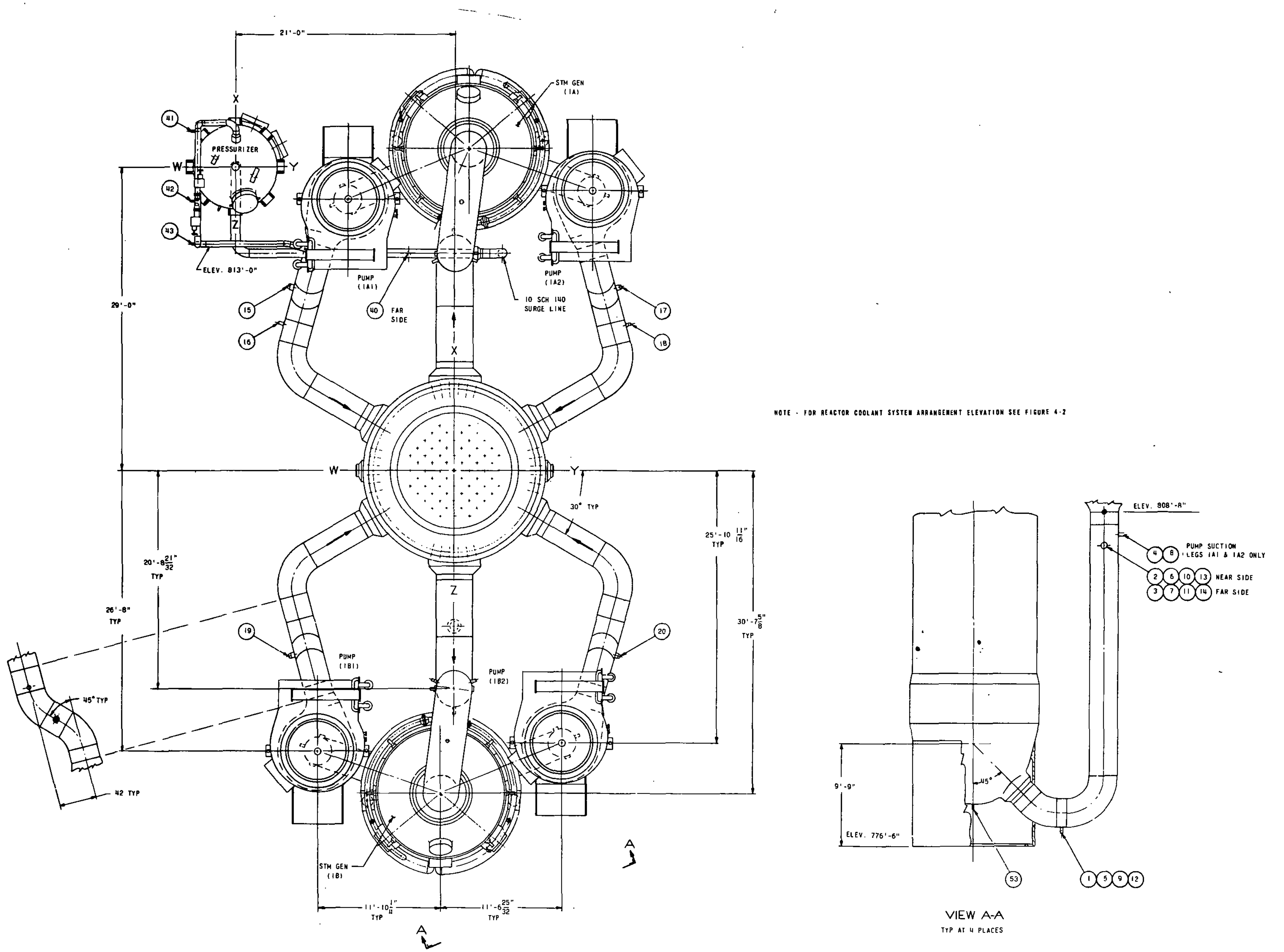
Rev. 35. 9/30/74



NOTE: Class A system except as noted

REACTOR COOLANT SYSTEM

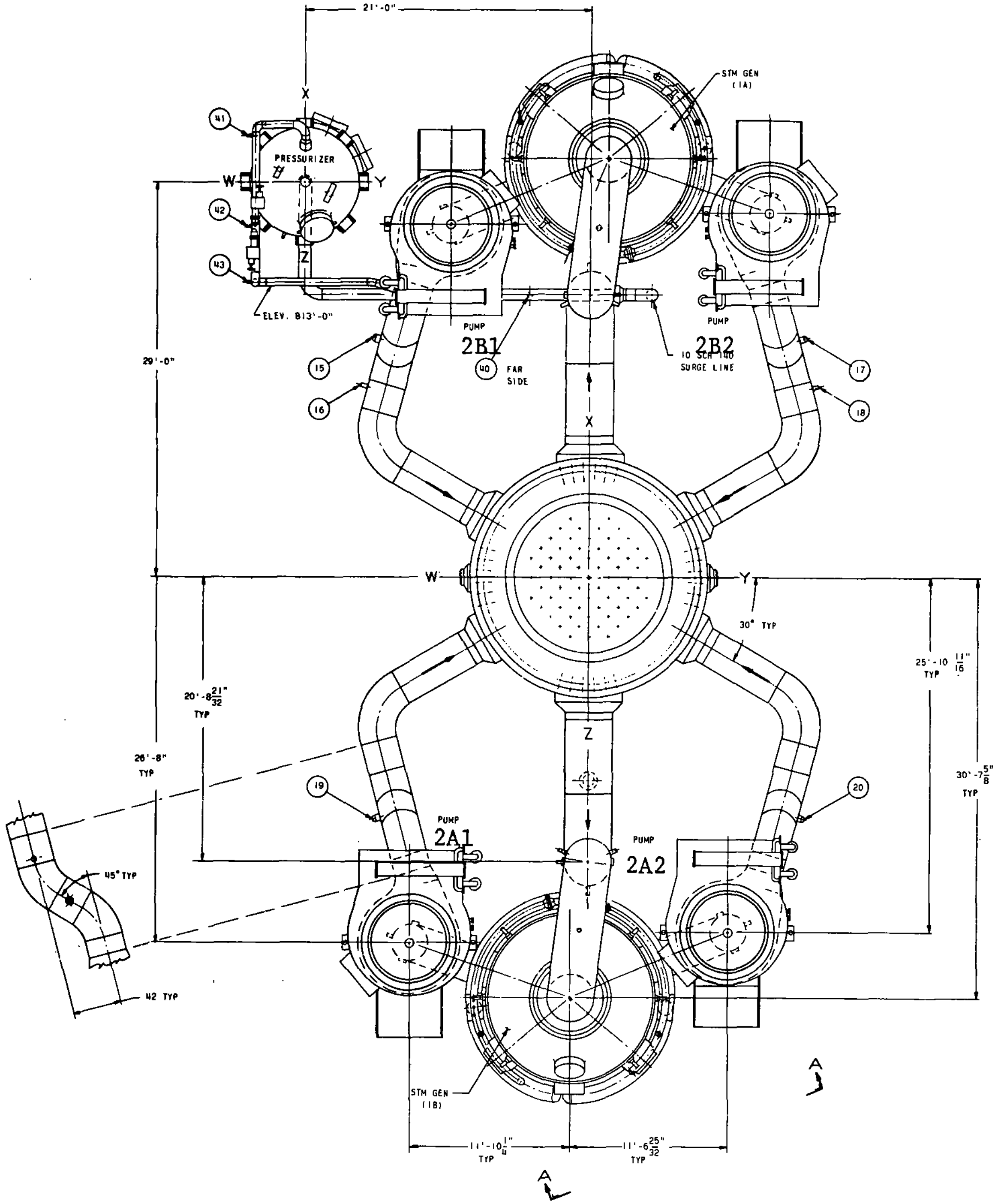
OCONEE NUCLEAR STATION
UNIT 3
Figure 4 - 1B
Rev. 16 7/30/71
Rev. 21 7/26/72
Rev. 35. 9/30/74



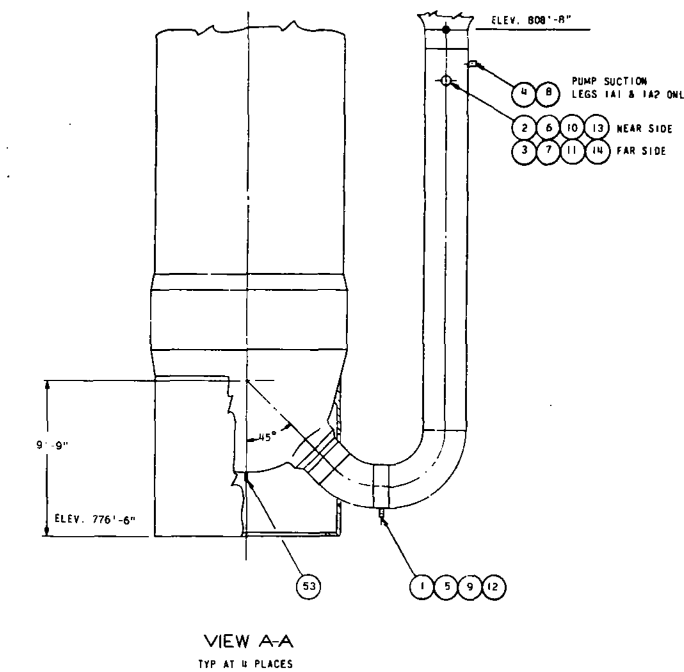
(Reference Supplement 6 revisions for Oconee 1)

REACTOR COOLANT SYSTEM ARRANGEMENT – PLAN

UNIT 1
OCONEE NUCLEAR STATION
Figure 4 - 3
Rev. 9 8/11/70
Rev. 21 7/21/72



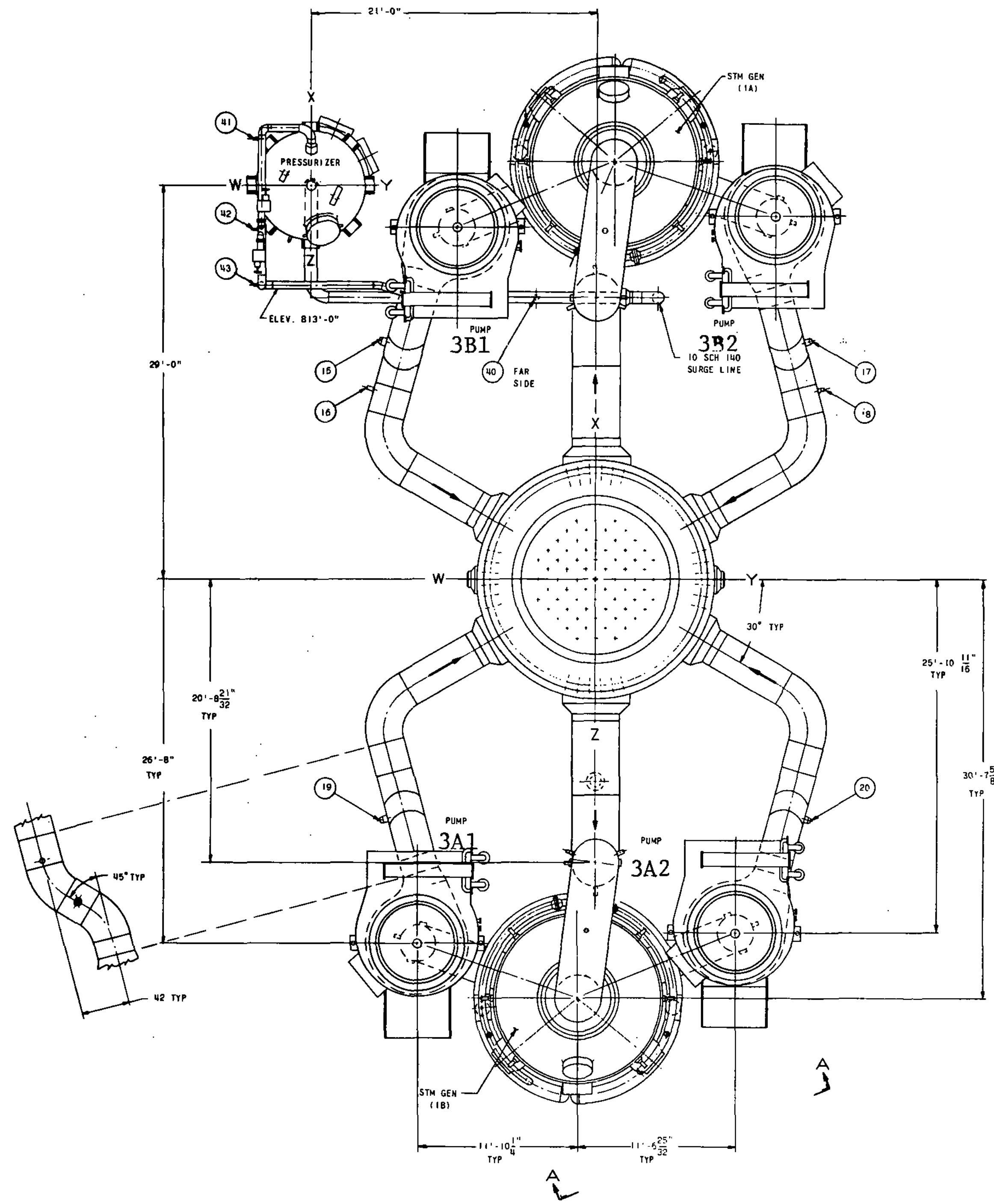
NOTE - FOR REACTOR COOLANT SYSTEM ARRANGEMENT ELEVATION SEE FIGURE 4-2



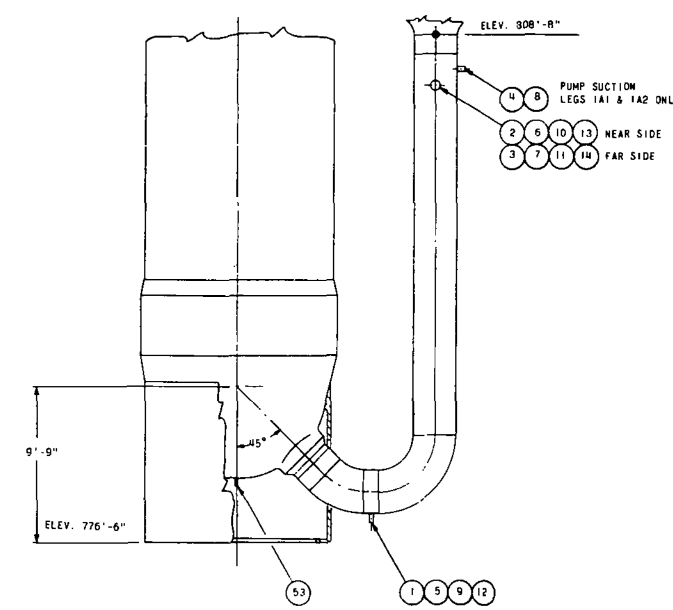
VIEW A-A
TYP AT 4 PLACES

REACTOR COOLANT SYSTEM ARRANGEMENT – PLAN

UNIT 2
OCONEE NUCLEAR STATION
Figure 4 - 3A
Rev. 9 8/11/70
Rev. 21 7/21/72



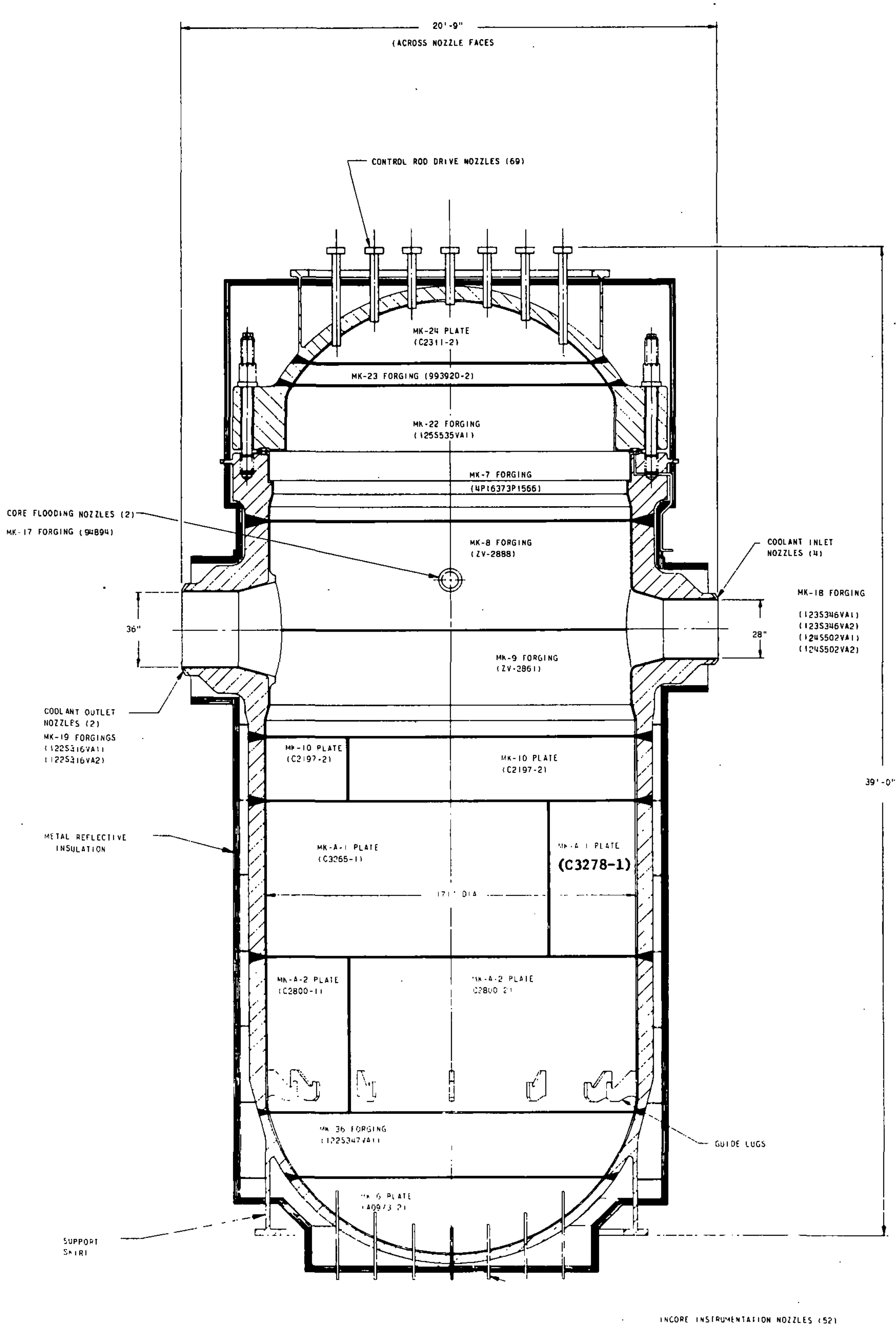
NOTE - FOR REACTOR COOLANT SYSTEM ARRANGEMENT ELEVATION SEE FIGURE 4-2



VIEW A-A
TYP AT 4 PLACES

REACTOR COOLANT SYSTEM
ARRANGEMENT – PLAN

UNIT 3
OCONEE NUCLEAR STATION
Figure 4 - 3 B
Rev. 9 8/11/70
Rev. 21. 7/26/72



REACTOR VESSEL OUTLINE

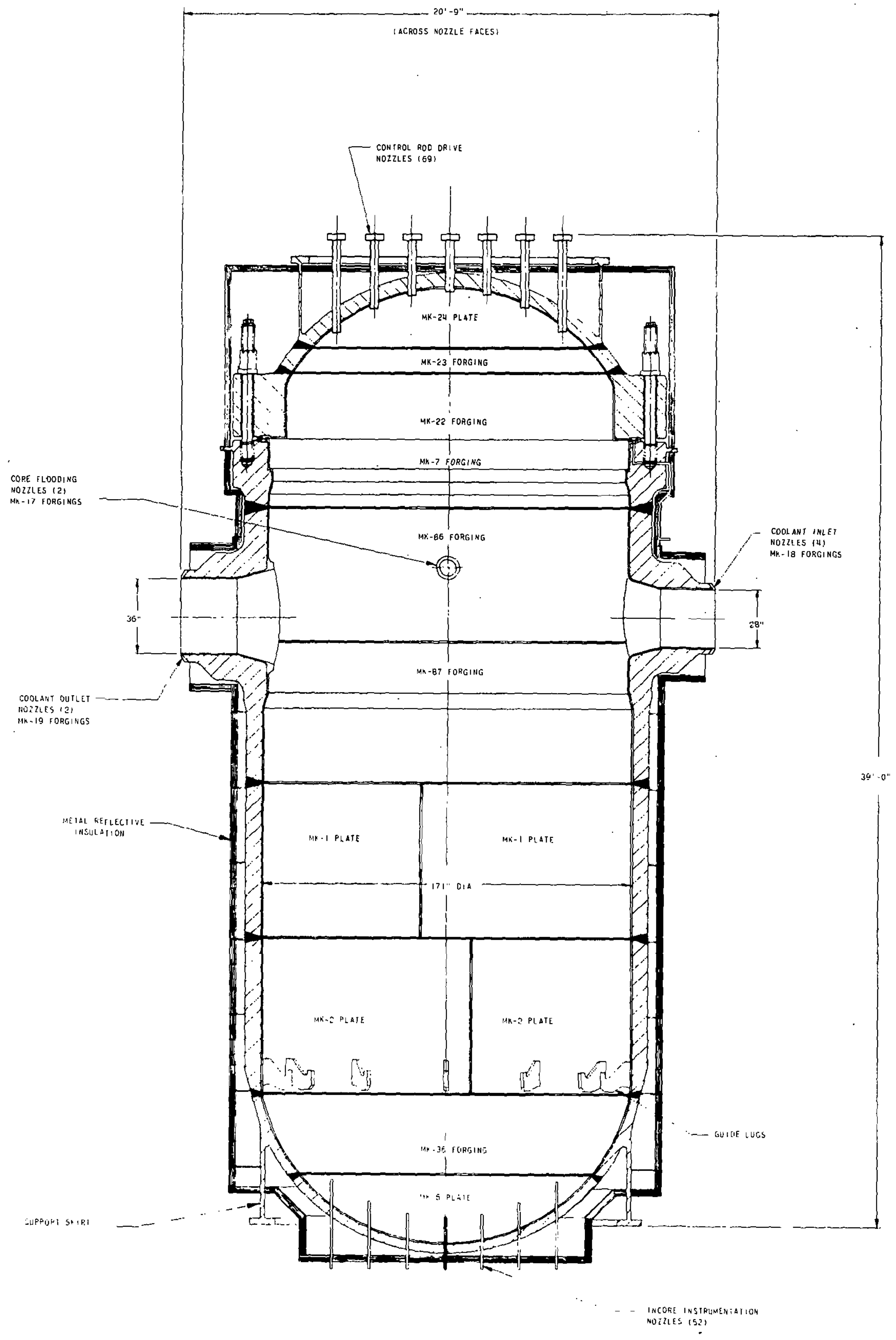
OCONEE I

OCONEE NUCLEAR STATION

Figure 4 - 4

Rev. 4 4/20/70

Rev. 16. 7/30/71



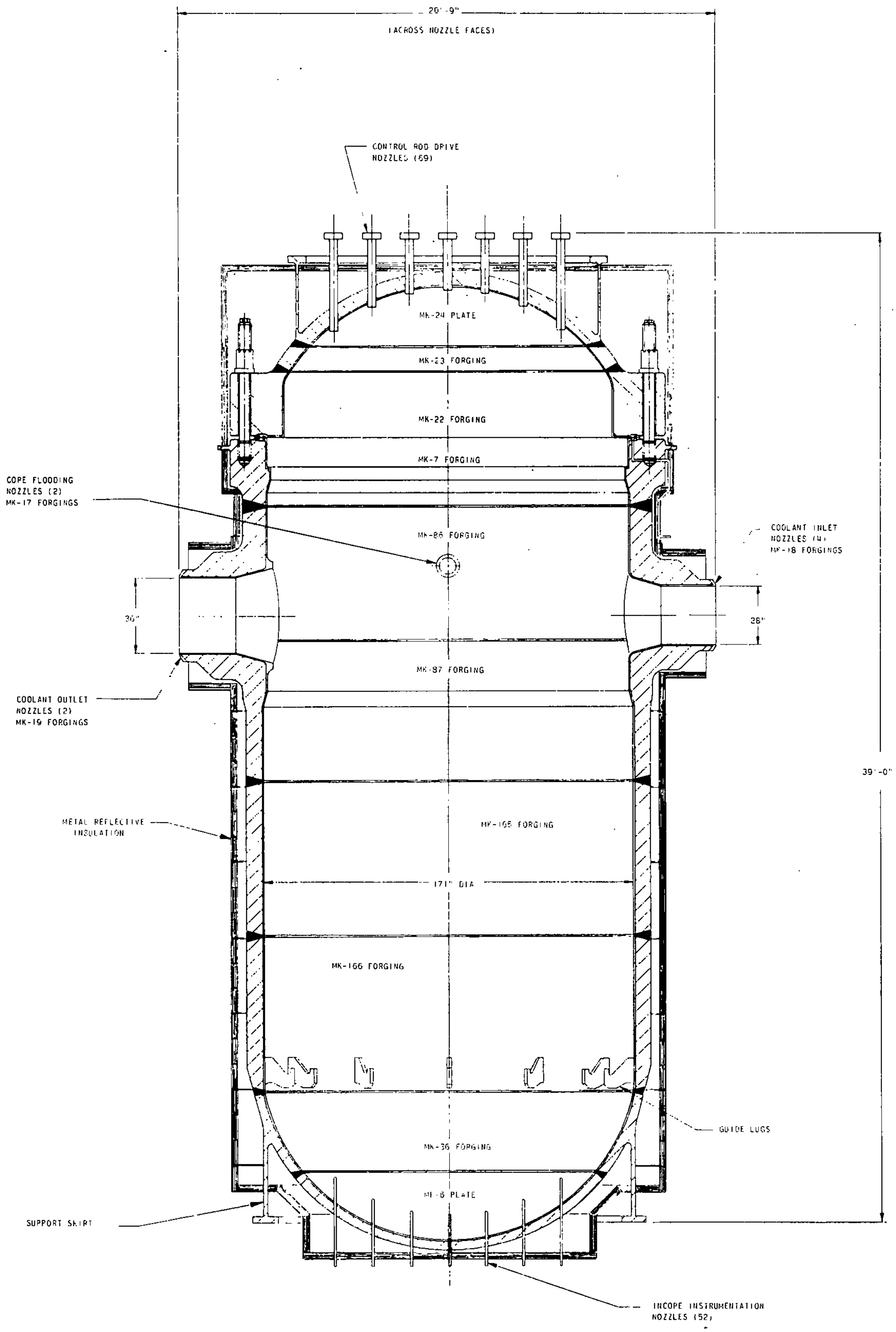
REACTOR VESSEL OUTLINE

OCONEE 2

OCONEE NUCLEAR STATION

Figure 4 - 4A

(New) Rev. 4 4/20/70



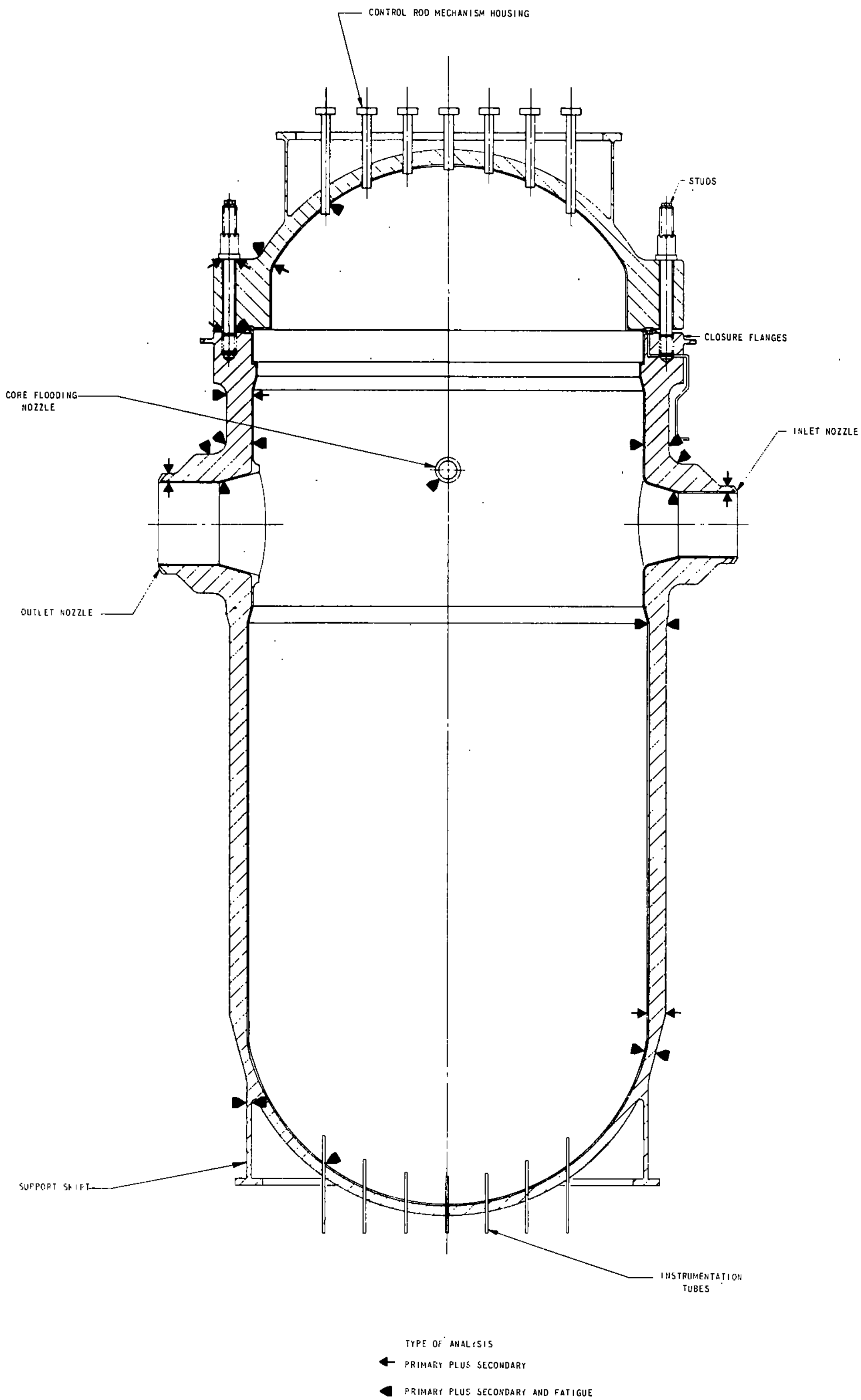
REACTOR VESSEL OUTLINE

OCONEE 3

OCONEE NUCLEAR STATION

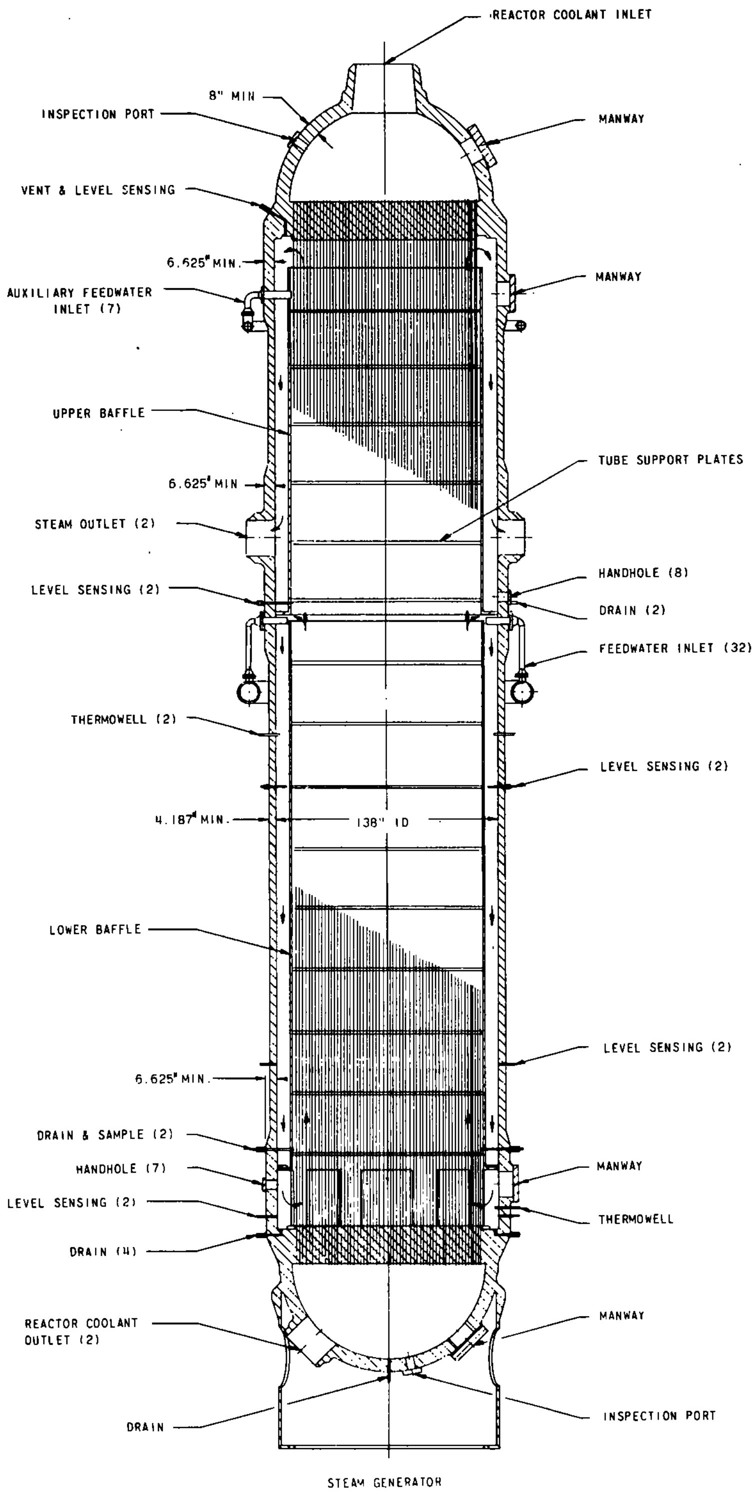
Figure 4 - 4B

(New) Rev. 4 4/20/70



POINTS OF STRESS ANALYSIS
FOR REACTOR VESSEL

OCONEE NUCLEAR STATION
Figure 4 - 4C
(New) Rev. 4 4/20/70



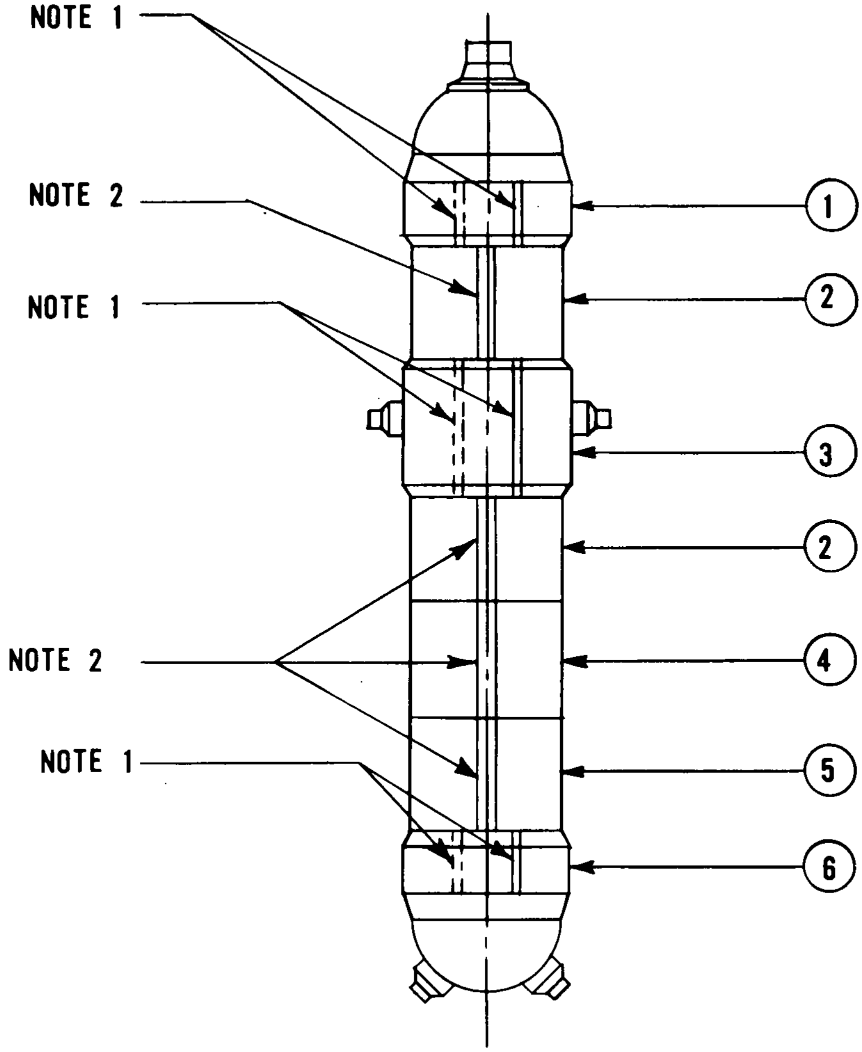
STEAM GENERATOR

STEAM GENERATOR

REFERENCE SUPPLEMENT 9
REVISIONS FOR OCONEE 3

OCONEE NUCLEAR STATION
Figure 4 - 5

Rev. 16. 7/30/71



STEAM GENERATOR

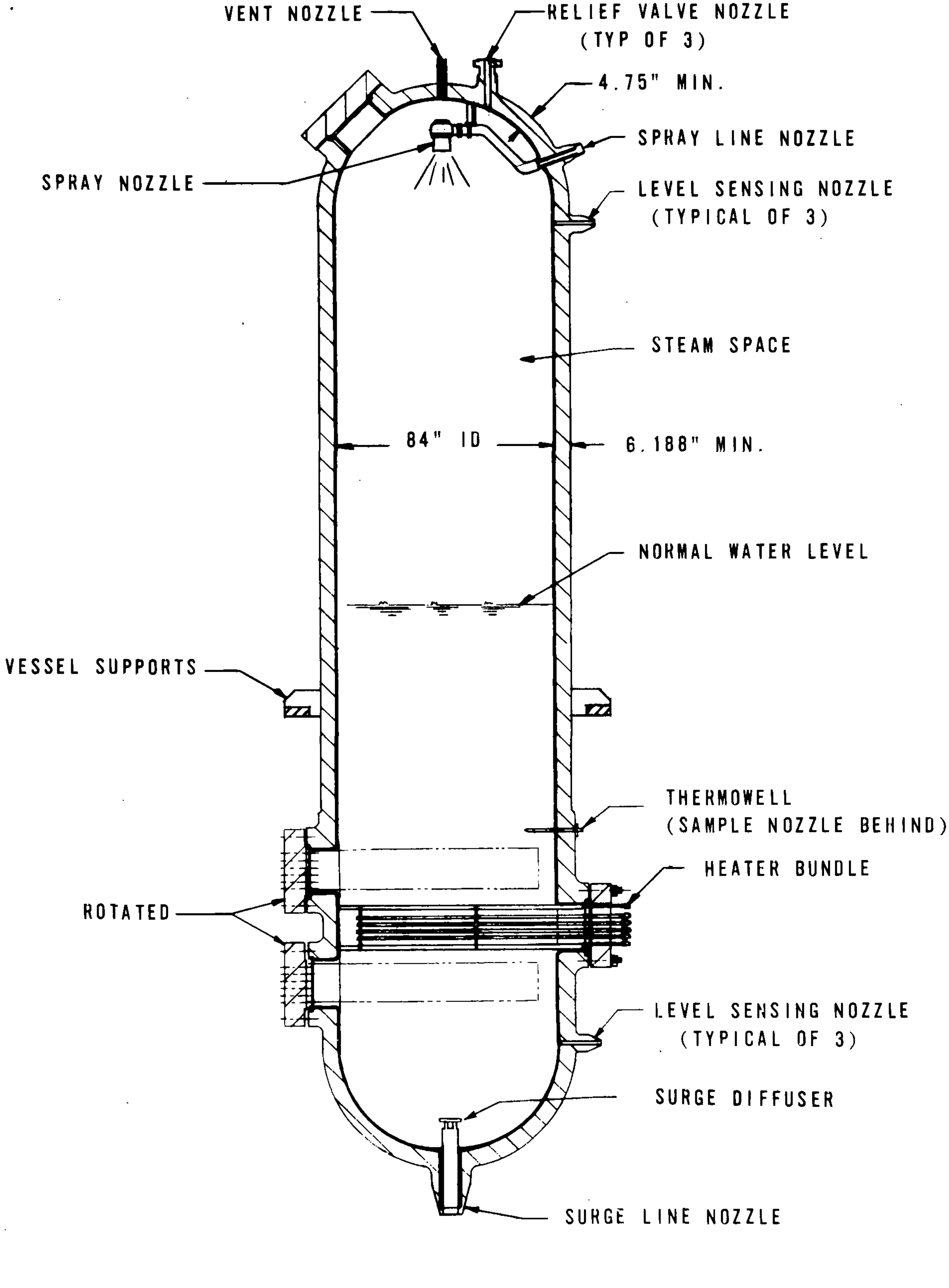
NOTE 1 - ELECTROSLAG WELDS ON OCONEE 1,2,3

NOTE 2 - ELECTROSLAG WELDS ON OCONEE 1,2. SUBMERGED ARC WELDS ON OCONEE 3

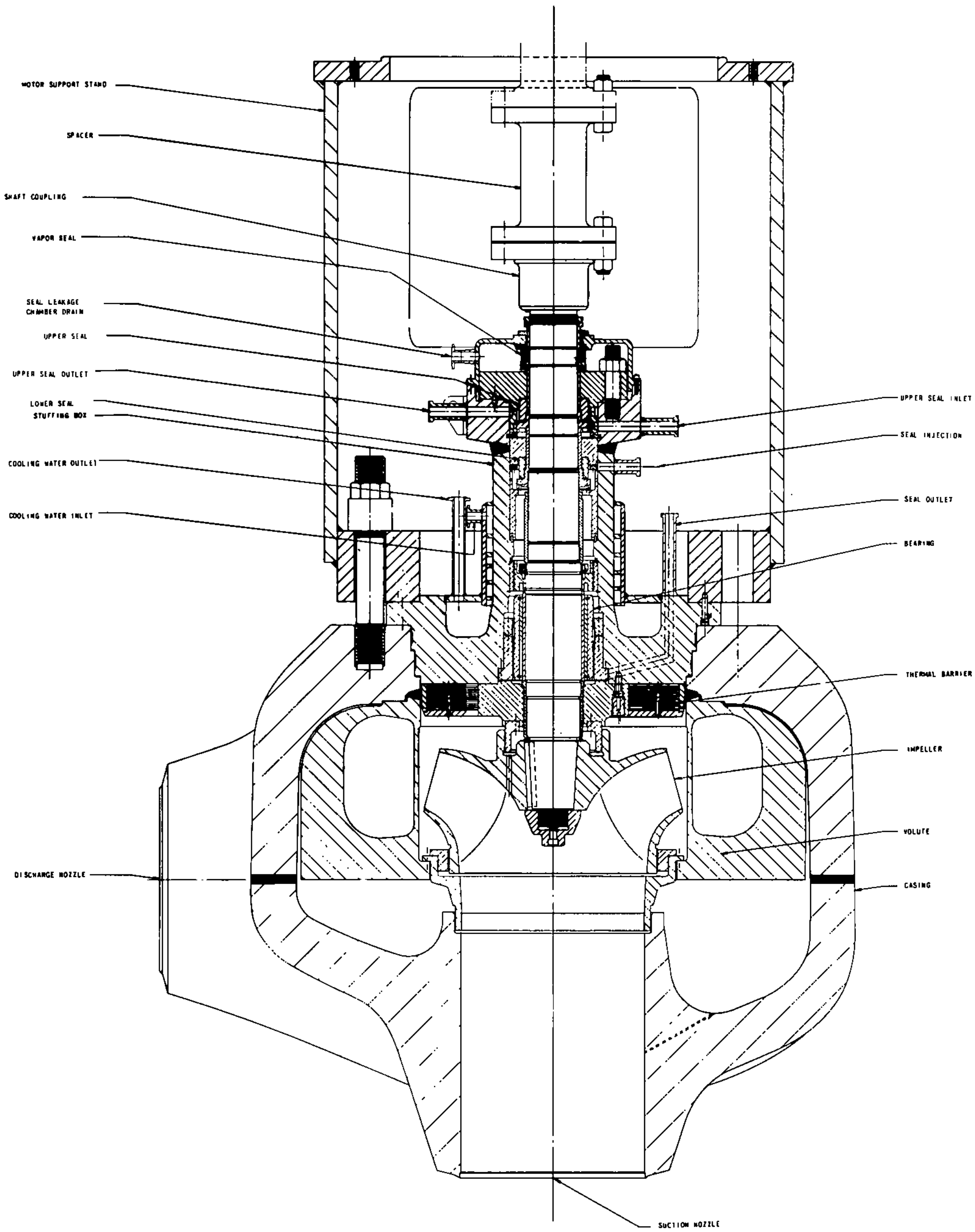
OCONEE NUCLEAR STATION

Figure 4 - 5A

(New) Rev. 5 5/25/70



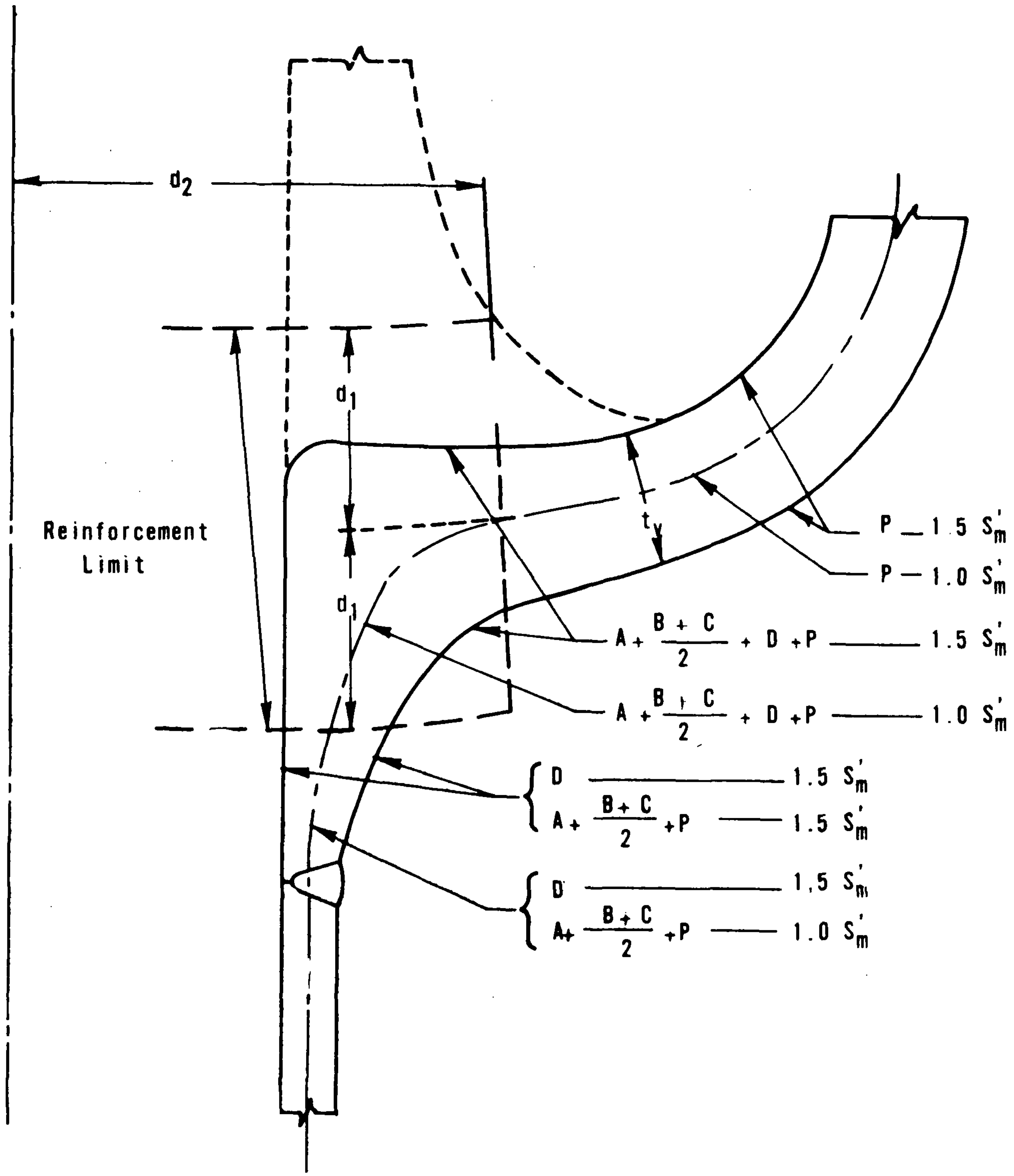
PRESSURIZER OUTLINE



(Reference Supplement 6 revisions for Oconee 1)

REACTOR COOLANT PUMP

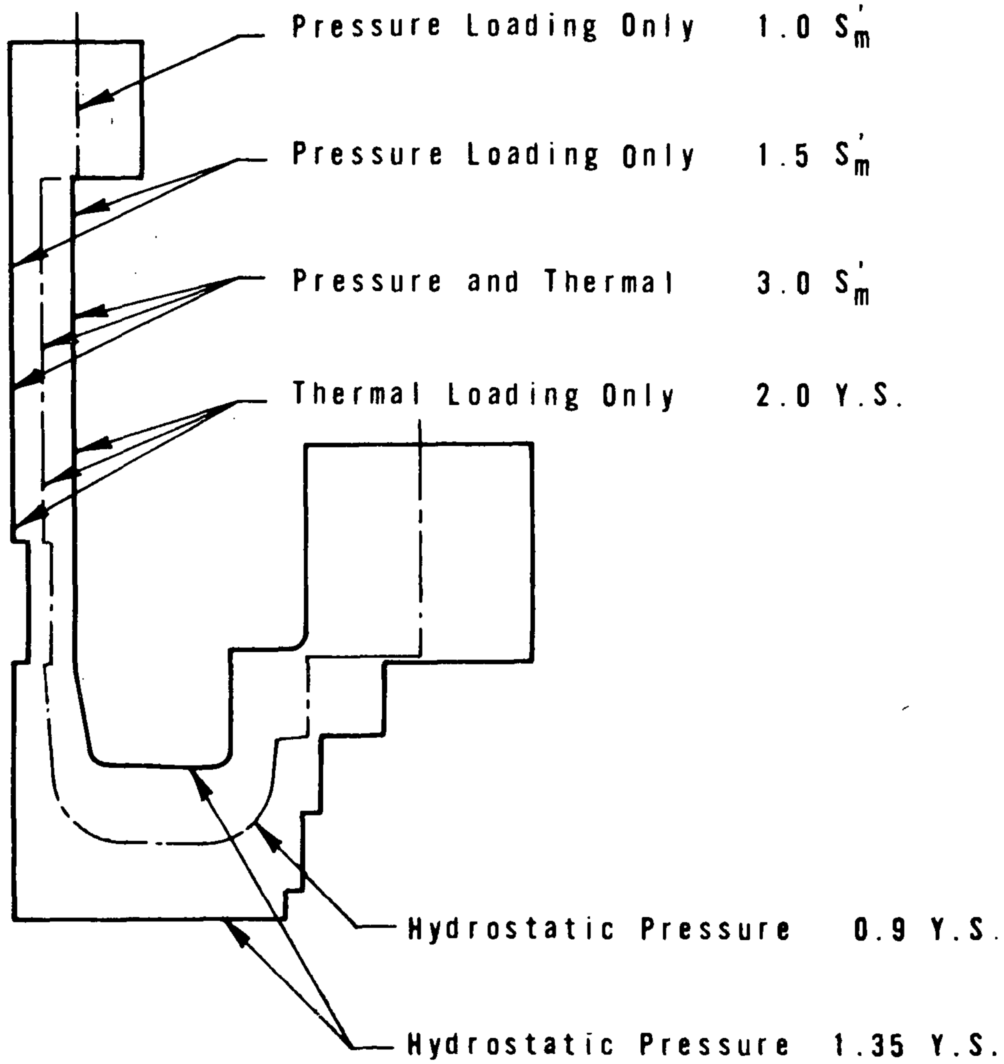
OCONEE NUCLEAR STATION
Figure 4 - 7
Rev. 9 8/11/70
Rev. 16 7/30/71



CODE ALLOWABLES AND REINFORCING LIMITS
NOZZLES AND BOWL

(Applies to Units 2&3)
OCONEE NUCLEAR STATION

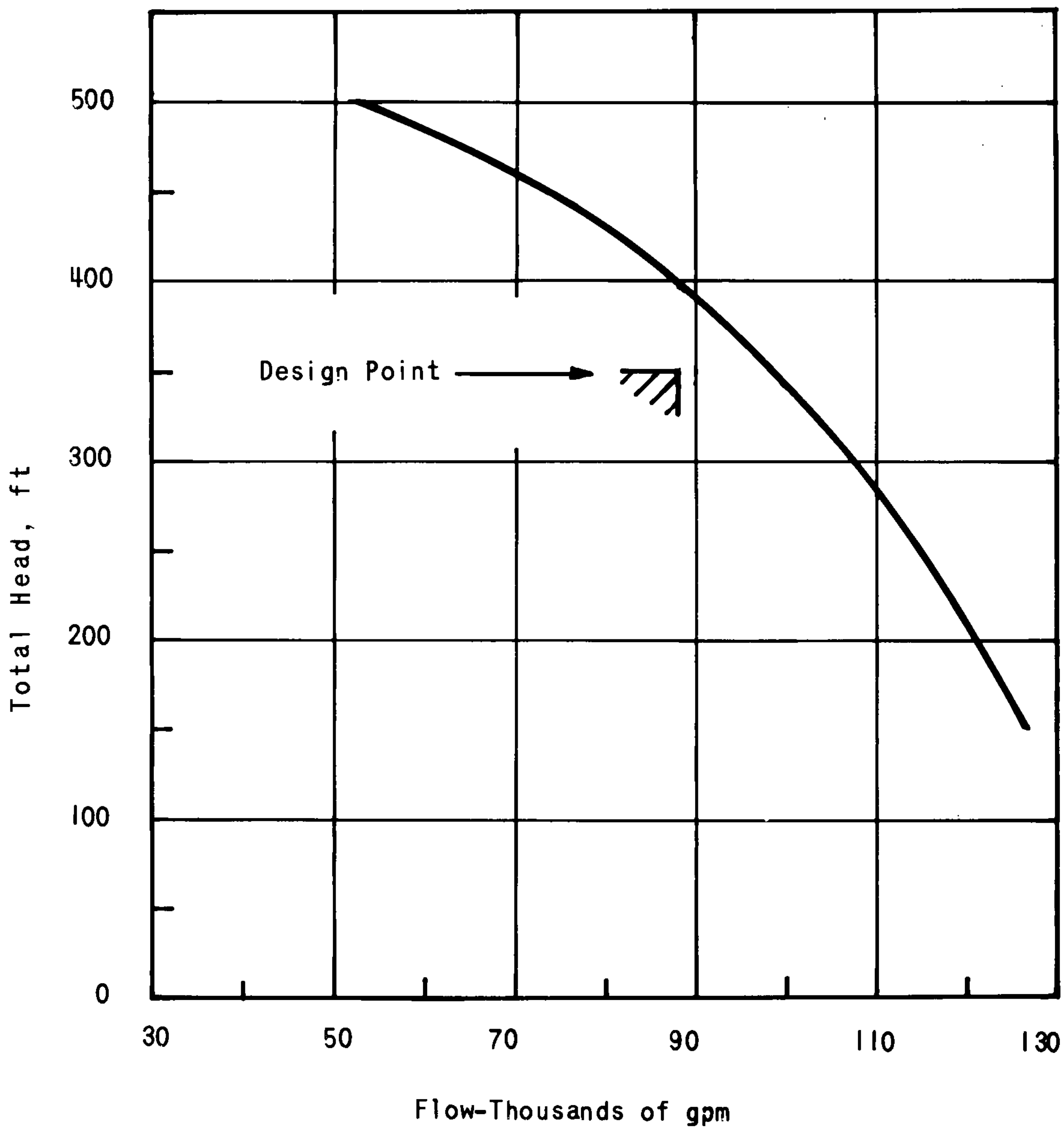
Figure 4 - 7A
(New) Rev. 5 5/25/70
Rev. 9 8/11/70



CODE ALLOWABLES, COVER

(Applies to Units 2&3)
OCONEE NUCLEAR STATION

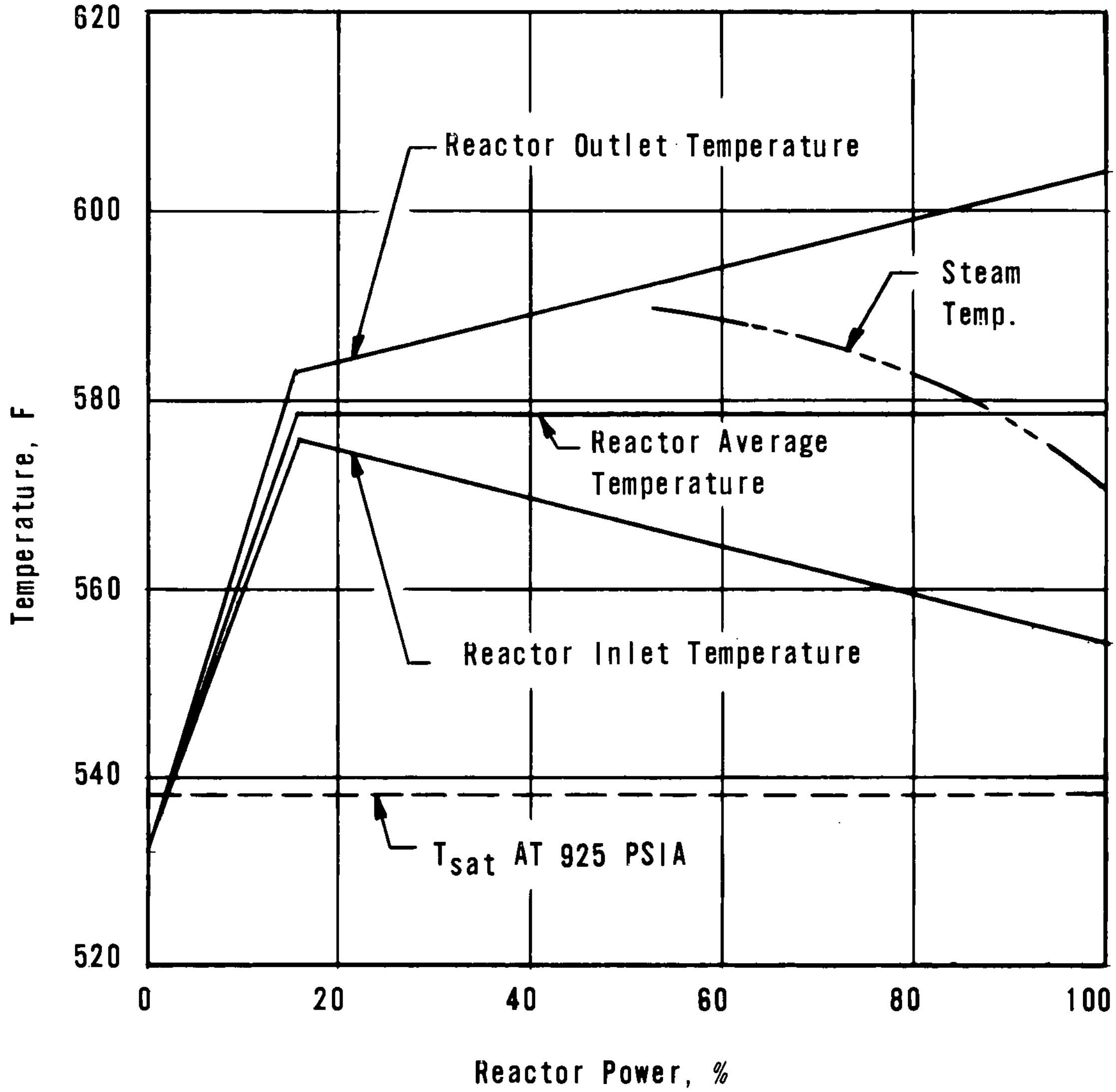
Figure 4 - 7B
(New) Rev. 5 5/25/70
Rev. 9 8/11/70



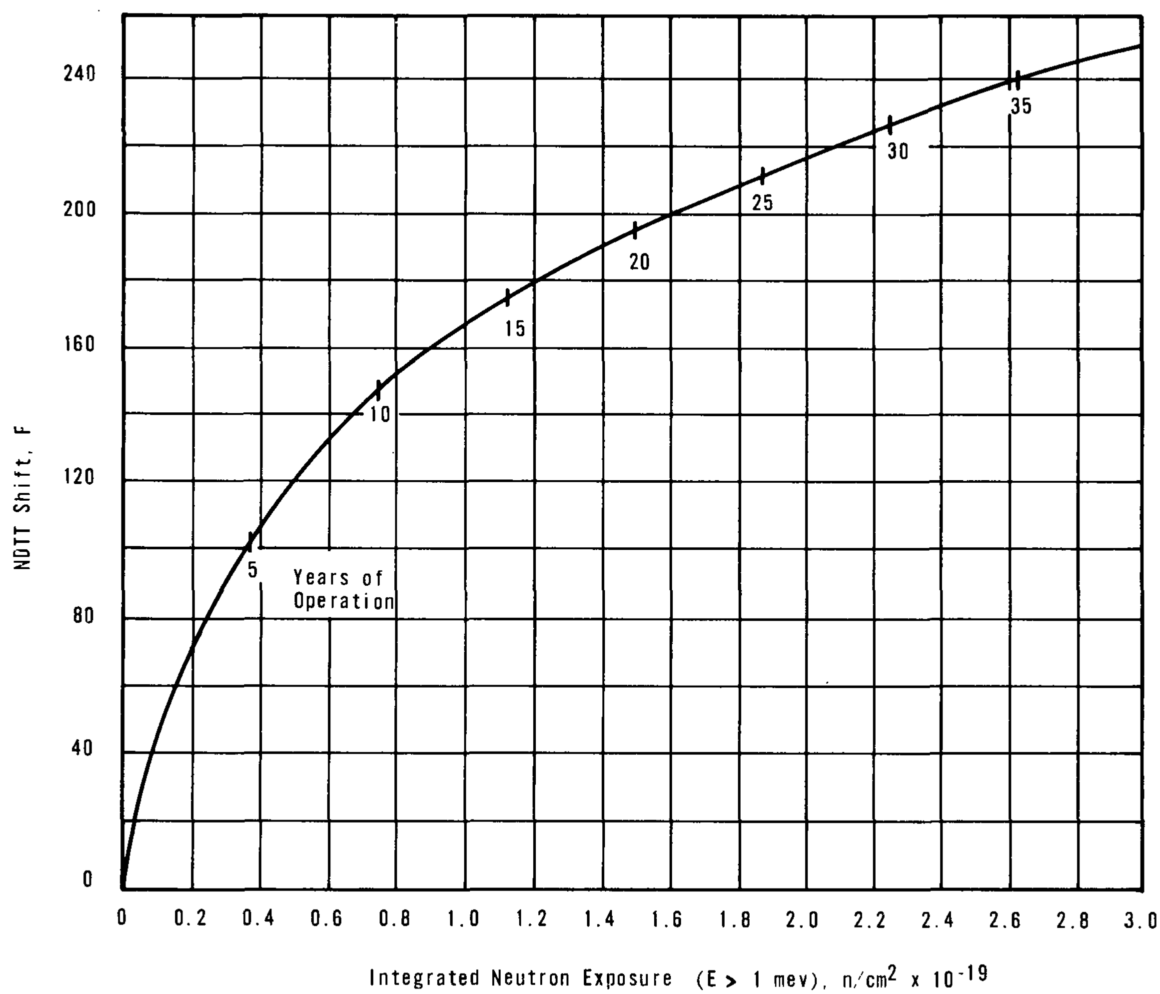
(Reference Supplement 6 revisions for Oconee 1)

REACTOR COOLANT PUMP ESTIMATED PREFORMANCE CHARACTERISTIC

OCONEE NUCLEAR STATION
Figure 4 - 8
Rev. 9 8/11/70



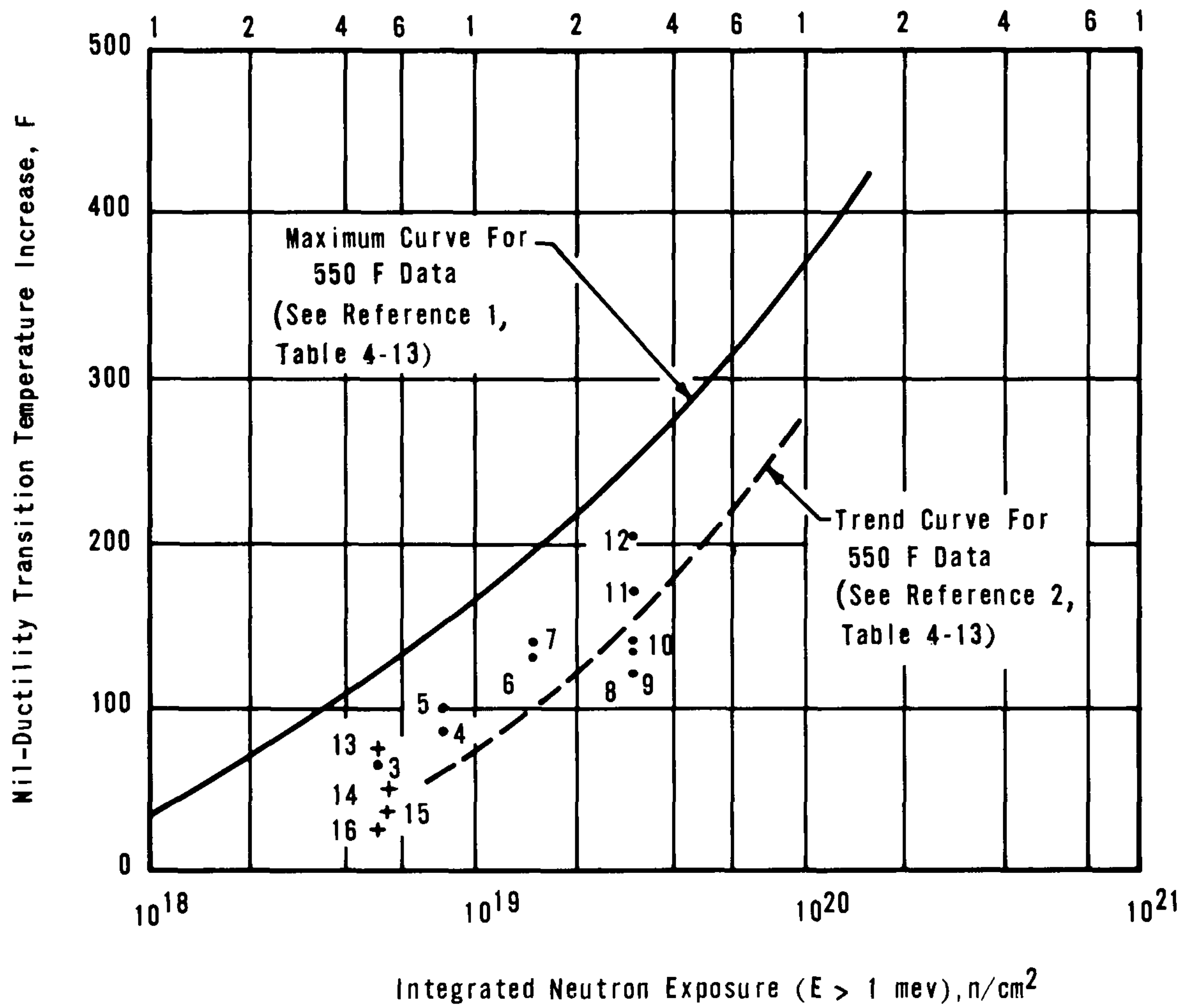
REACTOR AND STEAM TEMPERATURES
VERSUS REACTOR POWER



PREDICTED NDTT SHIFT VERSUS REACTOR VESSEL IRRADIATION

OCONEE NUCLEAR STATION

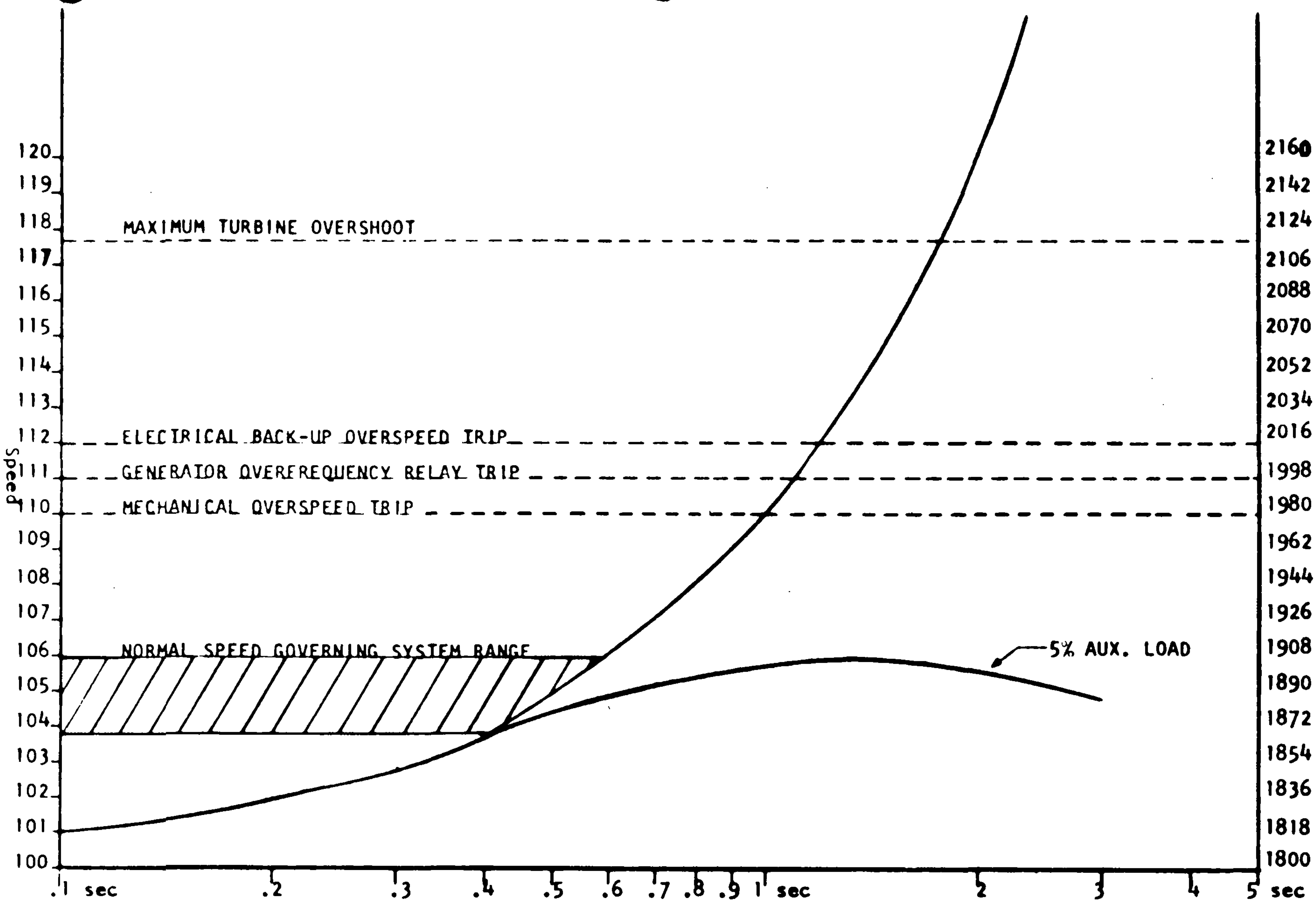
Figure 4 - 10



NOTES:

1. All data is for 30 ft-lb "Fix"
2. Curve from ASME Paper No.63-WA-100
   Numbers indicate references in Table 4-13

NDTT VERSUS INTEGRATED NEUTRON EXPOSURE FOR A302B STEEL



OCONEE NUCLEAR STATION
Turbine Generator Speed Response Following Load Rejection - with VWO Load - Showing Protection Points

OCONEE NUCLEAR STATION

Figure 4 - 12

(New) Rev. 4 4/20/70

LIST OF EFFECTIVE PAGES

FSAR APPENDIX 4A

INSERVICE INSPECTION

APPENDIX 4A
INSERVICE INSPECTION

This section has been revised as of April 20, 1970.

APPENDIX 4A

4A IN-SERVICE INSPECTION

The in-service inspection program is, except as noted below, in compliance with the ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Reactor Coolant Systems, 1970 edition.

The latest industry accepted equipment and techniques will be utilized in both the preoperational base-line inspections and in subsequent in-service inspections.

As improved equipment and techniques, to yield at least equivalent results and reduced personnel exposures, become available in the future, it is expected that these will be adopted.

The following exceptions are taken to the requirements of the code:

1. Due to system design limitations, the following item will be inspected at or near the end of the ten year inspection interval, rather than as prescribed in Section IS-242:

   Item 1.5, (Examination Category E-1) Instrumentation Penetrations.

2. In Item 1.4, (Examination Category D), one reactor coolant outlet nozzle will be inspected approximately 3-1/3 years after initial operation. The second reactor coolant outlet nozzle will be inspected approximately 6-2/3 years after initial operation. The four reactor coolant inlet nozzles and the two core flooding nozzles will be inspected at or near the end of the ten year inspection interval.

3. The following items are not applicable to the Oconee Nuclear Station:

Item 3.3	Primary nozzle to safe-end welds.
Item 4.3	Valve pressure retaining bolting larger than 2".
Item 5.3	Pump nozzle to safe-end welds.
Item 6.1	Valve body welds.
Item 6.3	Valve to safe-end welds.
Item 6.6	Integrally welded valve supports.
Item 6.7	Valve supports and hanger.

Rev. 7. 7/9/70
(Revised Page)

TABLE 4.1-1 Cont.

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
20. Reactor Building Spray System Logic Channel	NA	M	NA	
21. Reactor Building Spray System Analog Channels				
a. Reactor Building High Pressure Channels	NA	M	R	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	R(2)	(1) Check with Relative Position Indicator (2) Calibrate rod misalignment channel
24. Control Rod Relative Position	S(1)	NA	R(2)	(1) Check with Absolute Position Indicator (2) Calibrate rod misalignment channel
25. Core Flooding Tanks				
a. Pressure Channels	S	NA	R	
b. Level Channels	S	NA	R	
26. Pressurizer Level Channels	S	NA	R	
27. Letdown Storage Tank Levels Channels	D	NA	R	
28. Radiation Monitoring Systems	W(1)	M	Q	(1) Check functioning of self-checking feature on each detector.
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

TABLE 4.1-1 Cont.

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
30. Borated Water Storage Tank Level Indicator	W	NA	R	
31. Boric Acid Mix Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
32. Concentrated Boric Acid Storage Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
33. Containment Temperature	NA	NA	R	
34. Incore Neutron Detectors	M(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout
35. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check
36. Environmental Monitors	M(1)	NA	R	(1) Check Functioning
30. 37. Reactor Manual Trip	NA	P	NA	
38. Reactor Building Emerg. Sump Level	NA	NA	R	
39. Steam Generator Water Level	W	NA	R	
40. Turbine Overspeed Trip	NA	NA	R	
41. Engineered Safeguards Channel 1 HP Injection Manual Trip	NA	R	NA	

TABLE 4.1-1 Cont.

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
42. Engineered Safeguards Channel 2 HP Injection Manual Trip	NA	R	NA	
43. Engineered Safeguards Channel 3 LP Injection Manual Trip	NA	R	NA	
44. Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	R	NA	
45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	R	NA	
46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	R	NA	
47. Engineered Safeguards Channel 7 Spray Manual Trip	NA	R	NA	
48. Engineered Safeguards Channel 8 Spray Manual Trip	NA	R	NA	

30.

S - Each Shift

D - Daily

W - Weekly

M - Monthly

R - Each Refueling Period

NA- Not Applicable

Q - Quarterly

P - Prior to each startup if not done previous week

Table 4.1-2

Minimum Equipment Test Frequency

	Item	Test	Frequency
1.	Control Rods	Rod Drop Times of all full length rods	Each Refueling shutdown
2.	Control Rod Movement (1)	Movement of each rod	Every two weeks
3.	Pressurizer Safety Valves	Setpoint	50% each refueling period
4.	Main Steam Safety Valves	Setpoint	25% each refueling period
5.	Refueling System Interlocks	Functional	Each refueling period
6.	Turbine Steam Stop Valves(1)	Movement of each stop valve	Monthly
7.	Reactor Coolant System(2) Leakage	Evaluate	Daily
8.	Charcoal and High Efficiency Filters for Penetration Room, Control Room, and RB Purge Filters	DOP Test on HEPA filters. Freon Test on Charcoal Filter Units	Each refueling period and at any time work on filters could alter their integrity.
9.	Condenser Cooling Water System Gravity Flow Test	Functional	Each refueling period
10.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
11.	Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling

(1) Applicable only when the reactor is critical

(2) Applicable only when the reactor coolant is above 200°F and at a steady state temperature and pressure.

TABLE 4.1-3

MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency
1. Reactor Coolant	a. Gamma Isotopic Analysis	a. Monthly*
	b. Radiochemical Analysis for Sr 89, 90	b. Monthly*
	c. Tritium	c. Monthly*
	d. Gross Beta & Gamma Activity (1)	d. 5 times/week*
	e. Chemistry (Cl, F and O2)	e. 5 times/week*
	f. Boron Concentration	f. 2 times/week**
	g. Gross Alpha Activity	g. Monthly*
	h. $\bar{E}$ Determination (2)	h. Semi-annually
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly* and after each makeup
3. Core Flooding Tank	Boron Concentration	Monthly* and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly*** and after each makeup
5. Secondary Coolant	a. Gross Beta & Gamma Activity	a. Weekly*
	b. Iodine Analysis (3)	
6. Concentrated Boric Acid Tank	Boron Concentration	Twice weekly*

*Not applicable if reactor is in a cold shutdown condition for a period exceeding the sampling frequency.
**Applicable only when fuel is in the reactor.
***Applicable only when fuel is in the spent fuel pool.

Rev. 30. 9/4/73
(Entire Page Revised)

TABLE 4.1-3 Cont.

MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency	Sensitivity of Waste Analysis in Lab
7. Low Activity Waste Tank & Condensate Test Tank	a. Gross Beta & Gamma Activity	a. Prior to release of each batch	a. $<10^{-7}$ μCi/ml
	b. Radiochemical Analysis Sr 89, 90	b. Monthly	b. $<10^{-8}$ μCi/ml
	c. Gamma Analysis including Dissolved Noble Gases	c. Monthly	c. Gamma Nuclides $<5\times10^{-7}$ μCi/ml Dissolved Gases $<10^{-5}$ μCi/ml
	d. Tritium	d. Monthly	d. $<10^{-5}$ μCi/ml
	e. Gross Alpha Activity	e. Monthly	e. $<10^{-7}$ μCi/ml
	f. Ba-La-140, I-131	f. Weekly Proportional	f. $<5\times10^{-7}$ μCi/ml
8. Waste Gas Decay Tank	a. Gamma Isotopic Analysis	a. Prior to release of each batch	a. $<10^{-4}$ μCi/cc
	b. Gross Gamma Activity	b. Prior to release of each batch	b. $<10^{-11}$ μCi/cc
	c. Tritium	c. Prior to release of each batch	c. $<10^{-6}$ μCi/cc
9. Unit Vent Sampling	a. Iodine Spectrum (4)	a. Weekly	a. $<10^{-10}$ μCi/cc
	b. Particulates (4)		
	1) Gross Beta & Gamma Activity	1) Weekly	1) $<10^{-11}$ μCi/cc
	2) Gross Alpha Activity	2) Quarterly on a sample of one week duration	2) $<10^{-11}$ μCi/cc

4.4.3 Hydrogen Purge System

Applicability

Applies to testing Reactor Building Hydrogen Purge System.

Objective

To verify that this system and components are operable.

Specification

4.4.3.1 Operating Tests

An in-place system test shall be performed annually using the written emergency procedures. These tests shall consist of visual inspection, hook-up of system to either the Unit 1 or Unit 2 reactor building, a flow measurement using flow instruments in the portable purging station and pressure drop measurements across the filter bank. Flow shall be design flow or higher, and pressure drops across the filter bank shall not exceed two times the pressure drop when new. Fan motors shall be operated continuously for at least one hour, and valves shall be proven operable. This test shall demonstrate that under simulated emergency conditions the system can be taken from storage and placed into operation within 48 hours.

4.4.3.2 Filter Tests

Annually, leakage tests using DOP on HEPA units and Freon-112 (or equivalent) on charcoal units shall be performed at design flow on the filter. Removal of 99.5% DOP by each entire HEPA filter unit and removal of 99.0% Freon-112 (or equivalent) by each entire charcoal absorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing.

4.4.3.3 H_2 Detector Test

Hydrogen concentration instruments shall be calibrated annually with proper consideration to moisture effect.

Bases

The purge system is composed of a portable purging station and a portion of the penetration room ventilation system. The purge system is operated as necessary to maintain the hydrogen concentration below the control limit. The purge discharge from the Reactor Building is taken from one of the penetration room ventilation system penetrations and discharged to the unit vent. A suction may be taken on the Reactor Building via isolation valve PR-7 (Figure 6-5 of the FSAR) using the existing vent and pressurization connections.

The purge rate is controlled through the use of a portable purging station (Insert, Figure 14A-5.1 of the FSAR). The station consists of a purge blower, dehumidifier, filter train, purge flowmeter, sample connection and flowmeter and associated piping and valves.

The blower is a rotary positive type rated 60 scfm. The dehumidifier consists of two redundant heating elements inserted in a section of ventilation duct. The function of the dehumidifier is to sufficiently increase the temperature of the entering air to assure 70 percent relative humidity entering the filter train with 100 percent saturated air entering the dehumidifier. The purpose of the dehumidifier is to assure optimum charcoal filter efficiency. Heating element control is provided by a thermoswitch. Humidity indication is provided downstream of the heating elements by a humidity readout gage. The filter train provides prefiltration, high efficiency particulate filtration and charcoal filtration. The filter train assembly is identical in design to the waste gas filter train assembly which is rated at 200 scfm, thus conservatively capable of performing the assigned function. Face velocity to the charcoal filter is very low. The charcoal filter is composed of a module consisting of two inch deep double tray carbon cells. The purge flow to the unit vent is metered using a 0-60 scfm rotometer. The purge sample flow is metered using a 0-12 scfm rotometer. Both of these rotometers have an accuracy of $\pm$ two percent of full scale, and each has remote readout capability. The purge discharge rate is controlled by a blower discharge throttling valve. The purge sample activities can be collected, counted and analyzed in the radio-chemistry laboratory. Makeup air to the Reactor Building is supplied by a compressed air system connection to one of the aforementioned existing vent and pressurization connections.

That portion of the penetration room ventilation system piping and valves which is used as a part of the purge system is permanently installed and is designed for seismic loading through the existing vent and pressurization connections. The remainder of the purge system is the portable purging station which is stored in an area where an earthquake will not damage it. Following a LOCA, there is adequate time before purging is required to permit checkout of the portable purging station and to optimize the system operation to minimize the total dose to the public.

<u>References</u>
FSAR Section 14A

LIST OF EFFECTIVE PAGES
FSAR APPENDIX 4B

Stress Analysis - Reactor Coolant System

Page	Revision	Page	Revision
LOEP-- 1 of 2	Rev. 26	4B-18	Original
LOEP-- 2 of 2	Rev. 26	4B-19	Original
Cover Sheet Appendix 4B	Rev. 3	4B-20	Original
4B-i	Rev. 10	4B-21	Original
4B-ii	Original	4B-22	Original
4B-iii	Rev. 10	4B-23	Original
4B-1	Rev. 9	4B-24	Original
4B-2	Original	4B-25	Original
4B-3	Original	4B-26	Original
4B-4	Original	4B-27	Original
4B-5	Original	4B-28	Original
4B-6	Rev. 10	4B-29	Original
4B-7	Original	4B-30	Original
4B-8	Original	4B-31	Original
4B-9	Original	4B-32	Original
4B-10	Original	4B-33	Original
4B-11	Original	4B-34	Original
4B-12	Original	4B-35	Original
4B-13	Original	4B-36	Original
4B-14	Original	4B-37	Original
4B-15	Original	4B-38	Original
4B-16	Original	4B-39	Original
4B-17	Original	4B-40	Original

Stress Analysis - Reactor Coolant System

APPENDIX 4B

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

(At Rear of Section)

4B.1 INTRODUCTION

This report contains the following categories of information:

a. Pertinent information on the seismic design of the reactor coolant system.

b. A tabulation of reactor coolant piping stresses calculated by the static approach at the most critical locations.

c. A description of the type and location of each major component support analyzed, its design and the seismic amplification associated with the location of the support in the building.

d. An evaluation of results tabulated in (b) above.

e. A correlation between a free-standing spacial analysis of the nuclear steam system and a planar analysis considering building-loop interaction.

4B.2 SUMMARY AND CONCLUSION

A three dimensional model of the reactor coolant system was used to determine seismic mode shapes, frequencies, and inertia loads. The inertia loads along with thermal and dead load information was input to a piping flexibility program to obtain stresses and deflection. The reactor coolant system was considered uncoupled from any internal building structures. The resulting stresses were found to be within allowable limits. The maximum calculated piping stress for the design basis earthquake and the maximum hypothetical earthquake (considering dead loads and pressure) are, respectively, 24,968 psi and 27,269 psi. The allowable stresses for these two conditions are 27,600 psi and 33,120 psi, respectively.

As was pointed out in the above paragraph, the seismic analysis for the reactor coolant system was based upon a free standing system, i.e., no coupling. However, the steam generator will have lateral support at the upper elevation for hot plant conditions and the significance of this coupling was evaluated. A planar model of the building secondary shield wall, steam generator, reactor vessel, and 36" pipe was analyzed. Results show that coupling of the reactor coolant system to the building will not cause greater seismic responses or higher resulting pipe stresses than a free standing system.

The supports of the major components of the reactor coolant system were examined for effect of reaction loads resulting from dead weight, restrained thermal expansion, and seismic excitation. Results show that support integrity is not jeop-
9. ardized by any of the above loadings. (Reference Supplement 6 for Oconee 1.)

4B.3 ANALYSIS OF REACTOR COOLANT SYSTEM

4B.3.1 SCOPE OF ANALYSIS

The reactor coolant system consists of the reactor vessel, coolant pumps, steam generators, pressurizer, and interconnecting piping. For the purpose of seismic analysis, however, the reactor coolant system is considered to consist of the following:

a. Reactor vessel

b. One steam generator

c. Three interconnecting reactor coolant pipes, one being the 36" reactor outlet and two 28" reactor inlet pipes.

d. One pump and motor assembly on each 28" inlet pipe.

e. One 10" pressurizer surge line.

f. One 2-1/2" pressurizer spray line.

Reactor coolant system components are designated as Class 1 equipment and are designed to maintain their functional integrity during earthquake. The loading combinations and corresponding design stress criteria for pressure boundaries of both vessels and piping are given below. A discussion of each of the cases of loading combinations follows:

Case 1 - Design Loads Plus Design Earthquake Loads - For this combination, the reactor must retain operating capability; therefore, all components excluding piping are designed to Section III of the ASME code for Nuclear Vessels. The reactor coolant piping is designed according to the requirements of USAS B31.7. The S_m values for all components, excluding bolting, are those specified in Table N-421 of the ASME Code.

Case 11 - Design Loads Plus Maximum Hypothetical Earthquake Loads - In establishing stress levels for this case, a "no-loss-of-function" criterion applies, and higher stress values than in Case 1 can be allowed. The multiplying factor of (1.2) has been selected in order to increase the code-based stress limits and still insure that for the primary structural materials, i.e., 304 SST, 316 SST, SA302B, and SA212B, and SA106C, an acceptable margin of safety will always exist. A discussion of the adequacy of these margins of safety is given in B&W Topical Report BAW-10008 Part 1. The S_m value for all components are those specified in Table N-421 of the ASME Code.

Reactor coolant system seismic forces are defined in terms of an acceleration spectrum. An acceleration spectrum is an envelope of expected accelerations over a range of expected natural frequencies. A family of acceleration spectrum curves for various damping values was constructed. For piping systems the spectrum curve of interest is associated with 1/2% of critical damping.

Pursuant to the applicable codes, the following additional sources of loading must be considered in the seismic design of the reactor coolant system:

a. Pressure

b. Dead load

c. Thermal load

4B.3.2 METHOD OF ANALYSIS

4B.3.2.1 DESCRIPTION OF ANALYTICAL MODELS

4B.3.2.1.1 Seismic Analysis

The seismic analysis of the reactor coolant system is a dynamic analysis based on the theory and basic procedures outlined in References 1 and 5. The reactor coolant system as a basic piping structure is idealized in the analytical model by a concentrated mass system or structure. In this idealization, the mass of the structure is considered to be lumped or concentrated at a certain finite number of points. The resistance to deflection is caused by elastic members having strength and stiffness but are weightless.

The dynamic response of a concentrated-mass system having multiple degrees of freedom involve first determining the frequencies and shapes of the normal modes of vibration. The dynamic response of the system to a given dynamic load is evaluated by using the frequencies, mode shapes, modal participation factors and a dynamic load given as an acceleration-frequency spectrum.

Schematic drawings of the analytical model are shown on Figures 4B-1 and 4B-2. The model starts at the base of the reactor vessel support skirt Point A1-160. It extends through the reactor vessel, the three main reactor coolant pipes to one steam generator, then to the base of the steam generator support skirt at point A6-101. Only one steam generator was included in the model on the basis of symmetry. The only external anchors or restraints in the system are at the base of the support skirts, points A1-160 and A6-101. No intermediate reactor building to reactor coolant system dynamic coupling is included. The significance of building coupling to reactor coolant system is discussed in section 4B-5.

The mass of the system was considered to be acting or lumped at 16 points in the following manner:

a. The reactor vessel and steam generator are each represented by two masses.

b. Each reactor coolant pump motor is represented as one mass.

c. The reactor coolant pump on Branch 23 is represented by a single mass point.

d. The 36" reactor coolant pipe is represented by three mass points.

e. Each 28" reactor coolant pipe is represented by two mass points, one on each side of the pumps.

The lumped masses are connected by eighteen elastic members as follows:

a. Both the steam generator and reactor vessel are each represented by two masses connected by two elastic members which represent the vessel skirt and shell.

b. Each pump motor is connected to the pump by one member.

c. The remainder of members are the various straight lengths and bends in the three reactor coolant pipes.

Each pump body and the upper half of the steam generator were assumed to be rigid members in the system.

The dynamic analysis is done in three steps by computer programs:

a. The first program calculates the flexibility matrix for each branch in the system where each branch begins and ends at a mass point on anchor. The basic theory and equations are shown in References 3 and 4 with the additional consideration of flexibility due to axial and cross shearing deformations and using elbow flexibility factors, k, from USAS B31.1. The branch flexibility matrices are referred to the system coordinate axis at point A-919. They define the displacement of three deflections and three rotations resulting from forces and moments, i.e., for six degrees of freedom per mass point.

b. The second program requires the basic input of the branch matrices, the value of all the masses and the seismic spectrum. The program first inverts each flexibility matrix to obtain a 6 x 6 stiffness matrix with the translational elements in the upper left 3 x 3 submatrix. An overall stiffness matrix is then assembled having 6N x 6N elements where N is the number of modal branch or mass points. This results in a 96 x 96 matrix. The rows and columns are then arranged such that all the translational elements are in the upper left 3N x 3N or 48 x 48 submatrix. For this analysis, the rotational inertia is assumed to be zero; therefore, the rotational degree of freedom is eliminated and only the upper left 48 x 48 submatrix is needed. (This means the final stiffness matrix used in the motion equations represents three degrees of freedom per mass point as translations in the X, Y, and Z directions of the coordinate axes. Likewise, the inertia effects are three forces at each mass point.) The stiffness matrix and mass matrix provide the information needed for the next step, the program step of calculating the eigenvalues and natural frequencies. The Wilkinson method was used for this solution. The eigenvectors (mode shapes) corresponding to each frequency are then determined. The program lastly calculates for each mode the equivalent static loads at each mass point, using the seismic acceleration response spectra. The basic theory and equations used for mode shapes, participation factors and equivalent static forces are given in Reference 1. The equivalent static force at each mass point for all modes is calculated by taking the largest absolute value of the inertia load at each mass point and adding to it the root mean square of the remaining modes.

c. The third program is a piping flexibility program. It calculates the forces and moments at all the branch points using the equivalent static forces from step b.

4B.3.2.1.2 Dead Load Analysis

The analytical model for the dead load analysis is the same as for the seismic analysis and can also be represented by Figures 4B-1 and 4B-2. In this case, the weight of the piping is placed at the mass points as concentrated forces. The program used to calculate the system forces and moments is the piping flexibility program referred to in the previous section. The output of forces and moments were used to calculate stresses to be combined with other loadings as shown in Section 4B.3.2.3.

4B.3.2.1.3 Thermal Flexibility Analysis

As was the case with dead load, the analytical model used for thermal expansion analysis was the same as for the seismic analysis. Likewise, the piping flexibility program was used with input of thermal expansion effects. This model is valid for the 28" pipes entering the steam generator near the bottom which is anchored. The results of the thermal expansion analysis show center line horizontal deflections at this location, C901, of 0.010 in. in the X direction and 0.008 in. in the Z direction. Therefore, the steam generator can be assumed to be horizontally restrained at this location. However, the thermal expansion motion near the top of the generator (at the LOCA restraint) is shown from the analysis to be about 0.250". Since the generator will be restrained in the hot condition, this motion is excessive for a valid analysis of the 36" pipe which enters at the top of the generator.

An alternate thermal analysis was made of the 36" line from the nozzle attachment to the steam generator to the nozzle attachment to the reactor vessel. The analytical model is shown on Figure 4B-3. This model also includes the 10" pipe surge line. The anchor point motions at "A" and "B" are on the basis of no horizontal center line movement of either the steam generator or the reactor vessel. This will meet the requirement of the restrained generator.

The program used for the analysis of the 36" pipe is a direct adaptation of the method described in Reference 4. The output of forces and moments at branch points was used to calculate stresses to be combined with other loadings as shown in Section 4B.3.2.3.3.

4B.3.2.2 STRESS ANALYSIS OF REACTOR COOLANT SYSTEM PRESSURE VESSELS

The reactor vessel, steam generators and pressurizer and pumps are designed to meet the requirements of the ASME Section III Nuclear Vessels Code for Class A vessels. A complete stress analysis is performed on each vessel, in accordance with ASME Section III, to assure that the vessel will meet the stress limits and criteria specified by the Code and for seismic loading conditions.

4B.3.2.3 STRESS ANALYSIS OF REACTOR COOLANT PIPING

Stress calculations made at various locations throughout the piping system are done in accordance with the Nuclear Power Piping Code USAS B31.7. Several locations, as noted on the table of results, were analyzed in accordance with Paragraph 1-705.1 of B31.7. The remaining points were analyzed in accordance with Appendix F of B31.7. Primary and primary plus secondary stresses were calculated at each location and comparison made to 1.5 Sm and 3.0 Sm, respectively. In addition, primary stresses resulting from maximum hypothetical earthquake, dead load and pressure effects were compared to the value 1.8 Sm (1.2 x 1.5 Sm). The highest stress at any location (branch point

No. 10) was found to be 27,269 psi which is below the allowable value of 33,120 psi. For the points analyzed in accordance with Paragraph 1-705.1 the loading technique is as follows. The seismic moment is evaluated by adding the largest absolute value of either horizontal (X or Z) earthquake reaction plus the rms of the remaining modes to the absolute value of the vertical earthquake reaction. This value is then added to the dead weight moment and pressure loading to calculate primary stresses. Primary plus secondary stresses are calculated by doubling the seismic loading used for primary stresses and also including thermal loadings.

For the analysis described in Appendix F of USAS B31.7. the seismic loads are combined based on Table 4B-1 to attain the worst case condition. The dead weight and thermal loads are used as indicated in Table 4B-1 to calculate primary and primary plus secondary stresses.

Table 4B-2 through 4B-5 give pipe data, forces and moments, and final pipe stresses for various locations throughout the piping system.

4B.3.2.4 STRESS ANALYSIS OF PRESSURIZER SURGE LINE PIPING

Stress calculations made at various locations throughout the Surge Line were performed in accordance with the Nuclear Power Piping Code, USAS B31.7. Pursuant to the code, seismic, thermal, pressure and cyclic loadings were considered in the analysis.

The geometry, support conditions, joint and component descriptions are all shown in Figure 4B-6. The diameter thickness and material designation for the pipe is shown in Table 4B-2b.

The results of this analysis are presented in Table 4B-5c. All points were
10. analyzed in accordance with Paragraph 1-705 of B31.7. They indicate that the subject pipe meets all design criteria. At joint 2 the load combinations under Equation 9 do show a 0.9 percent overstress. However, this overstress is insignificant in that an Appendix F detailed analysis readily eliminates this condition. At joints 2, 3 and 12,* the Equation 10 stresses exceed the allowable 3.0 S_m. The USAS B31.7 code requires that the simplified elastic-plastic discontinuity analysis represented by Equations 12 and 13 be used in conjunction with Equations 10 and 11 under this condition. For the appropriate loading conditions, Equation 12 is satisfied, and the cumulative usage factors calculated with Equation 13 stresses meet the code requirements of unity.

*Joints 2 (C01), 3 (C01), 12(C04)

4B.4 REACTOR COOLANT SYSTEM COMPONENT SUPPORTS

4B.4.1 DESCRIPTION OF SUPPORTS

Both the reactor vessel and steam generator are supported by a cylindrical skirt rigidly attached to the vessels and bolted to the foundation by means of an integral base plate. The skirts are designed in accordance with ASME Section III and criteria stated in Section 4B.3.1 of this report. Lateral support is provided for the steam generator at the upper tube sheet level by means of a structural tie to the secondary shield wall.

The pressurizer is supported by 8 support pads spaced symmetrically around the circumference of the vessel. The pads are designed in accordance with Section III and criteria stated in Section 4B.3.1 of this report.

The reactor coolant piping is self-supporting with respect to dead weight, seismic, and thermal loading. The reactor coolant pumps are partially supported by hanger rods which are designed to support the dead weight of the pump motor, with the remainder of the dead weight of the pump being supported by the piping. To reduce seismic deflection, the pumps are supported laterally at the motor by means of hydraulic suppressors connected to the secondary shield wall.

4B.4.2 METHOD OF ANALYSIS

4B.4.2.1 CALCULATION OF FOUNDATION LOADS FOR REACTOR VESSEL AND STEAM GENERATOR

The steam generator and reactor vessel supports were designed for reaction loads from dead weight of the vessels, restrained thermal expansion of the reactor coolant piping, and seismic excitation.

Preliminary results were calculated as a part of the reactor coolant piping analysis. These seismic results indicated that the vessels act essentially independent of the piping. To examine each component in more detail, the steam generator and reactor vessel foundation loads were recalculated with each treated as an isolated component, independent of the reactor coolant piping.

The dynamic characteristics of the reactor vessel and steam generator and the loads on their supports were determined using a detailed lumped-mass dynamic model of each component. The models included one lateral degree of freedom per mass, with the discrete mass points connected by flexible beam segments. An additional rotational spring was included at the base of the models to represent the flexibility of the anchor bolts and concrete foundation beneath the vessels. In addition, the steam generator model assumed no connection to the secondary shield wall.

The seismic forces on the supports were calculated using the response spectra approach. Sufficient modes were included in the model to simulate the behavior of the actual structure. All the modes included in the model were calculated, and were combined as the square root of the sum of the squares of all the individual modal contributions.

Loads on the supports due to thermal expansion of the piping and dead weight which were calculated as a part of the piping analysis were incorporated into the support design also.

The results of foundation loads due to dead weight, thermal expansion, and seismic loadings for all major components are shown in Table 4B-6.

4B.4.2.2 CALCULATION OF FOUNDATION LOADS FOR PRESSURIZER

The first mode natural frequency of the pressurizer is greater than 30 cps and the vessel can be considered rigid. For rigid systems the maximum acceleration at the point of support can be considered to act at the center of gravity of the vessel and a static approach used. OK

Static loads equal to 0.2 x Full Wet Weight were applied at the center of gravity of the vessel in both vertical and horizontal directions. These loads were assumed to act simultaneously.

The equivalent horizontal shear and overturning moment at the vessel support level was found and used in the design and analysis of the support. The vessel wall was analyzed for local loading, from the attached support, by means of a method developed by P. P. Bijlaard. The resulting stress intensities were compared to stress allowables specified in ASME Section III and criteria stated in Section 4B.3.1.

The static analysis method, using 0.2g acceleration loads, is conservative. The 0.2g acceleration is greater than the accelerations given in the acceleration spectra for the various elevations of equipment supports. For the pressurizer supported at Elev. 821 ft. the spectra results give an acceleration of 0.06g for the design basis earthquake.

Loads due to thermal expansion were calculated as part of the piping analysis and included in the support design.

4B.4.2.3 ANALYSIS OF REACTOR VESSEL AND STEAM GENERATOR SUPPORTS

The reactor vessel and steam generator support skirts and support skirt flanges are designed and analyzed using procedures described in Chapter 10, Section 1, of Reference 6. That procedure is used to determine the tensile stress in the anchor bolts, the bearing stress on the support skirt flange and the location of the neutral axis of bending on the bolt-flange mechanism.

The skirt-flange mechanism was statically analyzed for the applied forces and moments due to seismic loading on the vessel, considering a free-standing vessel (see Table 4B-6).

The support skirt flange and foundation is assumed to be rigid. In regard to the reactor vessel, effects of anchor bolt pretension on the bending moment capacity of the support skirt were evaluated. With no anchor bolt pretension, the location of the neutral axis is found by trial and error methods so that the difference between the first moment of the bolt tension area and first moment of the flange compression area about the neutral axis is less than 5 percent of the smaller value. Increasing values of applied anchor bolt pretension result in less shift of the neutral axis.

The anchor pretension load necessary to prevent any separation of the support skirt flange from the foundation is the required load which will result in no shift of the neutral axis. In that case the neutral axis is located on the centerline of the vessel flange.

For a typical seismic load condition on the vessel, the support skirt flange was analyzed for flange bearing stress, anchor bolt loads, and location of neutral axis. Once the neutral axis was located, giving consideration to anchor bolt pretension loads, the flange, skirt, gusset mechanism was analyzed for applied tensile, compressive, and shear loads resulting from bending using methods from engineering mechanics.

The allowable stress criterion specified in Section 4B.3.1 of this report was used where applicable.

4B.5 EVALUATION OF SEISMIC ANALYSIS OF REACTOR COOLANT SYSTEM FOR EXISTING CONFIGURATION

The objective of this evaluation is to show by qualitative analysis that the final stresses from an uncoupled analysis described in Section 4B.3.2 would be conservative compared to results from a building NSS coupled analysis.

The dynamic seismic analysis was for a NSS uncoupled from any containment building structures. As stated in Section 4B.4.1 the steam generator will have lateral support for hot plant conditions and the significance of coupling was evaluated.

The effect of building coupling was evaluated from the results of a dynamic analysis on simplified models of the NSS and building secondary shield walls, as shown in Figures 4B-4 and 4B-5. In these models the NSS weight is lumped as: The steam generator mass points 7 and 8, the 36" reactor coolant pipe mass points 8, 9, and 10 and the reactor vessel masses 10 and 11. The elastic members for the NSS are: steam generator support skirt No. 7; steam generator No. 8; 36" pipe numbers 9, 10, 11, 12, 13, 14 and 15; reactor vessel No. 16; reactor vessel support skirt No. 17. This portion of the NSS is basically the same as modeled on Figures 4B-1 and 4B-2. The containment building secondary shield wall is represented by mass points 1 through 5 and elastic members 1 through 6. The lateral tie between the building and the NSS is elastic member 18.

These models were analyzed by a B&W computer program. This program performs a normal mode vibration calculation and applies a base motion spectra in a similar manner as the programs for the system described in Section 4B.3.2. This program is limited to single degree of freedom as translation per mass point.

Tables 7 and 8 show a summary of the results from the program. Table 4B-7 is a comparison of the absolute model summation of the inertia forces as effective static forces and deflections at each mass point.

Results from the ZY direction show that the building mass points will have an increase in effective static forces when coupled with NSS. Conversely, the NSS mass points have a decrease in effective static forces. The relative displacement between the steam generator mass points was reduced from 0.137 inches to 0.010 inches. The average relative displacement between piping mass points was reduced from 0.116 inches to 0.008 inches.

Results from the XY direction show an overall decrease in the mass point effective static forces for the NSS, with two individual points, one on the generator and one on the pipe, being larger. The average forces on the pipe masses, however do show a decrease when coupled. Similarly, the relative displacements show a decrease as in the ZY direction.

Table 4B-8 is a comparison of the internal forces and moments on the system elastic members. For the 36" pipe there is a marked decrease in values for coupled vs uncoupled in the ZY direction. In the XY direction, there is also an appreciable overall decrease in values with the exception of a local area near the juncture of element 10 and 11 at the tangent of the 180 degree bend. Here the coupled moment is greater than the uncoupled analysis moment. Fortuitously, this location is at a low stress area. To quantitatively evaluate this, the stresses at this location were calculated using an increase of seismic moment by the ratio of the coupled vs uncoupled moments. The stresses, when compared to the design calculations, show a change of less than one percent. This further indicates that the seismic forces at this location are relatively low.

The stiffness of the tie was selected on a preliminary basis. To evaluate the effects of increased rigidity, a computer run was made using a value 100 times stiffer for member 18. The results did not show any unfavorable changes in effective static forces or internal forces and moments on the piping.

The effect of rotational spring constants at the base of the reactor vessel and steam generator were examined for the coupled system. Nominal values were selected to represent the strain in the concrete and anchor bolts due to seismic loading. Results of this analysis show that inclusion of spring constants in the dynamic model actually decrease the element moments and forces slightly.

It can be concluded that the coupling of the NSS to the building will not cause greater seismic responses or higher resulting piping stresses.

Not sufficient info. — Should include torsional response in math model.

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(1) Norris, et al, "Structural Design for Dynamic Loads," McGraw-Hill Co., 1959.

(2) Brock, J. E., "A Matrix Method for Flexibility Analysis of Piping Systems," ASME Journal of Applied Mechanics, December 1952.

(3) Chen, L. H., "Piping Flexibility Analysis by Stiffness Matrix," ASME Journal of Applied Mechanics, December 1959.

(4) "Design of Piping System," The M. W. Kellogg Company, Second Edition, 1956.

(5) "Nuclear Reactors and Earthquakes," TID-7024, Chapters 5 and Appendix E, August 1963.

(6) Bornwell, R. and Young, G., "Process Equipment Design," John Wiley and Sons, 1959.

Table 4B-1
Loading Combinations

Earthquake Combination	Primary Stresses Design Pressure Single Amplitude	Primary + Secondary Stresses Operating Pressure and Temperature Double Amplitude Seismic Loads	
	Mid-Surface	Outside Surface	Inside Surface
+Y + X	1	9	17
+Y - X	2	10	18
+Y + Z	3	11	19
+Y - Z	4	12	20
-Y + X	5	13	21
-Y - X	6	14	22
-Y + Z	7	15	23
-Y - Z	8	16	24

Table 4B-2a
Pipe Data

(Reference Branch No. to Figures 4B-1 and 4B-2)

Branch Number	From	To	OD	Thickness	Material of Straight Pipe	Material of Curved Pipe	Design Temperature, F
1[a]	A6-101	C-901	138.0	1.5			650
2[a]	C-901	D-902	138.0	10.0			650
3	D-902	M-933	44.0	3.75	SA-106 Gr. C	SA-516-70	650
4	M-933	107	44.0	3.75		SA-516-70	650
4	107	N-904	42.75	3.3125	SA-106 Gr. C		650
5	N-904	0-905	42.75	3.3125	SA-106 Gr. C		650
6	0-905	111	42.75	3.3125	SA-106 Gr. C		650
6	111	113	44.0	3.75	SA-106 Gr. C	SA-516-70	650
6	113	A-906	42.75	3.3125	SA-106 Gr. C		650
7[b]	A2-115	G-907	0.554	0.027			650
8[b]	G-907	H-908	33.5	2.75			650
9	H-908	F-909	33.5	2.6875	SA-106 Gr. C		650
10	F-909	117	33.5	2.6875	SA-106 Gr. C		650
10	117	C-901	33.75	3.125		SA-516-70	650

Note: Material at connections of the pumps is SA-376 TP-316.

(a) Steam generator.

(b) Reactor coolant pump.

Table 4B-2b
Pipe Data

(Reference Branch No. to Figures 4B-1 and 4B-2)

Branch Number	From	To	OD	Thickness	Material of Straight Pipe	Material of Curved Pipe	Design Temperature, F
11	P-920	E-910	33.75	3.125	SA-106 Gr. C	SA-516-70	650
12	E-910	A-906	33.5	2.6875	SA-106 Gr. C	SA-516-70	650
13	C-901	145	33.75	3.125		SA-516-70	650
13	145	L-917	33.5	2.6875	SA-106 Gr. C		650
14	L-917	J-915	33.5	2.6875	SA-106 Gr. C		650
15[b]	J-915	I-916	33.5	2.75			650
16[b]	A4-150	I-916	0.554	0.027			650
17	J-915	K-912	33.75	3.125	SA-106 Gr. C	SA-516-70	650
18	K-912	A-906	33.5	2.6875	SA-106 Gr. C	SA-516-70	650
19[c]	A1-160	B-918	179.5	2.0			650
20[c]	B-918	A-906	179.5	10.0			650
21	A3-201	0-905	10.75	1.0	SA-376 TP-316	SA-403 Gr. WP-316	670
22	A7-215	P-920	2.875	0.375	SA-376 TP-316	SA-403 Gr. WP-316	670
23	H-908	P-920	33.5	2.6875	SA-106 Gr. C		650

Note: Material at connections of the pumps is SA-376 TP-316.

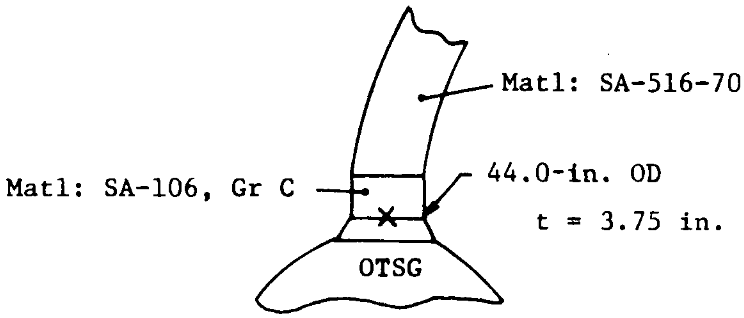
[b] Reactor coolant pump.

[c] Reactor vessel.

Table 4B-3a
Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 103 Branch Number: 3

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	13,944	4,773	999
Y — Direction earthquake	1,990	19,050	7,550
Z — Direction earthquake	2,760	69,374	-140
Dead weight	-670	-78,045	-2,290
Thermal expansion, 100% power	5,776	277,279	-117,036

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-13,687	13,658	-130,913
Y — Direction earthquake	-72,595	12,279	-24,314
Z — Direction earthquake	-281,764	-1,433	-89,890
Dead weight	415,637	-9,222	51,337
Thermal expansion, 100% power	-812,734	-77,120	-220,619

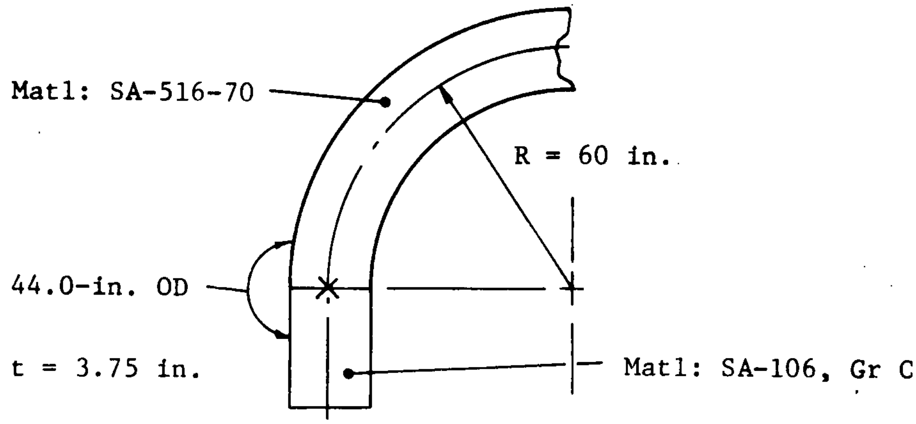
All moments are in ft-lb.

Table 4B-3b

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 104 Branch Number: 3

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-13,944	-4,773	-999
Y — Direction earthquake	-1,990	-19,050	-7,550
Z — Direction earthquake	-2,760	-69,374	140
Dead weight	670	78,045	2,290
Thermal expansion, 100% power	-5,776	-277,279	117,306

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	15,184	-13,658	109,998
Y — Direction earthquake	83,924	-12,279	21,331
Z — Direction earthquake	281,554	1,433	85,749
Dead weight	-419,074	9,222	-50,330
Thermal expansion, 100% power	681,069	77,120	214,121

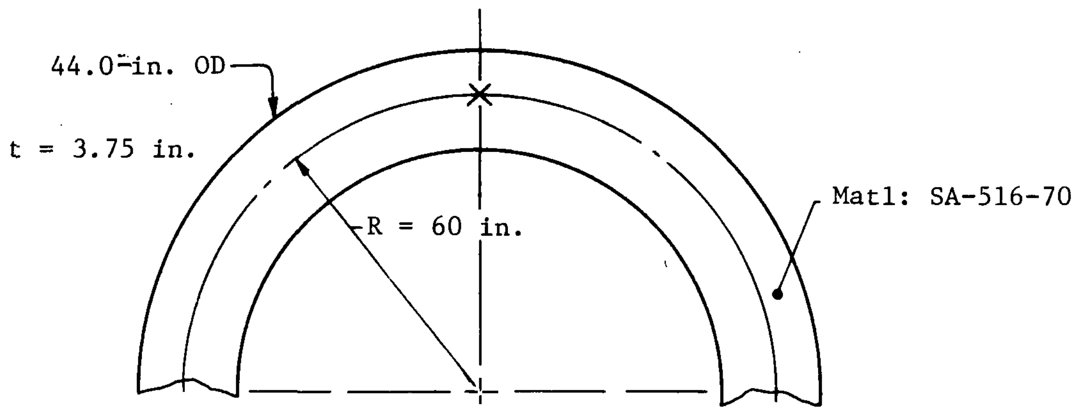
All moments are in ft-lb.

Table 4B-3c

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 106 Branch Number: 3

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-13,944	-4,773	-999
Y — Direction earthquake	-1,990	-19,050	-7,550
Z — Direction earthquake	-2,760	-69,374	140
Dead weight	670	78,045	2,290
Thermal expansion, 100% power	-5,776	-277,279	117,036

All forces are in lb.

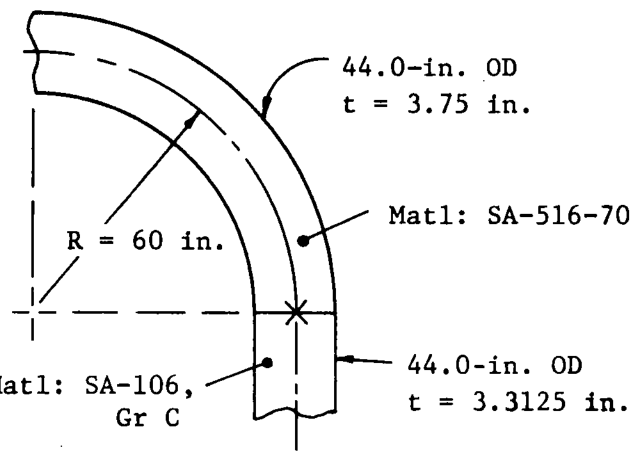
	M_x	M_y	M_z
X — Direction earthquake	-3,483	56,117	37,164
Y — Direction earthquake	27,242	2,508	-1,042
Z — Direction earthquake	-63,049	15,027	26,687
Dead weight	-43,645	4,400	3,941
Thermal expansion, 100% power	-1,278,824	29,287	4,072

All moments are in ft-lb.

Table 4B-3d
Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 107 Branch Number: 4

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	100	-4,140	690
Y — Direction earthquake	-530	-16,235	-1,564
Z — Direction earthquake	-90	-65,760	14,960
Dead weight	670	38,070	2,290
Thermal expansion, 100% power	-5,776	-277,279	117,036

All forces are in lb.

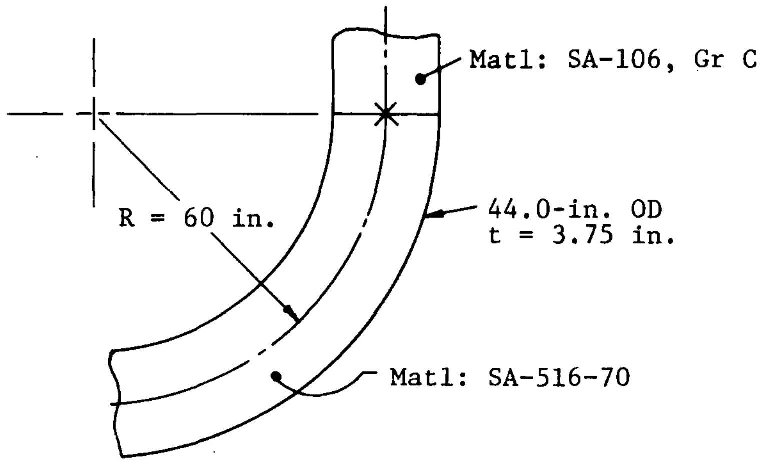
	M_x	M_y	M_z
X — Direction earthquake	-20,564	55,178	33,969
Y — Direction earthquake	-61,057	6,174	-8,964
Z — Direction earthquake	-314,190	5,706	-15,767
Dead weight	156,501	-422	25,416
Thermal expansion, 100% power	-2,068,156	-18,539	-148,191

All moments are in ft-lb.

Table 4B-3e
Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 111 Branch Number: 6

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	18,943	-1,650	2,368
Y — Direction earthquake	1,615	-6,744	19,187
Z — Direction earthquake	2,283	-49,840	30,345
Dead weight	753	-43,433	2,308
Thermal expansion, 100% power	3,074	-276,589	119,275

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	20,660	55,591	-142,542
Y — Direction earthquake	21,747	6,245	-8,637
Z — Direction earthquake	404,469	5,896	-36,072
Dead weight	240,675	-388	975
Thermal expansion, 100% power	2,030,379	127,201	-10,995

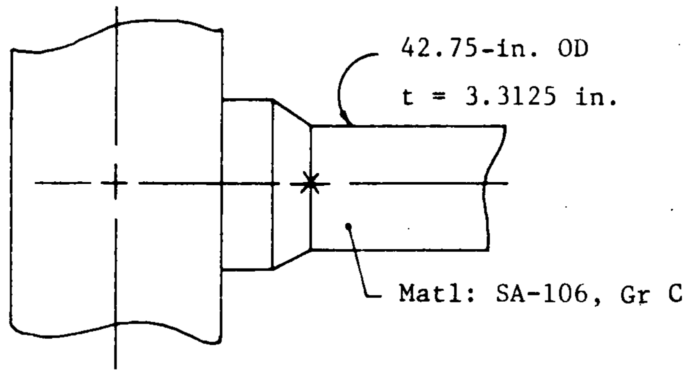
All moments are in ft-lb.

Table 4B-3f

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 114 Branch Number: 6

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-18,943	1,650	-2,368
Y — Direction earthquake	-1,615	6,745	-19,187
Z — Direction earthquake	-2,283	49,840	-30,345
Dead weight	-750	43,434	-2,308
Thermal expansion, 100% power	-3,074	276,585	-119,275

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-13,155	166,444	237,259
Y — Direction earthquake	-38,642	12,680	16,710
Z — Direction earthquake	27,986	20,867	47,489
Dead weight	256,872	9,219	2,792
Thermal expansion, 100% power	235,021	42,801	-111,831

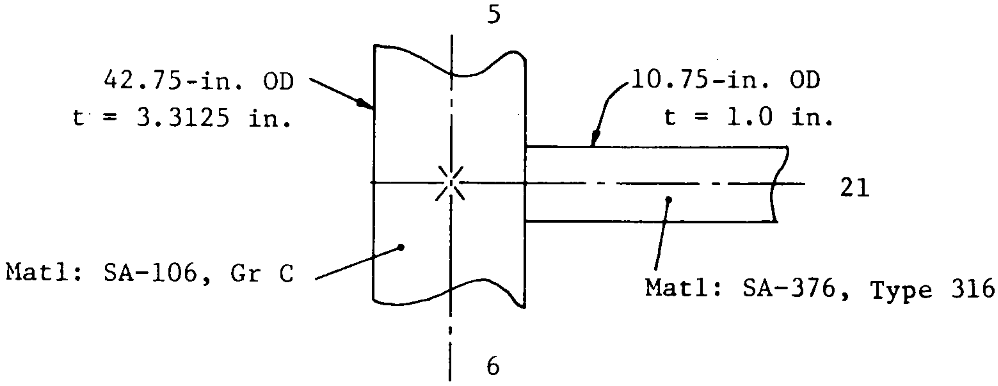
All moments are in ft-lb.

Table 4B-3g

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 905 Branch Number: 5

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-10,310	2,952	-1,644
Y — Direction earthquake	-584	11,872	-6,410
Z — Direction earthquake	-1,327	58,428	-25,277
Dead weight	-670	-424	-2,292
Thermal expansion, 100% power	5,776	277,279	-117,036

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-18,336	-55,178	122,729
Y — Direction earthquake	-2,697	-6,174	6,675
Z — Direction earthquake	-374,830	-5,706	33,840
Dead weight	-238,334	422	-1,449
Thermal expansion, 100% power	-1,935,450	18,539	-49,393

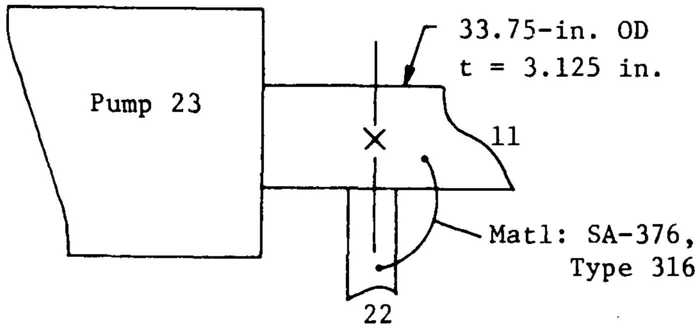
All moments are in ft-lb.

Table 4B-3h

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 920 Branch Number: 11

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	37,826	1,203	25,864
Y — Direction earthquake	35,729	5,788	16,537
Z — Direction earthquake	18,282	-15,958	50,944
Dead weight	-21,265	-34,657	4,584
Thermal expansion, 15% power	-12,152	83,150	83,094

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-13,452	72,485	-176,833
Y — Direction earthquake	-58,293	96,430	-125,667
Z — Direction earthquake	101,233	-28,275	-70,975
Dead weight	317,323	-52,857	-39,347
Thermal expansion, 15% power	-1,352,862	-308,574	118,494

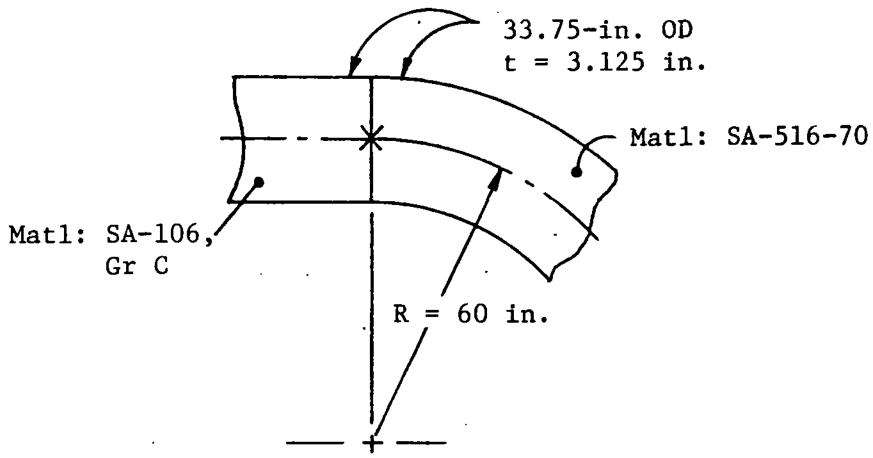
All moments are in ft-lb.

Table 4B-3i

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 133 Branch Number: 17

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	32,280	-23,456	39,188
Y — Direction earthquake	22,809	-3,554	12,973
Z — Direction earthquake	7,697	-20,709	57,048
Dead weight	19,851	-25,798	-3,995
Thermal expansion, 15% power	366	83,508	87,398

All forces are in lb.

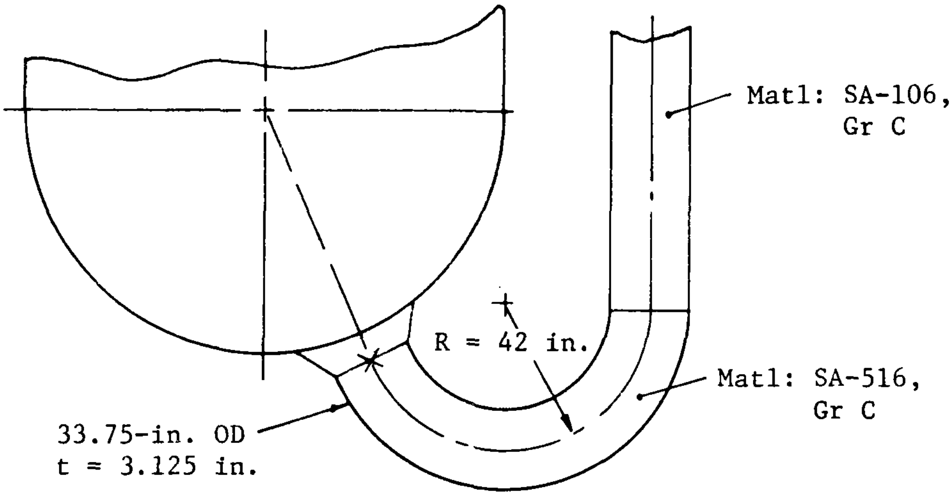
	M_x	M_y	M_z
X — Direction earthquake	158,589	52,326	234,438
Y — Direction earthquake	15,091	25,922	-154,652
Z — Direction earthquake	70,160	55,192	-100,626
Dead weight	190,746	-13,657	54,978
Thermal expansion, 15% power	-1,254,057	204,605	-204,562

All moments are in ft-lb.

Table 4B-3j
Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 140 Branch Number: 13

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-13,412	-47,872	425
Y — Direction earthquake	-16,853	-49,424	-1,352
Z — Direction earthquake	-11,365	-32,336	-3,964
Dead weight	19,851	237,655	-3,995
Thermal expansion, 15% power	367	94,103	87,398

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	46,913	230,772	114,347
Y — Direction earthquake	61,192	150,903	52,902
Z — Direction earthquake	30,478	92,399	42,066
Dead weight	-290,731	195,522	989,726
Thermal expansion, 15% power	776,013	-399,855	434,918

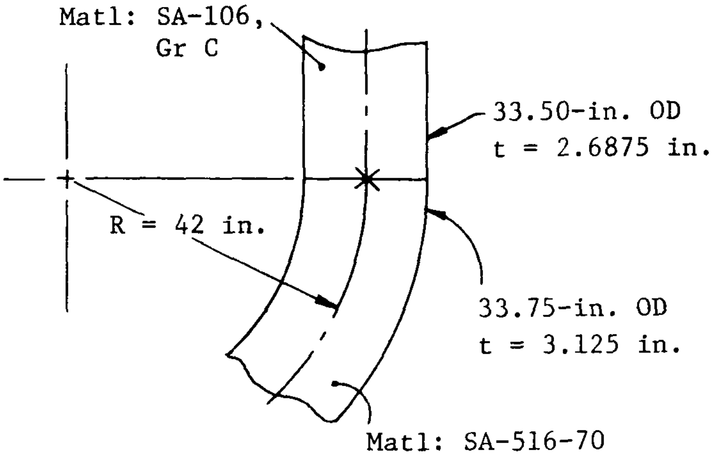
All moments are in ft-lb.

Table 4B-3k

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 145 Branch Number: 13

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	13,412	47,872	-425
Y — Direction earthquake	16,853	49,424	1,352
Z — Direction earthquake	11,365	32,336	3,964
Dead weight	-19,851	-237,665	3,995
Thermal expansion, 15% power	-367	-94,103	-87,398

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	60,964	-263,554	-402,742
Y — Direction earthquake	46,137	-179,543	-343,855
Z — Direction earthquake	32,783	-91,391	-231,660
Dead weight	-249,077	-124,587	547,596
Thermal expansion, 15% power	-788,665	-181,176	190,746

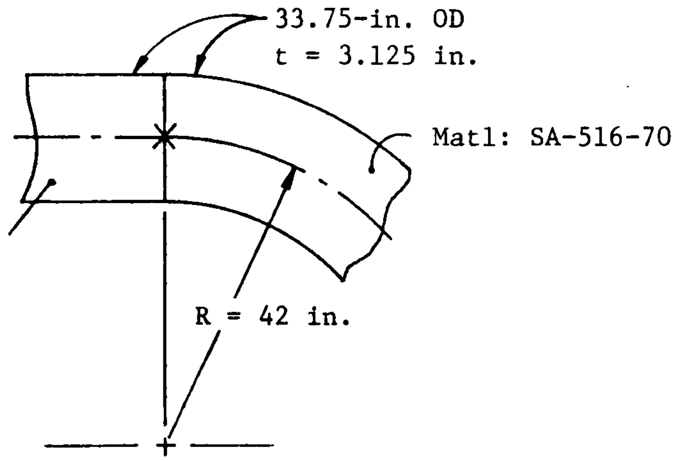
All moments are in ft-lb.

Table 4B-3ℓ

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 141 Branch Number: 17

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-32,280	23,456	-39,188
Y — Direction earthquake	-22,809	3,554	-12,973
Z — Direction earthquake	-7,697	20,709	-57,048
Dead weight	-19,851	25,798	3,995
Thermal expansion, 15% power	-366	-83,508	-87,398

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-20,684	308,315	336,708
Y — Direction earthquake	-18,815	235,623	232,860
Z — Direction earthquake	-26,983	9,019	118,110
Dead weight	125,767	248,263	2,722
Thermal expansion, 15% power	-31,104	-240,208	243,967

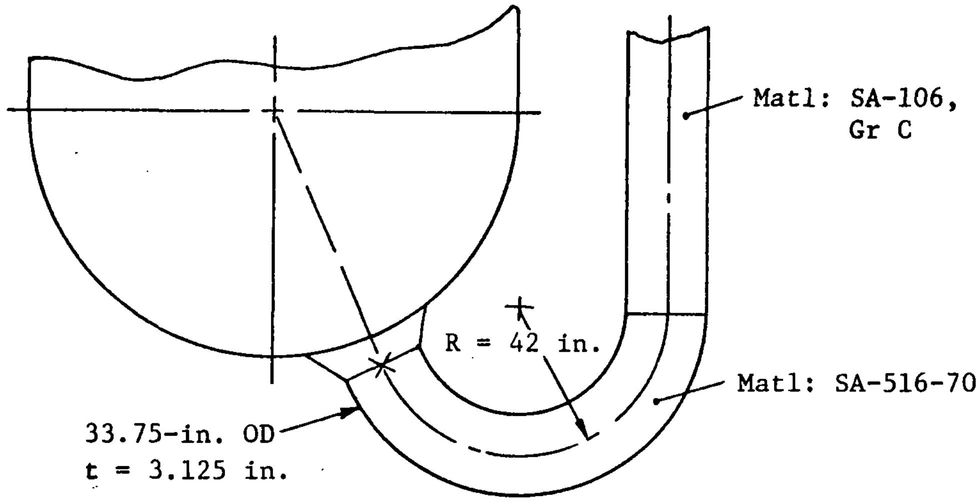
All moments are in ft-lb.

Table 4B-3m

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 122 Branch Number: 10

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-2,489	-24,233	-393
Y — Direction earthquake	-6,200	-41,770	-1,772
Z — Direction earthquake	-1,753	-31,972	740
Dead weight	-21,265	230,624	4,586
Thermal expansion, 15% power	-12,165	94,196	83,069

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-36,270	250,606	299,366
Y — Direction earthquake	-2,127	205,314	388,477
Z — Direction earthquake	15,103	133,892	209,246
Dead weight	-272,042	-160,745	-889,317
Thermal expansion, 15% power	728,833	356,660	-309,478

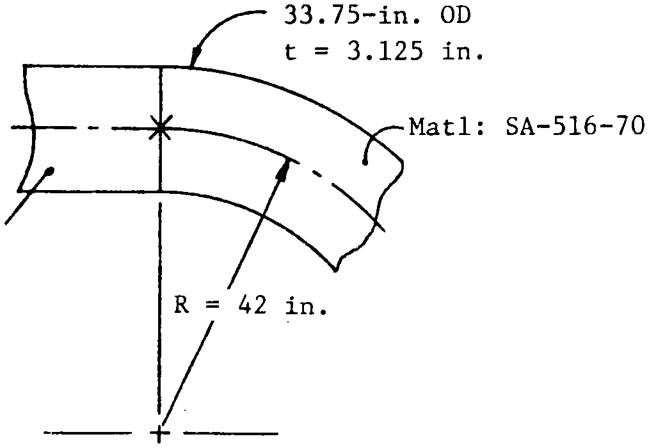
All moments are in ft-lb.

Table 4B-3n

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 123 Branch Number: 11

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	37,826	1,203	25,864
Y — Direction earthquake	-35,729	5,788	16,537
Z — Direction earthquake	18,282	-15,958	50,944
Dead weight	-21,265	-34,657	4,584
Thermal expansion, 15% power	-12,152	83,150	83,094

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-11,419	-3,129	-176,289
Y — Direction earthquake	-48,510	28,572	-123,052
Z — Direction earthquake	74,262	-82,185	-78,184
Dead weight	258,747	-18,987	-55,001
Thermal expansion, 15% power	-1,212,325	-325,568	156,053

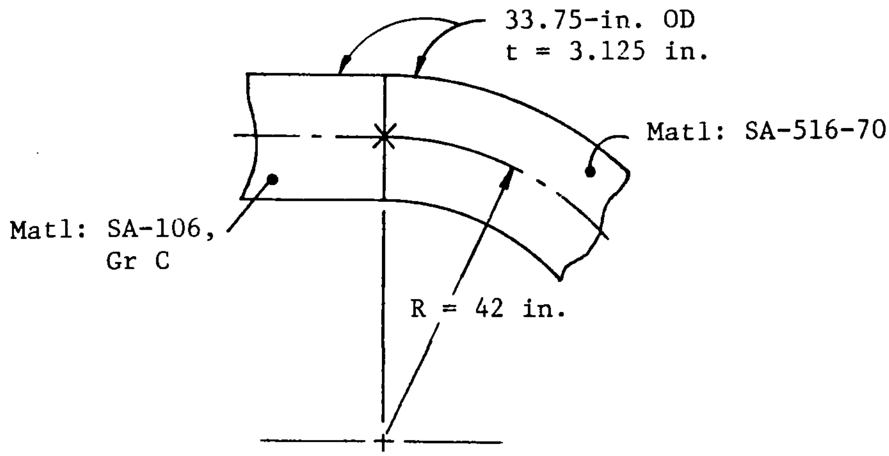
All moments are in ft-lb.

Table 4B-3o

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 131 Branch Number: 11

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-37,826	-1,203	-25,864
Y — Direction earthquake	-35,729	-5,788	-16,537
Z — Direction earthquake	-18,282	-15,958	-50,944
Dead weight	21,265	34,657	-4,584
Thermal expansion, 15% power	12,152	-83,150	-83,094

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-93,205	458,501	308,131
Y — Direction earthquake	-77,245	397,956	245,462
Z — Direction earthquake	-65,433	319,822	149,455
Dead weight	131,616	-228,282	3,603
Thermal expansion, 15% power	-53,567	220,999	-236,547

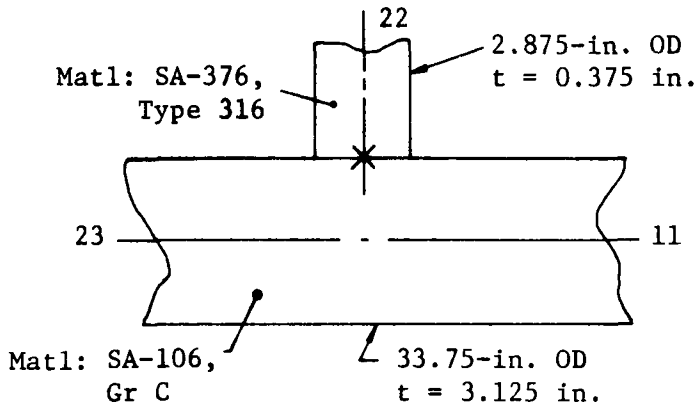
All moments are in ft-lb.

Table 4B-3p

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 920 Branch Number: 22

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	2	13	-3
Y — Direction earthquake	1	11	-2
Z — Direction earthquake	1	4	-1
Dead weight	0	-15	3
Thermal expansion	-13	130	-25

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	-19	-45	-177
Y — Direction earthquake	-18	-40	-154
Z — Direction earthquake	2	-15	-58
Dead weight	13	35	164
Thermal expansion	-220	-247	-1,098

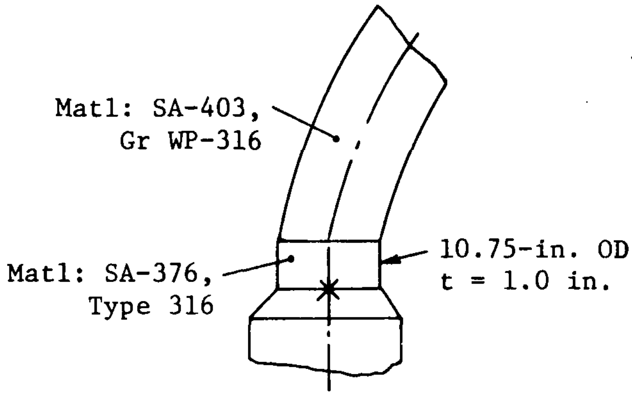
All moments are in ft-lb.

Table 4B-3q

Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 201 Branch Number: 21

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	-218	-89	-11
Y — Direction earthquake	-60	-25	-3
Z — Direction earthquake	-10	-3	20
Dead weight	82	75	16
Thermal expansion, 100% power	8,850	691	2,239

All forces are in lb.

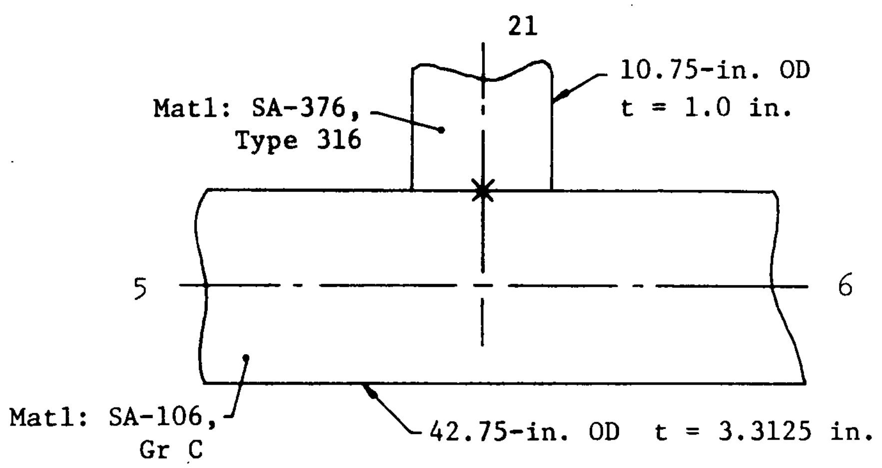
	M_x	M_y	M_z
X — Direction earthquake	573	-1,166	-376
Y — Direction earthquake	34	-363	-218
Z — Direction earthquake	-467	-313	101
Dead weight	-416	372	975
Thermal expansion, 100% power	-8,258	31,558	9,875

All moments are in ft-lb.

Table 4B-3r
Forces and Moments for Branch Points (Stresses Calculated per Appendix F, USAS B31.7)

Branch Point Number: 905 Branch Number: 21

Geometry:



Loads given below are applied to the straight pipe using straight pipe dimensions.

	F_x	F_y	F_z
X — Direction earthquake	218	89	11
Y — Direction earthquake	60	25	3
Z — Direction earthquake	10	3	-20
Dead weight	-82	-75	-16
Thermal expansion, 100% power	-8,850	-691	-2,239

All forces are in lb.

	M_x	M_y	M_z
X — Direction earthquake	44	-412	870
Y — Direction earthquake	138	-71	348
Z — Direction earthquake	705	-190	-51
Dead weight	-32	-34	-279
Thermal expansion, 100% power	24,346	-754	-80,882

All moments are in ft-lb.

Table 4B-4a

Resultant Moment Calculation for Branch Points (Stresses Calculated per Section 1-705, USAS B31.7)

Branch Point Number: 915 Branch Number: 14

Load	M_x	M_y	M_z
(1) X — Direction earthquake	118,455	-263,556	-116,163
(2) Z — Direction earthquake	56,177	-91,393	20,584
(3) Horizontal (max abs value)	118,455	263,556	116,163
(4) Y — (Vertical) earthquake	44,441	-179,545	18,061
(5) Total (3 + 4) single amplitude	162,896	443,101	134,224
(6) 2 x (5) Double amplitude	325,792	886,202	268,448
(7) Dead weight	-353,547	-124,583	28,780
(8) Thermal expansion, 15% power	1,495,602	-181,185	181,177
(9) Thermal expansion, 100% power	1,320,823	-162,348	158,231
(10) Applied thermal (larger of 8 or 9)	1,495,602	181,185	181,177
$M_i = [(\Sigma\|M_x\|)^2 + (\Sigma\|M_y\|)^2 + (\Sigma\|M_z\|)^2]^{1/2}$			
(11) M (5 + 7)	516,443	567,684	163,004
M_i = 784,569 ft-lb	Formula 9		
(12) M (6 + 7 + 10)	2,174,941	1,191,970	478,405
M_i = 2,525,873 ft-lb	Formula 10		
(13) M (6 + 10)	1,821,394	1,067,387	449,625
$M_i = \frac{(6 + 10)\ 2{,}158{,}461}{(10)\ 1{,}517{,}392}$	Formula 11		

Table 4B-4b
Resultant Moment Calculation for Branch Points (Stresses Calculated per Section 1-705, USAS B31.7)

Branch Point Number: 132 Branch Number: 17

Load	M_x	M_y	M_z
(1) X — Direction earthquake	-186,910	-78,623	242,026
(2) Z — Direction earthquake	-95,163	-46,030	107,326
(3) Horizontal (max abs value)	186,910	78,623	242,026
(4) Y — (Vertical) earthquake	-19,382	-49,264	155,802
(5) Total (3 + 4) single amplitude	206,292	127,887	397,828
(6) 2 x (5) Double amplitude	412,584	255,774	795,656
(7) Dead weight	-221,895	-11,603	-46,632
(8) Thermal expansion, 15% power	1,354,884	-176,774	177,547
(9) Thermal expansion, 100% power	1,215,714	-159,235	155,641
(10) Applied thermal (larger of 8 or 9)	1,354,884	176,774	177,547
$M_i = [(\Sigma\|M_x\|)^2 + (\Sigma\|M_y\|)^2 + (\Sigma\|M_z\|)^2]^{1/2}$			
(11) M (5 + 7)	428,187	139,490	444,460
M_i = 632,729 ft-lb	Formula 9		
(12) M (6 + 7 + 10)	1,989,363	444,151	1,019,835
M_i = 2,279,232 ft-lb	Formula 10		
(13) M (6 + 10)	1,767,468	432,548	973.203
$M_i = \frac{(6 + 10)\ 2,063,532}{(10)\ 1,377,854}$	Formula 11		

Table 4B-4c
Resultant Moment Calculation for Branch Points (Stresses Calculated per Section 1-705, USAS B31.7)

Branch Point Number: 232 Branch Number: 18

Load	M_x	M_y	M_z
(1) X — Direction earthquake	55,743	675,625	469,083
(2) Z — Direction earthquake	41,592	410,597	236,886
(3) Horizontal (max abs value)	55,743	675,625	469,083
(4) Y — (Vertical) earthquake	-12,189	402,503	244,337
(5) Total (3 + 4) single amplitude	67,932	1,078,128	713,420
(6) 2 x (5) Double amplitude	135,864	2,156,256	1,426,840
(7) Dead weight	278,187	293,935	266,723
(8) Thermal expansion, 15% power	-326,073	295,789	-266,937
(9) Thermal expansion, 100% power	-233,966	263,260	-224,509
(10) Applied thermal (larger of 8 or 9)	326,073	295,789	266,937
$M_i = [(\Sigma\|M_x\|)^2 + (\Sigma\|M_y\|)^2 + (\Sigma\|M_z\|)^2]^{1/2}$			
(11) M (5 + 7)	346,119	1,372,063	980,143
M_i = 1,721,347 ft-lb	Formula 9		
(12) M (6 + 7 + 10)	740,124	2,745,980	1,960,500
M_i = 3,454,237 ft-lb	Formula 10		
(13) M (6 + 10)	461,937	2,452,045	1,693,777
$M_i = \frac{(6 + 10)\ 3{,}015{,}757}{(10)\ 514{,}850}$	Formula 11		

Table 4B-4d
Resultant Moment Calculation for Branch Points (Stresses Calculated per Section 1-705, USAS B31.7)

Branch Point Number: 908 Branch Number: 9

Load	M_x	M_y	M_z
(1) X — Direction earthquake	-149,590	259,510	135,867
(2) Z — Direction earthquake	-154,615	143,079	22,305
(3) Horizontal (max abs value)	154,615	259,510	135,867
(4) Y — (Vertical) earthquake	-95,274	232,490	38,688
(5) Total (3 + 4) single amplitude	249,889	492,000	174,555
(6) 2 x (5) Double amplitude	499,778	984,000	349,110
(7) Dead weight	192,469	-134,692	-45,650
(8) Thermal expansion, 15% power	-1,404,569	-134,948	-64,455
(9) Thermal expansion, 100% power	-1,239,628	-120,134	-57,847
(10) Applied thermal (larger of 8 or 9)	1,404,569	134,948	64,455
$M_i = [(\Sigma\lvert M_x\rvert)^2 + (\Sigma\lvert M_y\rvert)^2 + (\Sigma\lvert M_z\rvert)^2]^{1/2}$			
(11) M (5 + 7)	442,358	626,692	220,205
M_i = 798,069 ft-lb	Formula 9		
(12) M (6 + 7 + 10)	2,096,816	1,253,640	459,215
M_i = 2,485,785 ft-lb	Formula 10		
(13) M (6 + 10)	1,904,347	1,118,948	413,565
$M_i = \frac{(6 + 10)\ 2,247,136}{(10)\ 1,412,508}$	Formula 11		

Table 4B-4e
Resultant Moment Calculation for Branch Points (Stresses Calculated per Section 1-705, USAS B31.7)

Branch Point Number: 119 Branch Number: 10

Load	M_x	M_y	M_z
(1) X — Direction earthquake	-67,079	254,931	227,806
(2) Z — Direction earthquake	-25,772	138,354	113,263
(3) Horizontal (max abs value)	67,079	254,931	227,806
(4) Y — (Vertical) earthquake	-56,354	218,514	267,088
(5) Total (3 + 4) single amplitude	123,433	473,445	494,894
(6) 2 x (5) Double amplitude	246,866	946,890	989,788
(7) Dead weight	22,030	-148,090	-162,203
(8) Thermal expansion, 15% power	932,119	117,876	-8,928
(9) Thermal expansion, 100% power	864,498	107,403	-5,222
(10) Applied thermal (larger of 8 or 9)	932,179	117,876	8,928
$M_i = [(\Sigma\|M_x\|)^2 + (\Sigma\|M_y\|)^2 + (\Sigma\|M_z\|)^2]^{1/2}$			
(11) M (5 + 7)	145,463	621,535	657,097
M_i = 916,101 ft-lb	Formula 9		
(12) M (6 + 7 + 10)	1,201,075	1,212,856	1,160,919
M_i = 2,064,300 ft-lb	Formula 10		
(13) M (6 + 10)	1,179,045	1,064,766	998,716
$M_i = \frac{(6 + 10)\ 1,876,515}{(10)\ 939,645}$	Formula 11		

Table 4B-4f

Resultant Moment Calculation for Branch Points (Stresses Calculated per Section 1-705, USAS B31.7)

Branch Point Number: 215 Branch Number: 22

Load	M_x	M_y	M_z
(1) X — Direction earthquake	-71	-7	-21
(2) Z — Direction earthquake	-24	-4	-5
(3) Horizontal (max abs value)	71	7	21
(4) Y — (Vertical) earthquake	-61	-7	-18
(5) Total (3 + 4) single amplitude	132	14	39
(6) 2 x (5) Double amplitude	264	28	78
(7) Dead weight	90	13	39
(8) Thermal expansion, 15% power	-783	-190	-516
(9) Thermal expansion, 100% power	-756	-184	-499
(10) Applied thermal (larger of 8 or 9)	783	190	516
$M_i = [(\Sigma\|M_x\|)^2 + (\Sigma\|M_y\|)^2 + (\Sigma\|M_z\|)^2]^{1/2}$			
(11) M (5 + 7)	222	27	78
M_i = 237 ft-lb	Formula 9		
(12) M (6 + 7 + 10)	1,137	231	633
M_i = 1,322 ft-lb	Formula 10		
(13) M (6 + 10)	1,047	218	594
$M_i = \frac{(6 + 10)\ 1,223}{(10)\ 957}$	Formula 11		

Table 4B-4g

Resultant Moment Calculation for Branch Points (Stresses Calculated per Section 1-705, USAS B31.7)

Branch Point Number: 230 Branch Number: 12

Load	M_x	M_y	M_z
(1) X — Direction earthquake	-103,516	431,799	325,990
(2) Z — Direction earthquake	-13,504	65,006	59,513
(3) Horizontal (max abs value)	103,516	431,799	325,990
(4) Y — (Vertical) earthquake	-104,852	416,740	293,278
(5) Total (3 + 4) single amplitude	208,368	848,539	619,268
(6) 2 x (5) Double amplitude	416,736	1,697,078	1,238,536
(7) Dead weight	313,262	-331,439	-318,219
(8) Thermal expansion, 15% power	-347,274	-330,292	272,161
(9) Thermal expansion, 100% power	-253,487	-292,899	234,959
(10) Applied thermal (larger of 8 or 9)	347,274	330,292	272,161
$M_i = [(\Sigma\|M_x\|)^2 + (\Sigma\|M_y\|)^2 + (\Sigma\|M_z\|)^2]^{1/2}$			
(11) M (5 + 7)	521,630	1,179,978	937,487
M_i = 1,594,781 ft-lb	Formula 9		
(12) M (6 + 7 + 10)	1,077,272	2,358,809	1,828,916
M_i = 3,173,236 ft-lb	Formula 10		
(13) M (6 + 10)	764,010	2,027,370	1,510,697
$M_i = \frac{(6 + 10)\ 2,641,239}{(10)\ 551,148}$	Formula 11		

Table 4B-5a

Final Pipe Stresses

(For 36-Inch Pipe)

Branch Point Number	Branch Number	Maximum[a] Primary Stress, psi	Allowable Primary Stress, psi	Max. Primary + Secondary Stress, psi	Allowable Primary + Secd Stress, psi	Usage Factor
103	3	13,417	29,100	14,134	59,100	0.0
104	3	18,349	27,600	21,224	56,100	0.004
106	3	18,381	27,600	24,610	56,100	0.003
107	4	14,883	29,100	15,432	59,100	0.015
905	5	14,883	29,100	16,186	59,100	0.133
114	6	14,905	29,100	15,526	59,100	0.0
111	6	18,430	27,600	32,966	56,100	0.011

10.

[a] Design basis earthquake.

Table 4B-5b
Final Pipe Stresses

(For 28-Inch Pipe)

Branch Point Number	Branch Number	Maximum[a] Primary Stress, psi	Allowable Primary Stress, psi	Max. Primary + Secondary Stress, psi	Allowable Primary + Secd Stress, psi	Usage Factor
140	13	19,858	27,600	42,013	56,100	0.016
145	13	17,677	27,600	51,044	56,100	0.015
915[b]	14	12,860	25,050	33,938	51,960	0.005
132[b]	17	10,347	25,050	35,970	51,960	0.0
133	17	17,688	27,600	32,635	56,100	0.001
141	17	18,056	27,600	39,570	56,100	0.001
232[b]	18	18,912	29,100	39,936	59,100	0.001
908[b]	9	12,947	25,050	33,679	51,960	0.004
119[b]	10	24,698	27,600	53,864	56,100	0.017
122	10	20,670	27,600	54,381	56,100	0.015
920	11	12,433	25,050	14,288	51,960	0.03
123	11	17,612	27,600	31,650	56,100	0.001
131	11	18,352	27,600	45,785	56,100	0.001
230[b]	12	18,095	29,100	38,120	59,100	0.001
920	22	17,860	25,050	15,740	51,960	0.003

[a] Design basis earthquake.
[b] Analyzed in accordance with Paragraph 1-705.1 of B31.7.

Table 4B-5c
Final Pipe Stresses

(Pressurizer Surge Line Piping)

Joint Number	Maximum Primary Stress, psi (a)	Allowable Primary Stress, psi	Max. Primary + Secondary Stress, psi	Allowable Primary + Secd Stress, psi	Usage Factor
1	11,130	24,100	35,380	48,200	.001
2	10,950		35,090		
3	10,190		31,620		
4	8,635		25,910		
5	8,833		27,340		
6	8,746		24,620		
7	11,250		25,330		
8	8,902		24,420		
9	7,955		26,290		
10	8,084		24,470		
11	10,740		24,210		
12	11,690		28,170		
13	11,750		29,910		.001
2 (C01)	24,432		*		.026
3 (C01)	21,420		*		.005
4 (C02)	17,250		44,130		.001
5 (C02)	16,640		47,930		.001
9 (C03)	15,890		45,150		.001
10 (C03)	16,150		40,320		.001
11 (C04)	21,430		39,360		.001
12 (C04)	23,310	24,100	*	48,200	.001

(a) Design Basis Earthquake
*See discussion in Section 4B.3.2.4

10.

Table 4B-6

Foundation Loads for Major Components

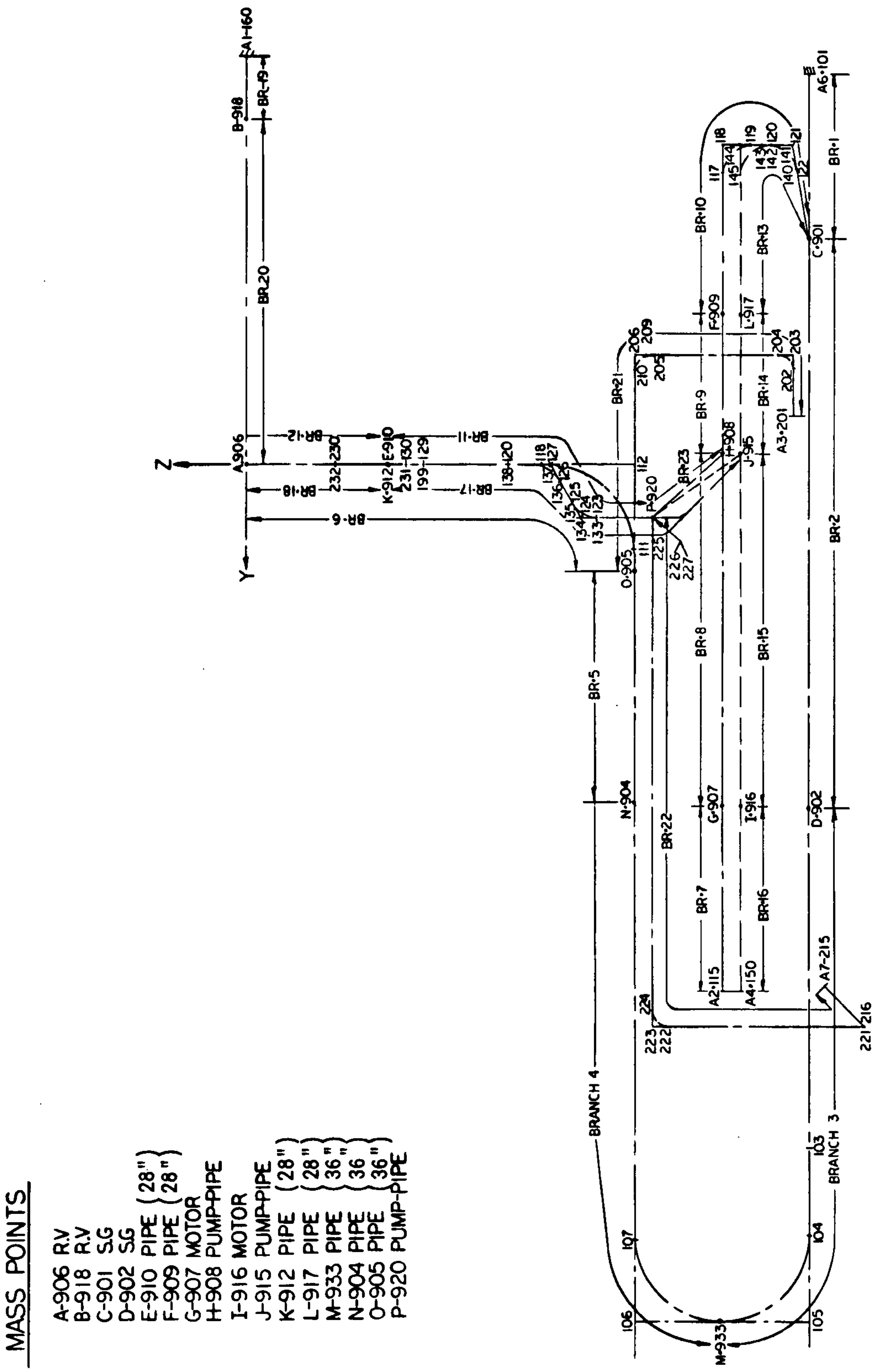
Component	Load Description	Horizontal Force, kips	Vertical Force, kips	Overturning Moment, ft-kips	Twisting Moment ft-kips
Reactor Vessel	Reactor Coolant Piping - Thermal Expansion	40	-280	1,200	135
	Dead Load	--	2,120	--	--
	Seismic Horizontal + Vertical	275	180	8,300	2,960
Steam Generator	Reactor Coolant Piping - Thermal Expansion	365	135	--	20
	Steam and Feedwater Piping Thermal Expansion Load	20	20	--	--
	Dead Load	--	1,845	--	--
	Seismic Horizontal + Vertical	175	336	7,920	735
Pressurizer	Surge and Spray Line Thermal Expansion Load	10	-1	115	30
	Dead Load	--	390	--	--
	0.2 g Seismic Load in Any Direction	90	470	655	30

Table 4B-7
Comparison of Coupled System Vs Uncoupled System
Inertia Loads at Mass Points

	ZY Seismic				XY Seismic			
	Building Coupled		Building Uncoupled		Building Coupled		Building Uncoupled	
Mass	Kips Force	Inches Defl.	Kips Force	Inches Defl.	Kips Force	Inches Defl.	Kips Force	Inches Defl.
1	428.3	0.0072	414.0	0.0068	356.7	0.0039	350.6	0.0036
2	125.7	0.0080	121.9	0.0075	106.0	0.0045	104.9	0.0042
3	536.2	0.0103	525.3	0.0097	485.8	0.0062	475.2	0.0057
4	749.5	0.0146	732.2	0.0137	689.9	0.0098	673.5	0.0090
5	498.7	0.0174	478.1	0.0163	496.7	0.0124	483.3	0.0115
6	21.3	0.0022	20.9	0.0154	23.0	0.0018	19.2	0.0174
7	144.2	0.0125	170.7	0.1522	129.0	0.0085	158.3	0.1752
8	16.4	0.0194	18.1	0.2505	9.5	0.0146	14.7	0.2788
9	15.3	0.0149	16.7	0.0759	10.7	0.0353	12.8	0.1770
10	14.0	0.0074	15.0	0.1300	16.3	0.0524	12.1	0.1544
11	102.8	0.0046	103.5	0.0069	98.9	0.0040	99.0	0.0052
12	50.9	0.0008	51.5	0.0010	50.6	0.0008	51.1	0.0009

Table 4B-8
Comparison of Coupled System Vs Uncoupled System
Element Forces and Moments

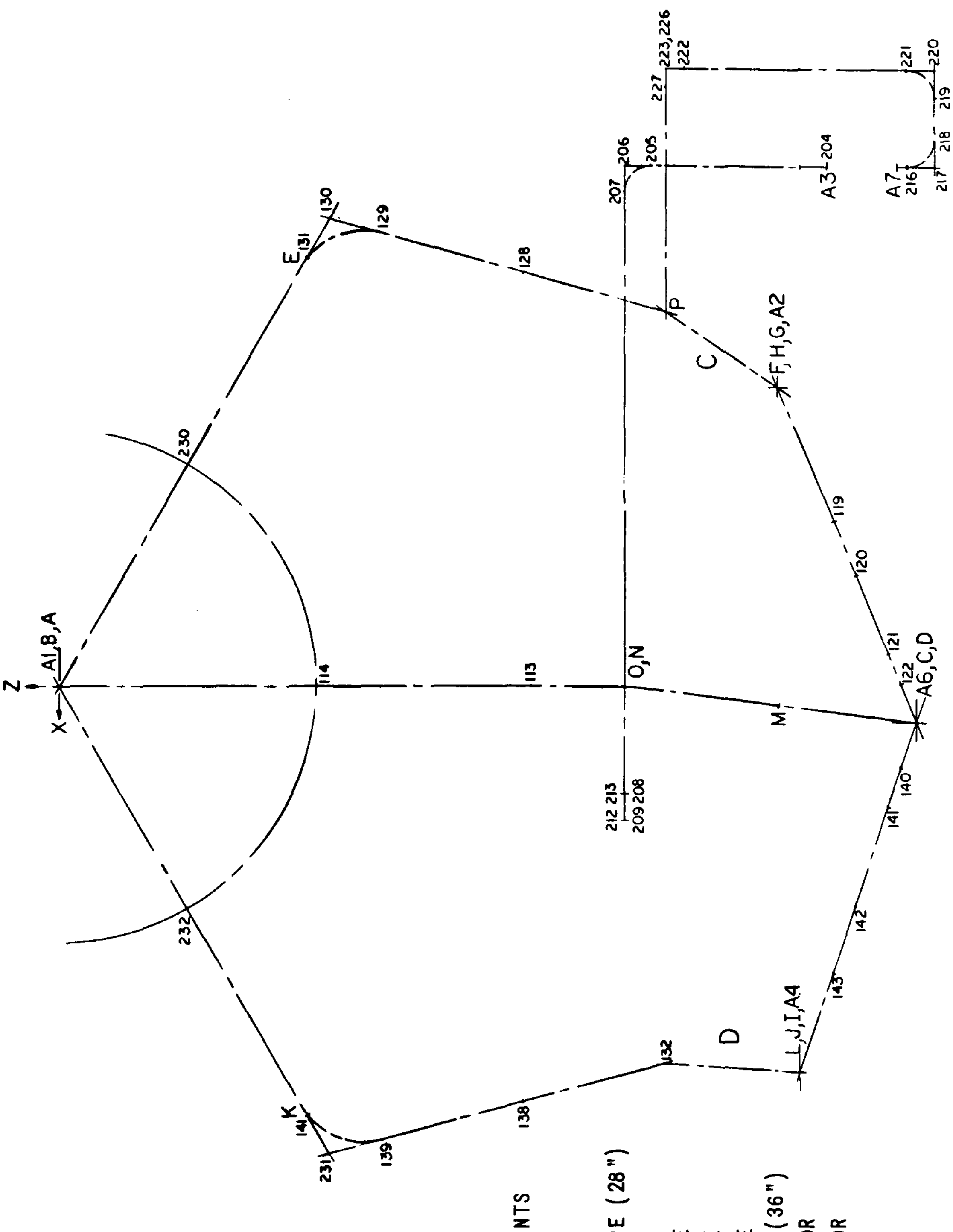
	ZY Seismic				XY Seismic			
	Building Coupled		Building Uncoupled		Building Coupled		Building Uncoupled	
Elements	Force Kips	Mom. In.-Kips	Force Kips	Mom. In.-Kips	Force Kips	Mom. In.-Kips	Force Kips	Mom. In.-Kips
9	18.6		23.9		17.5	1,443.6	12.0	4,855.0
10	6.1	469.4	30.8	2,879.2	13.8	1,678.4	7.2	704.4
11	6.1	1,054.5	30.8	4,769.6	13.8	1,760.2	7.2	2,097.7
12	11.7	1,297.6	37.2	9,475.7	8.4	2,820.0	15.1	4,162.9
13	22.9	1,448.0	49.4	9,928.3	8.7	2,732.1	18.7	4,365.8
14	22.9	--	49.4	--	8.7	1,647.3	18.7	3,536.1
15	22.9	--	49.4	--	8.7	943.3	18.7	2,024.9
7	45.8	6,132.4	163.3	71,886.3	43.2	4,314.3	177.5	82,728.8
8	30.0	9,579.9	146.8	3,354.6	25.9	9,515.5	160.2	3,432.5
16	111.3	29,077.8	136.9	45,341.0	105.2	25,339.6	116.2	33,302.4
17	151.1	29,077.8	177.4	45,341.0	144.5	25,339.6	156.0	33,302.4



SEISMIC, THERMAL AND DEAD LOAD ANALYTICAL MODEL FOR REACTOR COOLANT SYSTEM – ELEVATION

OCONEE NUCLEAR STATION

Figure 4B–1



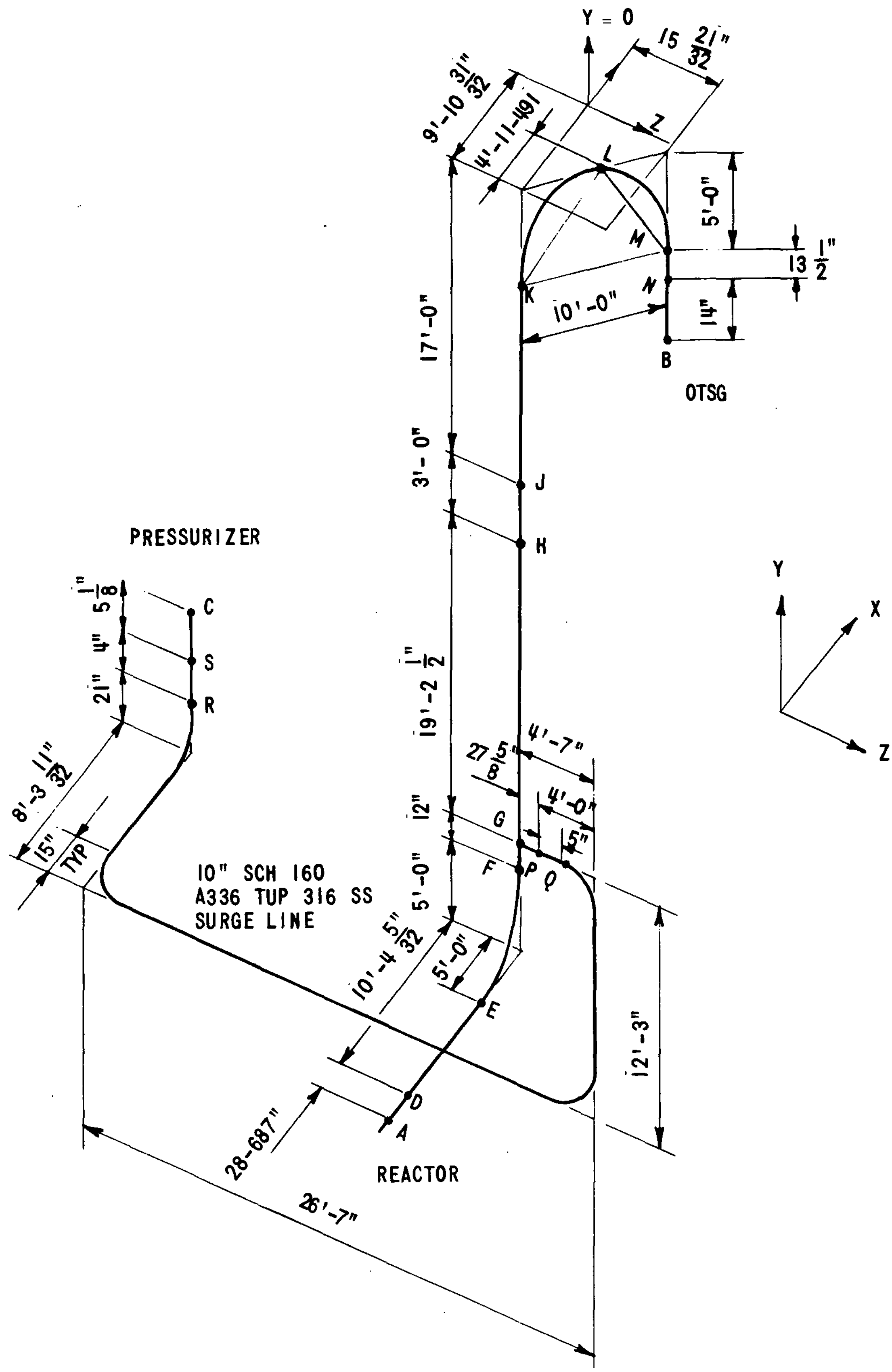
MASS POINTS

A,B - R.V.
E,F,L,K - PIPE (28")
C,D - S.G.
H - PUMP-PIPE
J - PUMP-PIPE
P - PUMP-PIPE
M,N,O - PIPE (36")
I - PUMP MOTOR
G - PUMP MOTOR

SEISMIC, THERMAL AND DEAD LOAD ANALYTICAL MODEL FOR REACTOR COOLANT SYSTEM – PLAN

OCONEE NUCLEAR STATION

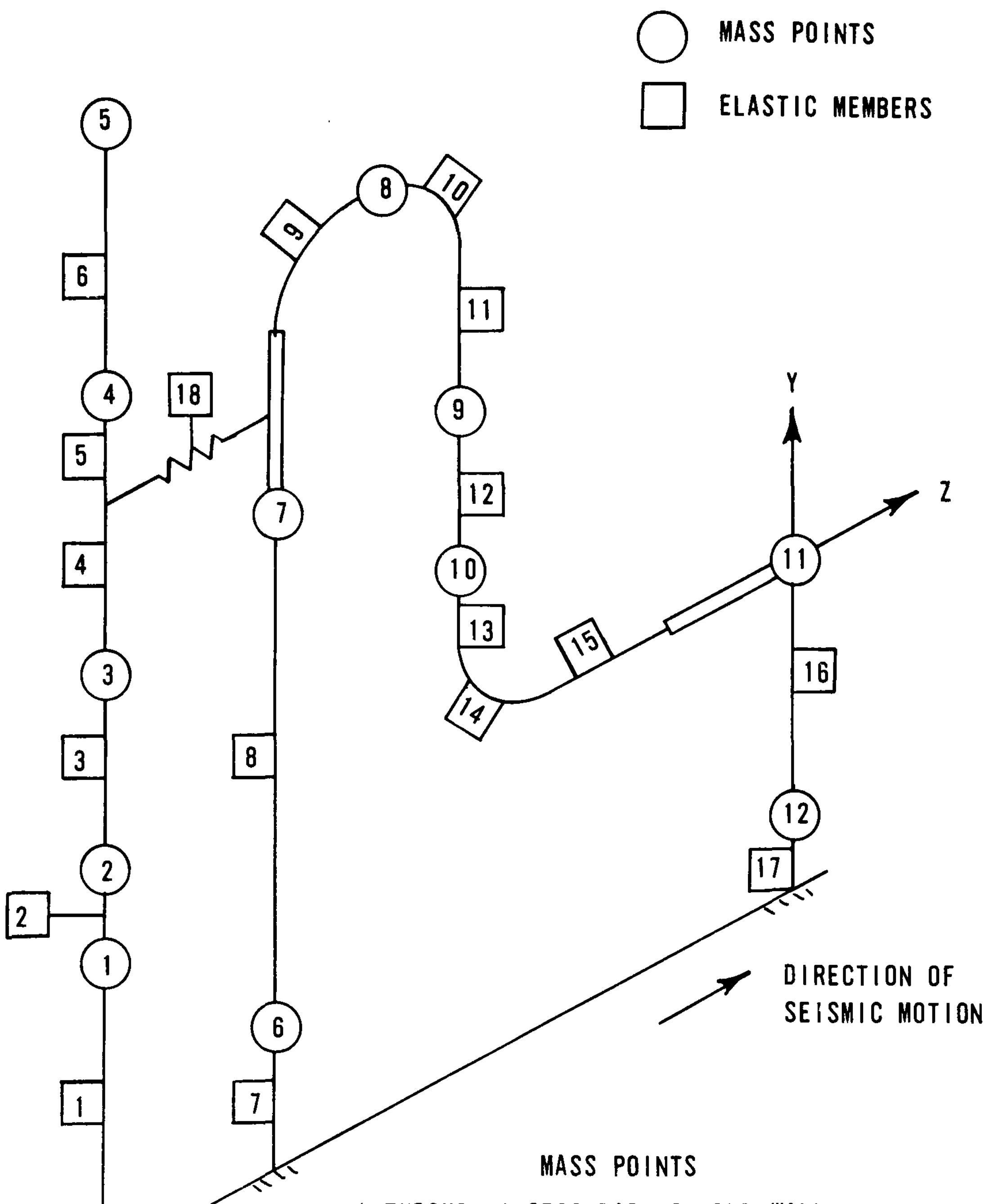
Figure 4B–2



36" REACTOR COOLANT PIPE THERMAL EXPANSION MODEL FOR B & W PROGRAM

OCONEE NUCLEAR STATION

Figure 4B–3



MASS POINTS

1 THROUGH 5 SECONDARY SHIELD WALL

6 AND 7 STEAM GENERATOR

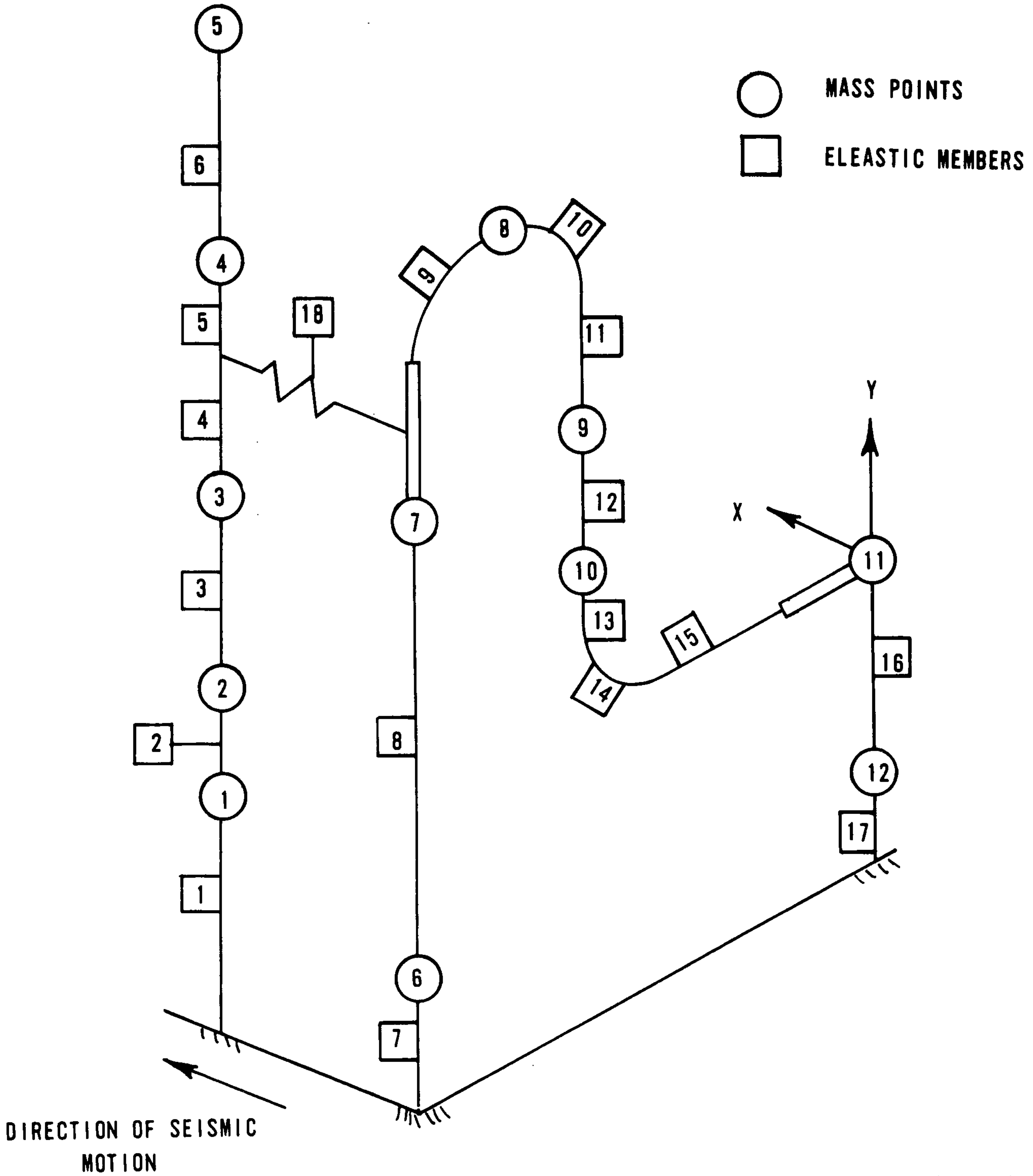
8 THROUGH 10 36" REACTOR COOLANT PIPE

11 AND 12 REACTOR VESSEL

DYNAMIC MODEL OF SECONDARY SHIELD
WALL AND NSS IN THE ZY PLANE

OCONEE NUCLEAR STATION

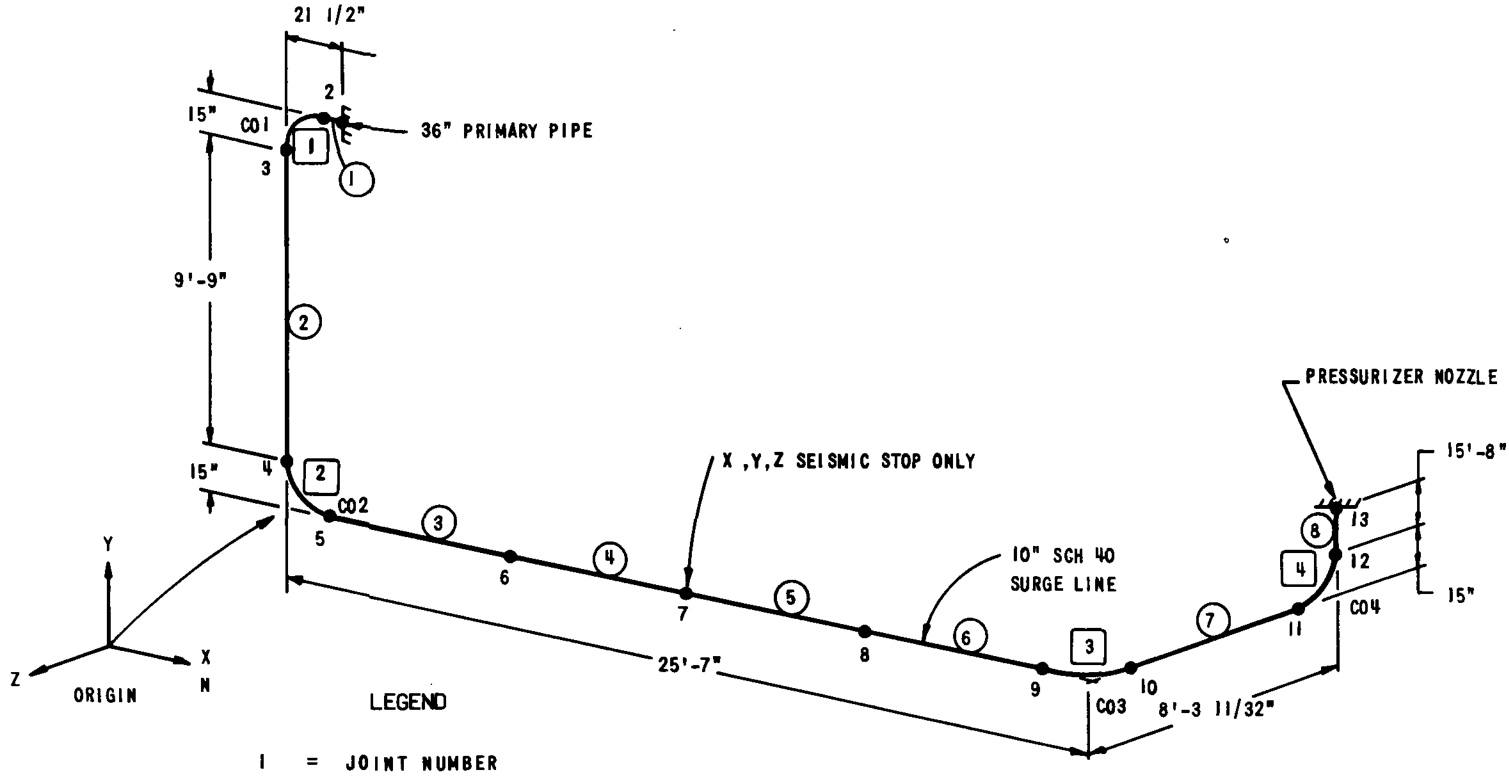
Figure 4B - 4



DYNAMIC MODEL OF SECONDARY SHIELD WALL AND NSS IN THE XY PLANE

OCONEE NUCLEAR STATION

Figure 4B–5



LEGEND

1 = JOINT NUMBER

(1) = STRAIGHT MEMBER NUMBER

[1] = CURVED MEMBER NUMBER

C01 = CURVED COMPONENT DESIGNATION

SEISMIC, THERMAL AND DEAD LOAD ANALYTICAL
MODEL FOR THE PRESSURIZER SURGE LINE PIPING

OCONEE NUCLEAR STATION

Figure 4B-6
(New) Rev. 10 8/28/70

LIST OF EFFECTIVE PAGES

FSAR APPENDIX 4C

SUMMARY OF BAW- 10,008 PART 2

APPENDIX 4C

SUMMARY OF FUEL ASSEMBLY STRESS & DEFLECTION ANALYSIS DUE TO LOCA AND SEISMIC EXCITATION

Submitted With FSAR Revision No. 6

June 22, 1970

APPENDIX 4C

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES
(At Rear of Section)

4C.1 INTRODUCTION

This appendix summarizes an analysis of the fuel assembly for loads caused by the depressurization transient following an instantaneous reactor coolant pipe rupture and/or seismic excitation. Four separate loading conditions were investigated: loads due to (1) the design basis earthquake (DBE) (2) the maximum hypothetical earthquake (MHE), (3) a loss-of-coolant accident (LOCA), and (4) the simultaneous occurrence of a MHE and a LOCA timed so that the combined deflections are maximum. It was found that the LOCA loads are most severe for an outlet pipe rupture. Loads for an inlet pipe rupture are less severe because of the higher flow resistance of the smaller pipe and better equalization of pressures permitted by the internals vent valves.

The maximum loads or deflections occurring in the fuel assembly are discussed and tabulated. The deflections caused by these loads are tabulated for cases and locations where deformations have a potential safety implication. The analysis is based on a conservative application of loads and end conditions, etc., resulting in calculated internal loads (or deflections) that exceed the actual internal loads (or deflections) for the given applied load.

Investigation of the effects of the foregoing loadings identified two areas to be investigated:

1. Horizontal - contact between fuel assemblies due to motions in a horizontal plane, where the contact occurs primarily at the mid-span grid spacers.

2. Vertical - contact of fuel assemblies with the internals due to upward pressure, where the contact occurs between the end fittings and the grid plates.

The seismic accelerations used in these analyses are those specified for the Oconee site. The time-history method was used to evaluate the seismic effects. The calculation of LOCA forces is described in Topical Report BAW-10008, Part 1, Rev. 1.

The discussion and tabulated loads and deflections are submitted as a basis for the conclusion that the fuel assemblies can withstand a LOCA, the combined effects of a maximum hypothetical earthquake and loss-of-coolant accident, a maximum hypothetical earthquake, and a design basis earthquake without exceeding the respective allowable limits.

4C.2 DESCRIPTION

4C.2.1 REACTOR VESSEL

The general arrangement of the reactor pressure vessel is shown on Figure 4C-1. The reactor vessel consists of a cylindrical shell, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. A cylinder welded to the vessel's shell supports the vessel and extends downward to a flanged base ring which is bolted to the building's foundation. The reactor vessel's ring flange includes an internal ledge to support the core and the internal structural components. The vessel has two outlet nozzles through which reactor coolant is transported to the steam generators, and four inlet nozzles through which reactor coolant re-enters the reactor vessel.

4C.2.2 REACTOR INTERNALS

The internal components of the reactor include the plenum assembly and the core support assembly. The reactor internals assembly is shown on Figure 4C-1. The reactor internals are supported by a ledge on the inside of the reactor vessel closure flange and are designed to support the core, maintain fuel assembly alignment, and limit fuel assembly movement.

4C.2.2.1 PLENUM ASSEMBLY

The plenum assembly, located directly above the reactor core, consists of a plenum cover, upper grid, control rod guide tube assemblies, and a flanged plenum cylinder with openings for reactor coolant outlet flow. The upper grid assembly is attached to a flange which is bolted to the lower flange of the plenum cylinder.

4C.2.2.2 CORE SUPPORT ASSEMBLY

The core support assembly consists of the core support shield, core barrel, lower grid, flow distributor, and thermal shield.

4C.2.3 FUEL ASSEMBLY

The canless fuel assembly (Figure 4C-2) consists of six intermediate and two end spacer grids, 16 guide tubes, and two end fittings. These components form the basic structural cage. There is an instrumentation tube in the central position of the array. Seven spacer sleeve segments are positioned along the length of the fuel assembly on the instrumentation tube - one segment between each spacer grid.

The spacer grids are constructed from slotted strips assembled in egg-crate fashion and welded at each intersection. Each grid has 32 strips, 16 perpendicular to 16, to form a 15 by 15 array. Contact points are formed in each strip which extends into each square opening to contact and support fuel rods and control rod guide tubes in two mutually perpendicular planes. The end spacer grids differ from the intermediate spacer grids in that the peripheral strip is extended to form a square box which is screwed to the end fitting. The spacer grids locate fuel rods, maintain coolant channel geometry, and contribute to the lateral stiffness of fuel assemblies.

The lower end fitting is a weldment of two castings. The base casting, to which the grid casting is welded, fits into and rests on the lower grid assembly of the reactor core support assembly. The upper end fitting assembly is similar to the lower end fitting assembly. Penetrations in the upper end fitting grid are provided for the guide tubes. Attached to the end fitting are a coiled compression spring and a cast holddown spider. The spider consists of a ring of the same diameter as the spring with four radial bars that contact the upper grid assembly of the reactor plenum assembly.

The end fittings serve as rigid connectors for the 16 guide tubes, provide lateral and vertical location of fuel assemblies, and act to restrain the fuel rods vertically, thereby positively determining the vertical location of the fuel. In addition, the upper end fitting must be capable of housing the holddown spring to prevent fuel assembly lift off, and it must be capable of absorbing energy during LOCA contact of the fuel assembly with the upper grid. The guide tubes provide guidance for control rods. The spacer sleeves provide positive axial positioning of the spacer grids.

Table 4C-1 is a list of the materials used.

Table 4C-1. List of Fuel Assembly Materials

Component	Material
Guide tubes	Zircaloy-4
Spacer sleeves	Zircaloy-4
Fuel rod cladding	Zircaloy-4
Instrument tube	Zircaloy-4
Spacer grids	Inconel-718
Holddown spring	Inconel X-750
Guide tube nuts	Type 304 stainless steel
End grid assembly screws	Type 304 stainless steel
Holddown spider	Stainless steel, grade CF-3M
End fittings	Stainless steel, grade CF-3M

4C.2.4 FUEL ASSEMBLY STRUCTURAL DESIGN CRITERIA

Loads and permanent deflection for the design basis earthquake (DBE) will be limited as follows:*

1. Loads on the fuel assembly spacer grid shall not exceed the elastic limit of the spacer grid determined from tests performed on production grids.

2. There shall be no permanent deformation of the fuel assembly spacer grids.

---

* Actual numerical limits are shown in Table 4C-2, and the tests conducted to determine these limits are described in Section 4C.5.

Loads and permanent deflection for the maximum hypothetical earthquake (MHE), LOCA, and simultaneous LOCA and MHE will be limited as follows:*

1. Loads on the fuel assembly spacer grid will be allowed to exceed the elastic limit, but the permanent deformation of the spacer grid shall not exceed that which would distort the guide tubes and prevent the insertion of the control rods. This value of permanent deformation is to be determined by tests on production grids.

2. To provide stability, loads on the control rod guide tubes and end spacer grid assembly will be limited to 85% of the critical Euler buckling load. The value of 85% is chosen as a value, based on engineering judgment, so as not to design to failure.

3. Loads on the spacer grid welds shall be limited to 85% of the load that would cause failure. The value of this load is to be determined by tests on production grids.

4. Loads on the bolts connecting the end spacer grid skirt to the end fitting shall be limited to 85% of the load that would cause the bolts to fail in shear. The value of this load is to be determined by tests on production end grid assemblies.

The preceding criteria provide sufficient safety margin against failure. All margins in this report are calculated as follows:

$$\text{Margin} = \frac{[\text{Allowable (load) - Applied (load)}] \times 100\%}{\text{Applied Load}}.$$

Since the allowable loads are based on the foregoing criteria, the margins quoted in this report are in excess of those required by the criteria. Thus, any positive margin, including zero, is acceptable. A zero margin, for example, indicates that the criterion has just been met.

4C.3 LOADS

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*Actual numerical limits are shown in Table 4C-2, and the tests conducted to determine these limits are described in section 4C.5.

4C.3.1 VERTICAL LOADS ON CORE DURING LOCA

The total force acting on a single fuel assembly for the outlet rupture is given in Figure 4C-3. Figure 4C-3 is a combination of Figure 6 (ΔP across core for a 36-inch-ID outlet break) and Figure 10 (shear force on core for a 36-inch-ID outlet break) from Topical Report BAW-10008, Part 1, Rev. 1. It is found in the following manner:

$$\text{Figure 4C-3} = \frac{\text{(Fig. 6)(blocked area of core)} + \text{(Fig. 10)} - \text{(weight of core)}}{\text{177 fuel assemblies}}.$$

This combined pressure and fluid friction force is sufficient to cause the fuel assemblies to lift off of the lower grid and contact the upper grid. They deflect the upper grid, causing axial loads in the control rod guide assemblies and subsequent deflection of the plenum cover beams. The resisting force from the plenum cover stops the fuel assemblies and causes them to return to the lower grid.

4C.3.2 HORIZONTAL THRUST FORCE DURING LOCA

The LOCA thrust force acting at the vessel's outlet nozzle was analyzed using the FLASH computer code and the relationship

Thrust = pressure x area.

Testing associated with the LOFT program tends to confirm that the horizontal thrust can be calculated by this relationship. The FLASH program has been used to correlate the vessel pressure and, therefore, the thrust for some of the semi-scale blowdown tests.

The results for a 36-inch outlet pipe rupture are shown in Figure 4C-4.

4C.3.3 SEISMIC EXCITATION

The specific seismic time history used in this analysis was determined for the Oconee site. The record used was the El Centro 1940 NS Earthquake normalized to the Oconee site level.

4C.4 MODELS USED IN ANALYSIS

4C.4.1 HORIZONTAL CONTACT ANALYSIS

Structurally, the fuel assemblies are long slender beams which are responsive to horizontal excitations. Because of the proximity of the assemblies, these motions could result in midspan contact. The concern is that such contacts could produce unacceptable damage to the spacer grids and thus reduce coolant flow or restrict control rod motion. Two possible forms of horizontal excitation are seismic and LOCA. Seismic excitation occurs at the vessel's foundation, and the LOCA produces a thrust force (as described in section 4C.3.2) at the nozzle. The vertical component of the earthquake was considered with the horizontal analysis. However, because of the vertical stiffness of the reactor internals, the seismic contribution to the displacement of the core is negligible (about 0.002 to 0.003 inch) with respect to the horizontal motion.

4C.4.1.1 OVERALL MODEL

Both of these excitations can produce horizontal motion of the fuel assemblies, so that a dynamic model including all the components involved - the reactor vessel, the control rod drives, the internals, and the fuel assemblies - was needed.

4C.4.1.2 FIRST SEGMENT

This overall model was divided into two segments. The first segment included all the components named above except individual fuel assemblies, and involved recording the motions of the upper and lower grid plates, the core support shield, and earth velocity versus time. These motions were input excitations for the second segment of the overall model.

The first step in the solution was to determine a model that accurately represented the structure being investigated. The more masses used, the more accurate (but also the more complex) the solution. The investigation of different models showed that for these horizontal contacts a nine-mass model (Figure 4C-5) was sufficient to describe the motions of the components.

The method used in this dynamic model was the far coupled, "lumped mass" approach, which may be considered as the vibration equivalent of the finite element technique in static stress analysis problems. The distributed mass of components of the structure is considered to be concentrated at discrete points. These mass points are connected by massless flexible elements. The behavior of the total structure is then determined from the response of these mass points. No damping is included.

Once the model was fixed, a set of simultaneous equations was written for the structure. For a system of N masses, the equations in matrix form are

$$\underset{N\times N}{[M]}\ \underset{N\times 1}{\{\ddot{X}\}} + \underset{N\times N}{[K]}\ \underset{N\times 1}{\{X\}} = \underset{N\times 1}{\{F\}}$$

where $[M]$ = mass matrix,

$\{\ddot{X}\}$ = acceleration matrix,

$[K]$ = stiffness matrix,

$\{X\}$ = displacement matrix,

$\{F\}$ = force matrix.

This model was then programmed for a digital computer. A modification was made to invert the final flexibility matrix generated and thus obtain a stiffness matrix. Each row of the stiffness matrix was then divided by the mass corresponding to that row to obtain a "K/M" matrix. These values were then substituted into scaled equations and solved on the analog computer.

To validate the analog representation of the model, initial displacement tests were performed. The digital program generated frequencies and mode shapes. The nine-mass model on the analog was displaced into a particular mode shape and then allowed to vibrate. The frequency and mode shape of vibration from the analog compared well with the digital results.

4C.4.1.3 EXCITATION OF FIRST SEGMENT

As shown in Figure 4C-5, the seismic excitation was applied at the base of the reactor vessel and the results recorded. These time-history records have been compared with the published spectrum.

The simultaneous occurrence of the MHE and LOCA was also recorded. The seismic excitation was applied at the vessel's skirt, as described above, and the LOCA thrust force was applied at the nozzle. Owing to the relative timing of these two events, maximum fuel assembly displacement was obtained. The investigation indicated that maximum displacement gave maximum contacts and hence maximum loads.

4C.4.1.4 SECOND SEGMENT

In the second part of the program the model comprised five fuel assemblies, two core baffles, and associated circuitry. Each fuel assembly was modeled by far-coupling techniques with three lumped masses. Each mass had an individual damper and an elastic-plastic spring that represented the transverse structural properties of the grid. The core baffle was represented by an elastic spring. The clearances between assemblies and between the core baffle and outside assemblies were also established during the program. The elastic-plastic properties of the spacer grid were determined by test and used as program input. The frequency and damping properties of the fuel assembly were established by test, as described in Section 4C.5, and were used as program input.

Because of the input excitation, contacts occurred between adjacent assemblies or between assemblies and the core baffle. The elastic-plastic grid spring allowed a maximum value of force to be exerted, and any remaining motion of the assembly created permanent deformation of the grid. The program considered the energy loss involved as well as the change in cross-sectional size of the impacting grid. These effects influenced the event as it occurred and also influenced the succeeding contacts. The program summed the total grid deformation from all contacts during the seismic history and presented this value as program output. These results were compared with the general design criteria as well as the specific criterion.

4C.4.2 VERTICAL CONTACT ANALYSIS

This analysis was conducted to determine the loads acting on the various parts of the fuel assembly as a result of vertical contact with the upper grid assembly during a LOCA. The fuel assembly, when subjected to the upward pressure force caused by the instantaneous rupture of a primary outlet pipe, will suddenly accelerate vertically toward the upper grid assembly. When the fuel assembly does contact the upper grid assembly, the fuel rods will tend to slip in the upper end spacer grid. Since the stiffness of the end spacer grid assembly is substantially greater than the axial stiffness of the guide tubes, once slippage occurs, a compressive load is applied to the guide tubes by way of the lower end grid-end fitting assembly. Since any dynamic buckling of the guide tubes during a LOCA could prevent control rod insertion, investigation of the loads applied to the guide tubes was the primary concern in this analysis.

The following conservative assumptions were made:

1. Based on the flexibility of the other members, the fuel rods are considered to be rigid.

2. There is no slip of fuel pellets relative to the fuel rod cladding.

The mathematical model of the fuel assembly is shown in Figure 4C-6. By appropriately combining springs in series and parallel, this model was simplified to a single-degree-of-freedom system, i.e., a single mass and spring combination. However, the single spring reflected the nonlinear characteristics of the overall structural system. A typical load-deflection curve for this spring, based on data obtained from production fuel assemblies, is shown in Figure 4C-7. It should be noted that this spring curve is for the beginning of life (BOL). Its shape will change considerably as a function of full-power operational time because of irradiation growth, other irradiation effects, and material yield strength variation. These effects were considered in the analysis. The forcing function acting on the model is based on the LOCA pressure curves given in Topical Report BAW-10008, Part 1, Rev. 1 and is shown in Figure 4C-3. Section 4C-3.1 gives details of its determination.

To account for the nonlinear spring characteristics of the fuel assembly model and also for the rapidly changing forcing function, a digital program was developed. Utilizing a numerical integration routine based on the linear acceleration assumption, the program calculated the three dynamic parameters of interest-displacement, velocity, and acceleration- at every quarter-millisecond. A logic system monitored the displacement and adjusted the spring rate to reflect the nonlinearities of the spring. Calculation was stopped when a negative velocity, indicating that the fuel assembly was moving in a downward direction, was encountered.

From the digital output, the maximum displacement of the fuel assembly was used to enter the spring load-deflection curve and read directly the total fuel assembly load. This load was used to determine the maximum guide tube load.

4C.5 TESTS CONDUCTED

4C.5.1 FREQUENCY AND DAMPING TESTS

The fuel assembly frequency and damping values were established from several test programs in which full-sized test specimens were used. Tests were performed in air, in still water at temperatures up to 200 F, and in still and flowing water at reactor operating conditions (650 F and 2200 psi). Both displacement loading (pluck tests) and steady-state sinusoidal excitation were used.

This extensive testing confirmed that the natural frequency of the assembly is in the low frequency range, and provided the damping values for use in the analysis. Both frequency and damping are also dependent on the amplitude of vibration; this dependence is due to fuel rod slippage in the spacer grids, and the slippage is the prime source of the damping values. The tests also established that damping increases with the coolant flow velocity owing to the effect of coolant flow on the spacer grids.

4C.5.2 SPACER GRID COMPRESSION TESTS

The analysis of fuel assemblies during conditions of horizontal acceleration and contact required knowledge of certain transverse characteristics of the spacer grids. These characteristics are (1) the elastic and plastic load abilities, and (2) the amount of permanent deformation and energy that can be absorbed without interfering with control rod motion. This information was obtained by performing compression or crush tests on individual spacer grids.

Each grid was filled with simulated fuel rods and guide tubes and then mounted in a vertical plane in the tensile machine (Figure 4C-10). A vertical compressive load was applied while recording the grid deflection and other significant data. During the loading, efforts were continually made to insert poison pin segments into the guide tubes. The load and the grid distortion at which this was no longer possible were recorded. These test results are corrected for temperature effects by applying a ratio of the grid material's (Inconel-718) yield strength at temperature (600 F) to its yield strength at room temperature. The results of this procedure give the initial elastic load ability for the grids, the load cycling as the horizontal rows of the grid fail, and the permanent distortion of the grid at the time when the poison pins could no longer be inserted.

These results were obtained from essentially static tests. The first two results were used as input data for the horizontal contact analysis, and the third was used as the acceptance criterion for grid deformation as though it had been obtained dynamically. The results have been checked qualitatively against the effect of dynamic loading with a drop test as described below.

4C.5.3 SPACER GRID DROP TEST

The results of the compression tests depended to some extent on the mode of failure, which was transverse displacement of individual fuel rod rows. To check this failure mode qualitatively under dynamic loading conditions, a single grid loaded with short lengths of fuel rod was dropped on a solid base so as to land on its side. Only those rows nearest the impact surface were crushed, and this result supported the assumption that dynamic loading would decrease with distance from the impacting surface. Therefore, since the analysis of the static crush test assumed that the maximum impact load was applied uniformly over the three rows of spacer grid nearest the impact surface, the test indicated that the analysis was conservative.

4C.6 RESULTS

4C.6.1 HORIZONTAL CONTACT ANALYSIS

4C.6.1.1 DESIGN CRITERION

The level of permanent distortion suffered during the maximum hypothetical earthquake, LOCA, or MHE plus LOCA must not prevent control rod insertion.

4C.6.1.2 RESULTS AND MARGINS OF SAFETY

The results of the analysis described in Section 4C.4.1 show that the canless fuel assembly meets the general design criteria as well as the specific design criterion stated above; the margins of safety for the three excitation levels are as follows:

Excitation	Margin, %
DBE	80
MHE	300
MHE + LOCA	400

These margins were calculated as follows:

Oconee level = 0.10G

Level to obtain 150 mils = 0.40G

$$\text{Margin} = \frac{0.40 - 0.10}{0.10} \times 100 = 300\%.$$

For the MHE case, for example, the allowable deformation equals 150 mils.

4C.6.1.3 CONCLUSION

The reference fuel assembly design can withstand the horizontal contact loads.

4C.6.2 VERTICAL CONTACT ANALYSIS

4C.6.2.1 GUIDE TUBE BUCKLING

4C.6.2.1.1 Specific Design Criteria

The compressive load in guide tubes should not exceed 85% of the static Euler buckling load. The holddown spider will be allowed to yield since it serves no safety function.

4C.6.2.1.2 Results and Margins of Safety

The results of this analysis are presented in Figure 4C-8, which shows that

the maximum loads experienced by the guide tubes during a LOCA are less than the allowable guide tube loads defined in the design criteria. The
16. margin of safety is 4%.

4C.6.2.1.3 Conclusion

The fuel assembly design can withstand a vertical LOCA contact.

4C.6.2.2 UPPER END SPACER GRID WELDS

As shown in Figure 4C-9, the spacer grid is formed of strips assembled in egg-crate fashion. The top and the bottom of each such intersection are tungsten-inert-gas welded. During a loss-of-coolant accident, the fuel assemblies contact the upper internals grid. The resultant fuel rod deceleration loads are transmitted through the end spacer grid. The load carried by the grid is limited to the slip load of the fuel rods in the grid.

The bending moment at the middle of a grid strip was calculated on the basis that the grid strip was a simply supported beam subjected to the slip load of the fuel rods. The moment divided by the grid depth is the force carried by the grid welds.

To verify this method of calculating the force on a spacer grid weld, a series of tests was conducted using portions of spacer grids loaded as shown in Figure 4C-9. The grid strips were loaded to failure, and the corresponding force on the welds was compared with previous pull-test results for welds loaded only in tension. These tests indicated that the weld strength can be predicted using the analytical methods described.

4C.6.2.2.1 Specific Design Criteria

The spacer grid welds are capable of supporting a tensile load of 225 pounds at room temperature. The comparable load at temperature (600 F) is 200 pounds. The stresses from normal LOCA and earthquake loads are limited to 85% of ultimate stress, reducing the 200-pound load to an allowable load of 170 pounds. The force carried by the end spacer grid welds was calculated using the method described above. For the total maximum possible fuel rod slip load, the maximum weld force during a LOCA is 150 pounds. The allowable load is 170 pounds, and the margin is 13%. The loads due to a LOCA and/or earthquake are not additive to those due to normal operation because the maximum loads are limited by the available friction loads between the end grids and the fuel rods.

4C.6.2.2.2 Conclusion

The end spacer grid welds can withstand the vertical contact loads.

4C.6.2.3 END SPACER GRID ASSEMBLY

This assembly consists of an end spacer grid, a skirt that connects the end spacer grid to the end fitting, and the end fitting. The skirt is formed by extending the 20-mil outside strips of the spacer grid and reinforcing them with a 30-mil doubler. This composite plate is attached to the end fitting with sixteen 3/8-inch-diameter screws countersunk into sixteen 1-inch-wide bosses on the end fitting, as well as butting against a shoulder on the end fitting.

The Euler critical load of this box section was calculated assuming that only the width of material in contact with the 1-inch-wide bosses on the end fitting is effective as column material; i.e., the "column" consists of sixteen 1-inch-wide strips which are assumed to be pin-ended. The remainder of the material was evaluated as cantilever springs providing lateral support at the column midheight.

The skirt and attaching bolts were evaluated for a compressive load equal to the maximum possible slip load of the fuel rods in the end spacer grid due to vertical contact at the beginning of life.

The assembly was also evaluated for transverse loads. An arbitrary and conservative lateral deflection of 1 inch from the horizontal contact analysis was used, and the moment at the bolted joint was determined.

4C.6.2.3.1 <u>Specific Design Criteria</u>

The skirt must not buckle. The allowable load is 85% of the critical buckling load.

4C.6.2.3.2 <u>Conclusion</u>

The end spacer grid assembly is adequate for the maximum anticipated loads as given in Table 4C-2. Some minimum margins are as follows:

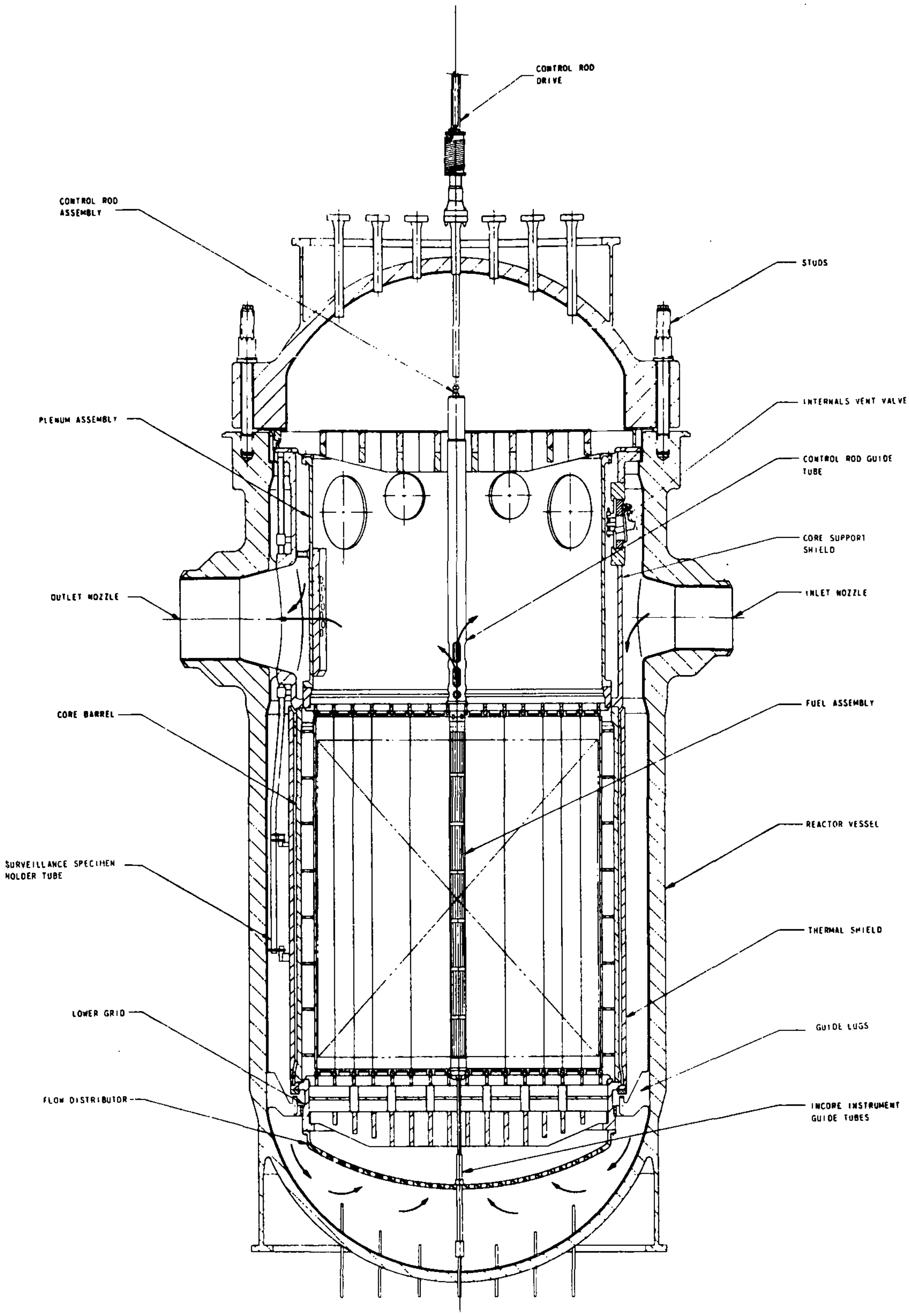
Component	Margin, %
Skirt buckling	16
Bolt shear	16

Table 4C-2. Results

	Calculated Deflection or Load	Allowable Deflection or Load	Margin %
Horizontal contact analysis - spacer grid permanent deformation, in.			
DBE	0.000	0.000	80[a]
MHE	0.010	0.150	400[a]
MHE + LOCA	0.012	0.150	300[a]
Vertical contact analysis, lb			
16. Guide tube buckling	5,400	5,588	4
Upper end spacer grid welds	150	170	13
End spacer grid assembly - buckling	11,000	12,800	16
End spacer grid bolts - shear	690	800	16

[a] Calculated from the relation

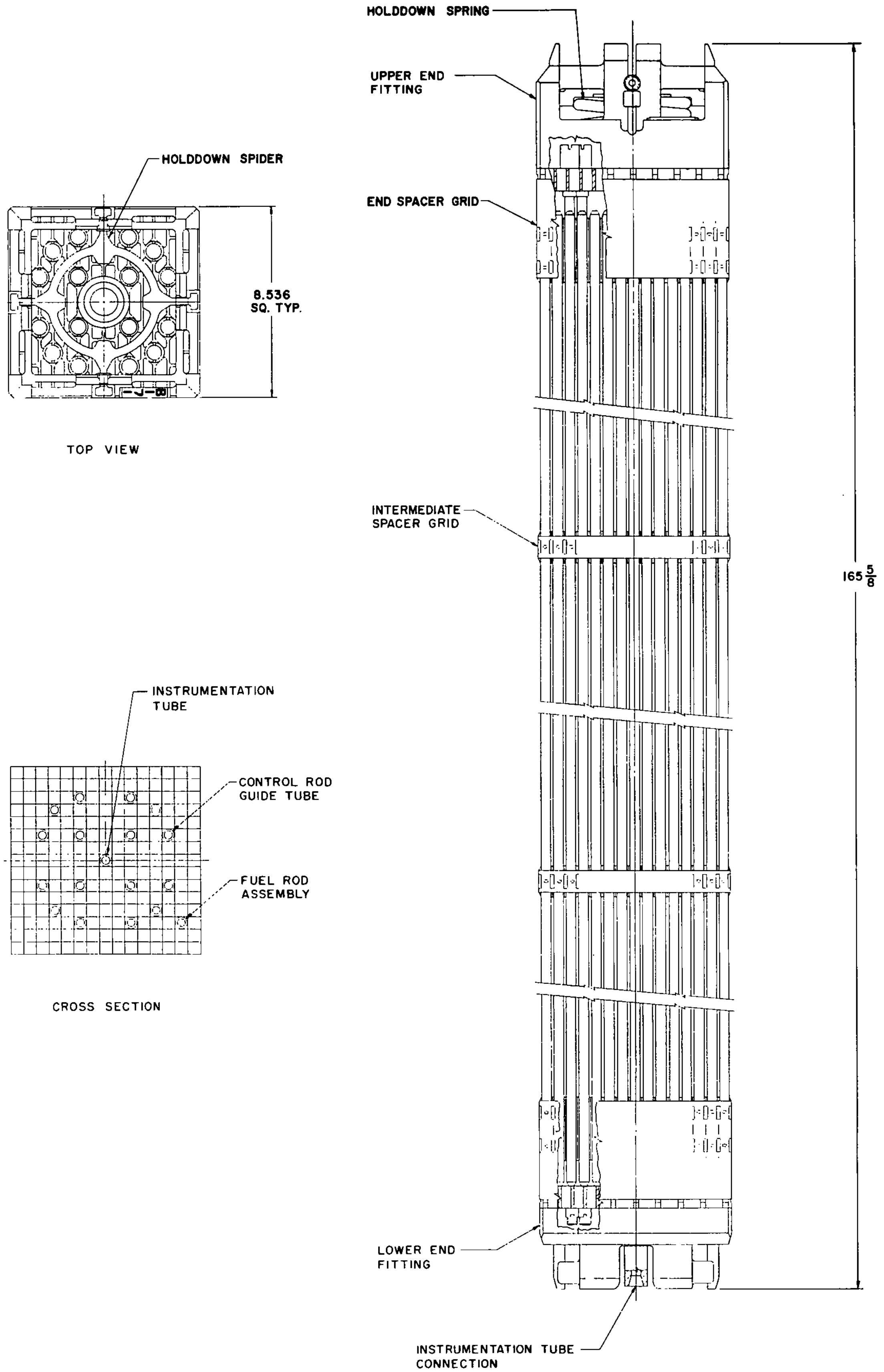
$$\text{Margin} = \frac{\text{level to obtain limit} - \text{Duke level}}{\text{Duke level}} \times 100.$$



REACTOR VESSEL AND INTERNALS

OCONEE NUCLEAR STATION

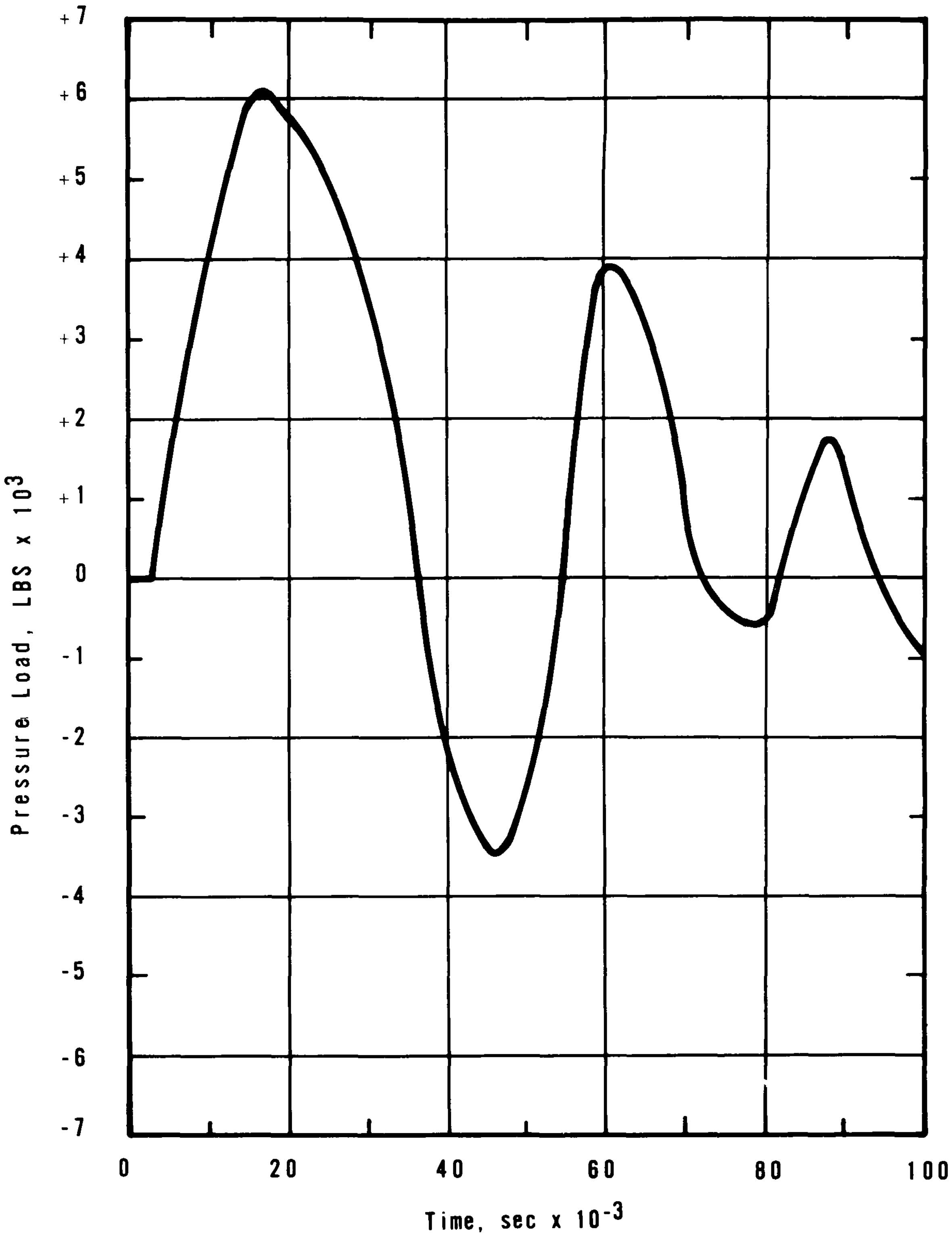
Figure 4C - 1



CANLESS FUEL ASSEMBLY

OCONEE NUCLEAR STATION

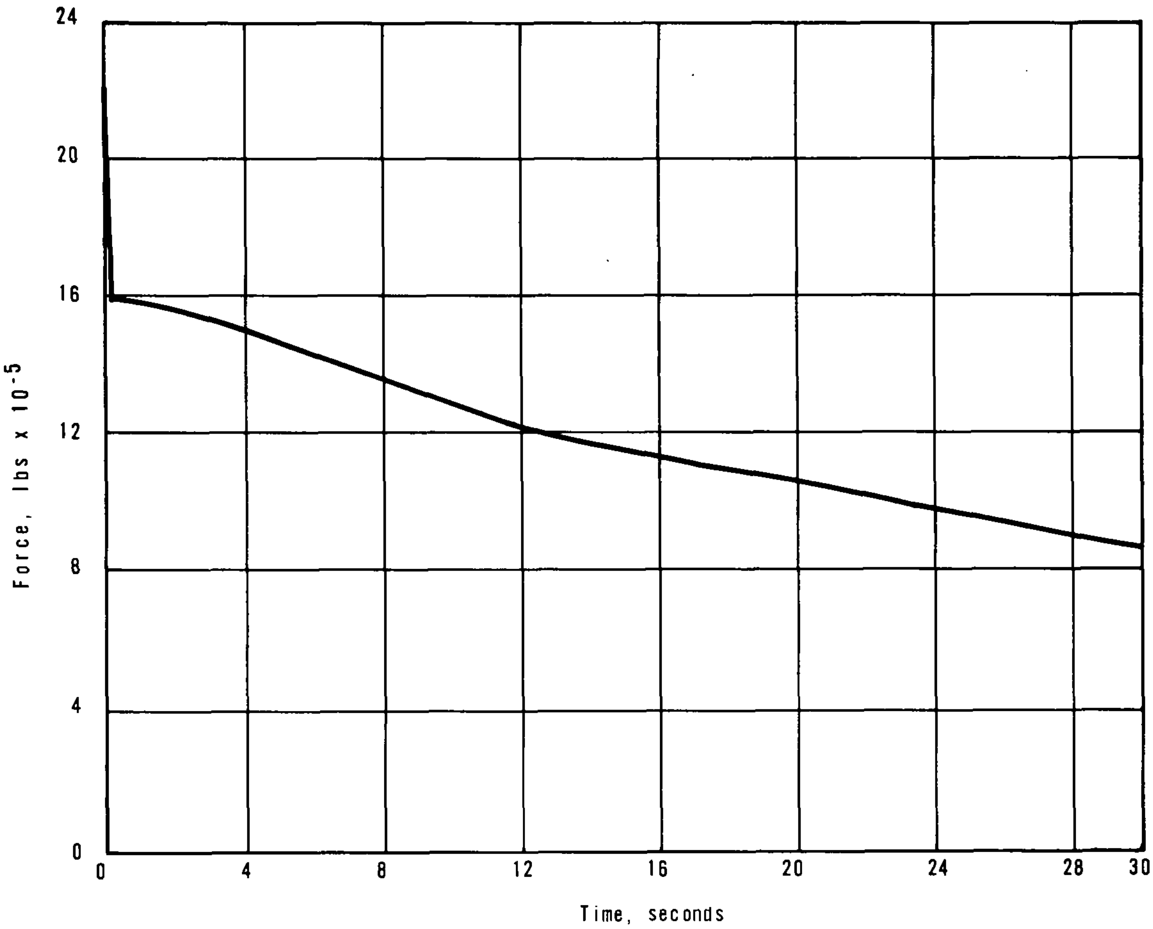
Figure 4C - 2



VERTICAL CONTACT LOADING CURVE

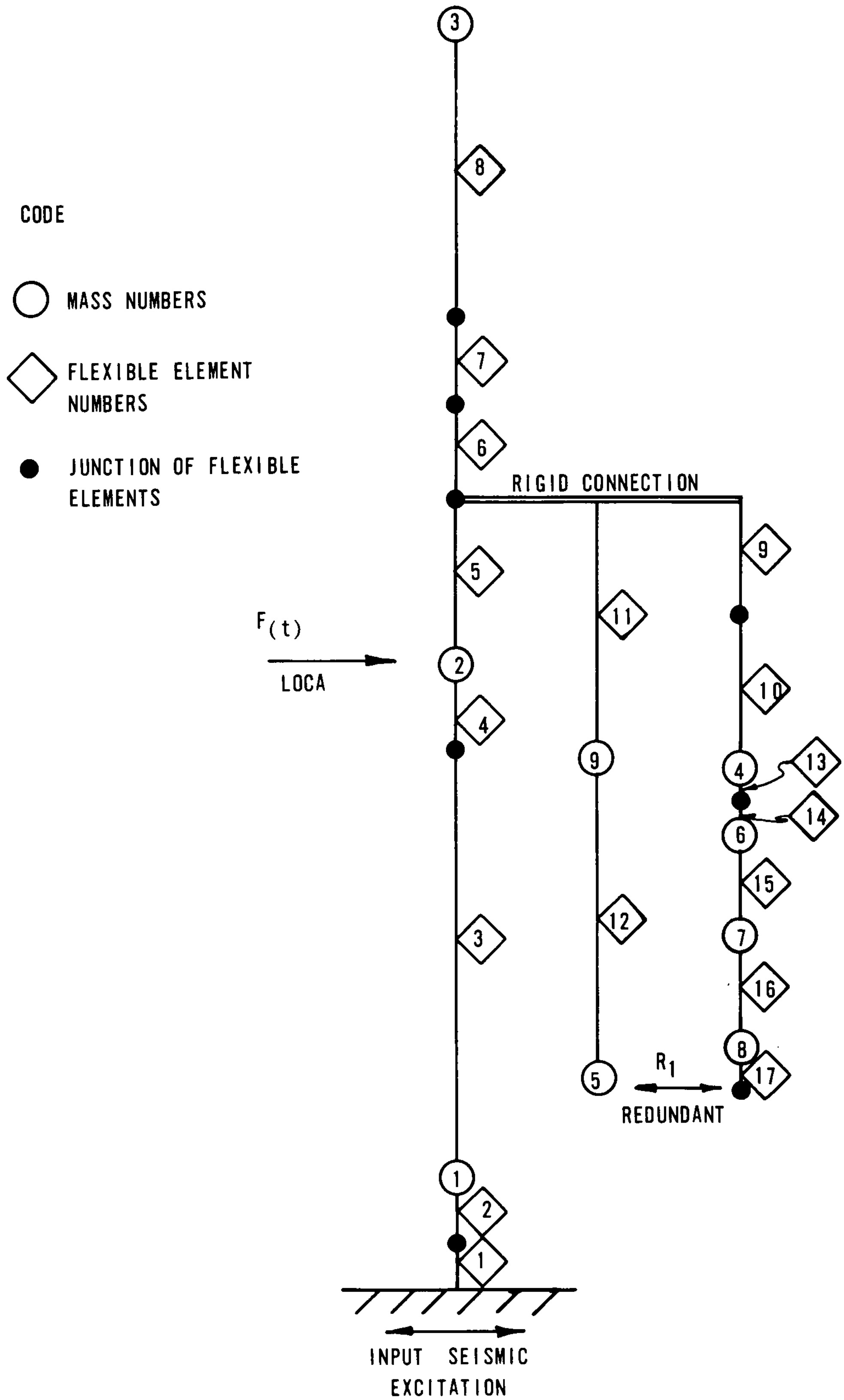
OCONEE NUCLEAR STATION

Figure 4C - 3



THRUST-TIME CURVE FOR CIRCUMFERENTIAL
OR LONGITUDIAL BREAK OF 36 - INCH-ID PIPE

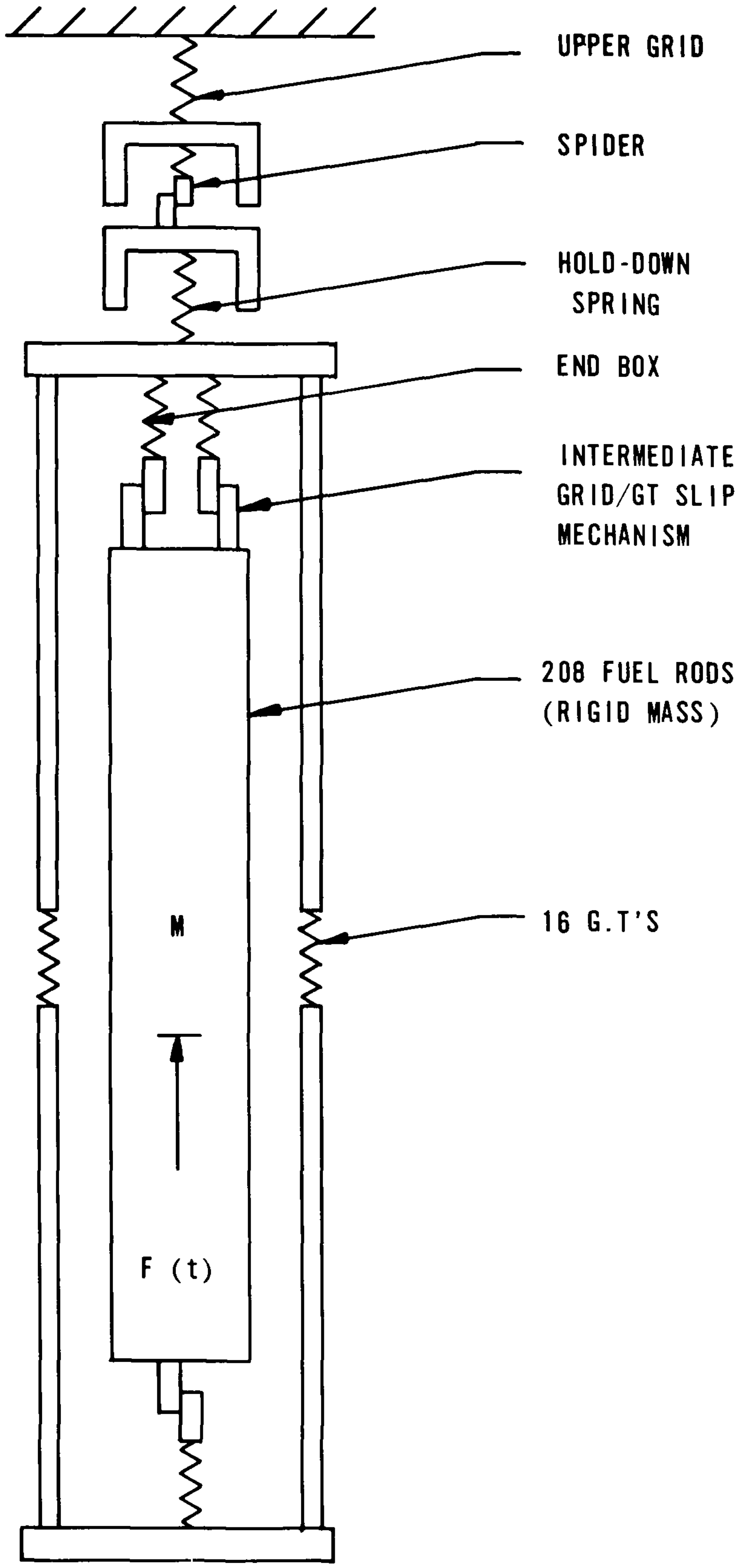
OCONEE NUCLEAR STATION
Figure 4C - 4



FIRST - SEGMENT MODEL

OCONEE NUCLEAR STATION

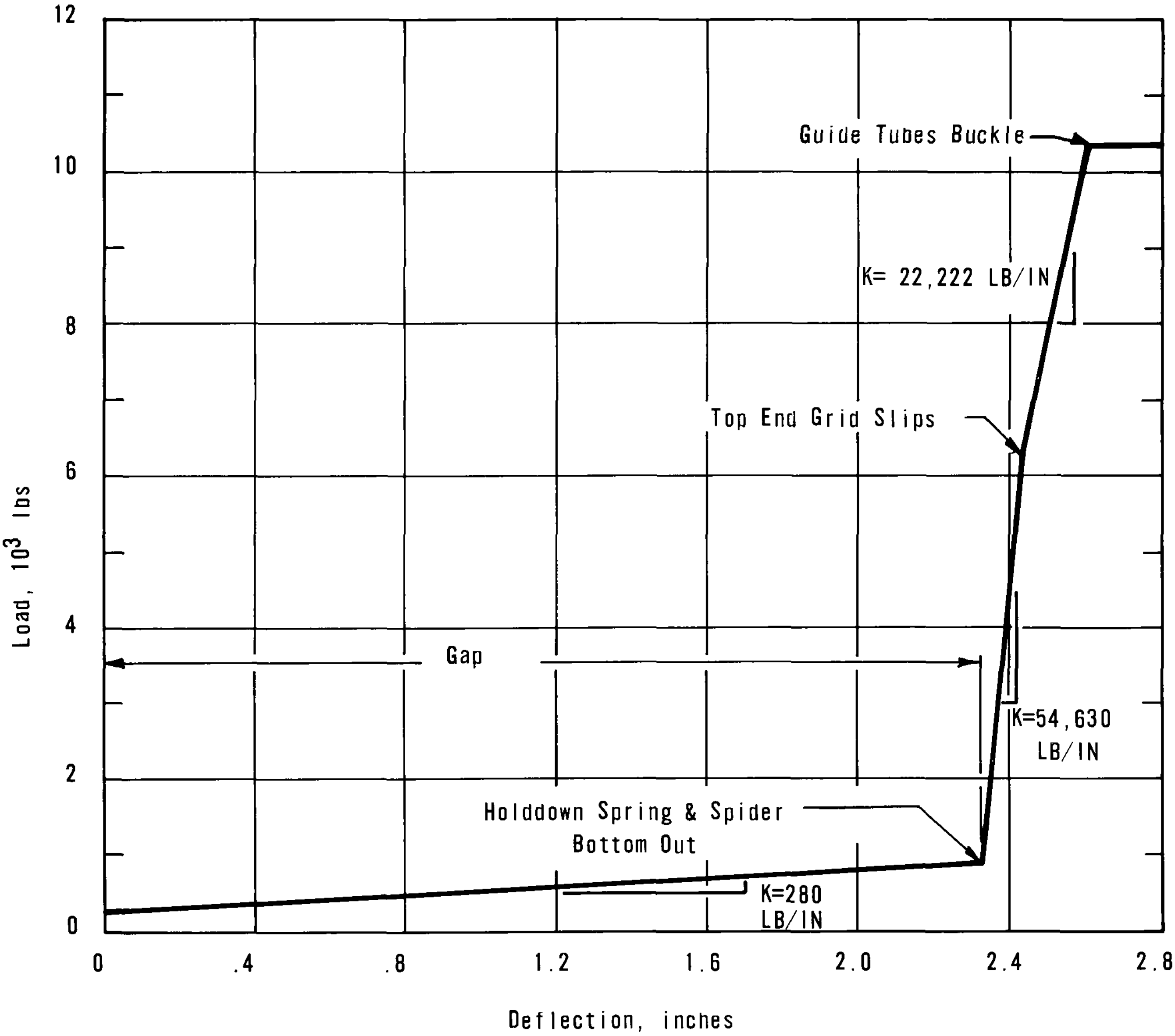
Figure 4C - 5



FUEL ASSEMBLY CONTACT MODEL

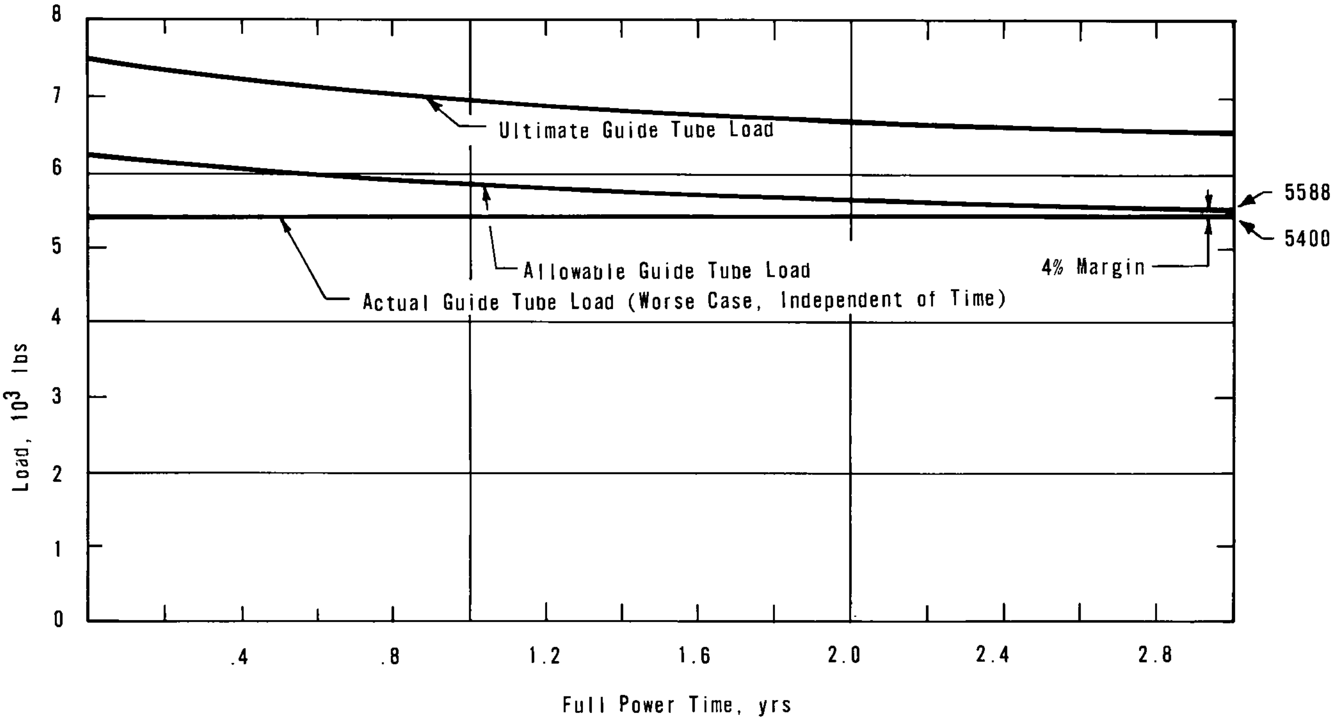
OCONEE NUCLEAR STATION

Figure 4C - 6



BEGINNING – OF – LIFE SPRING CURVE

OCONEE NUCLEAR STATION
Figure 4C - 7
Rev. 16 7/30/71

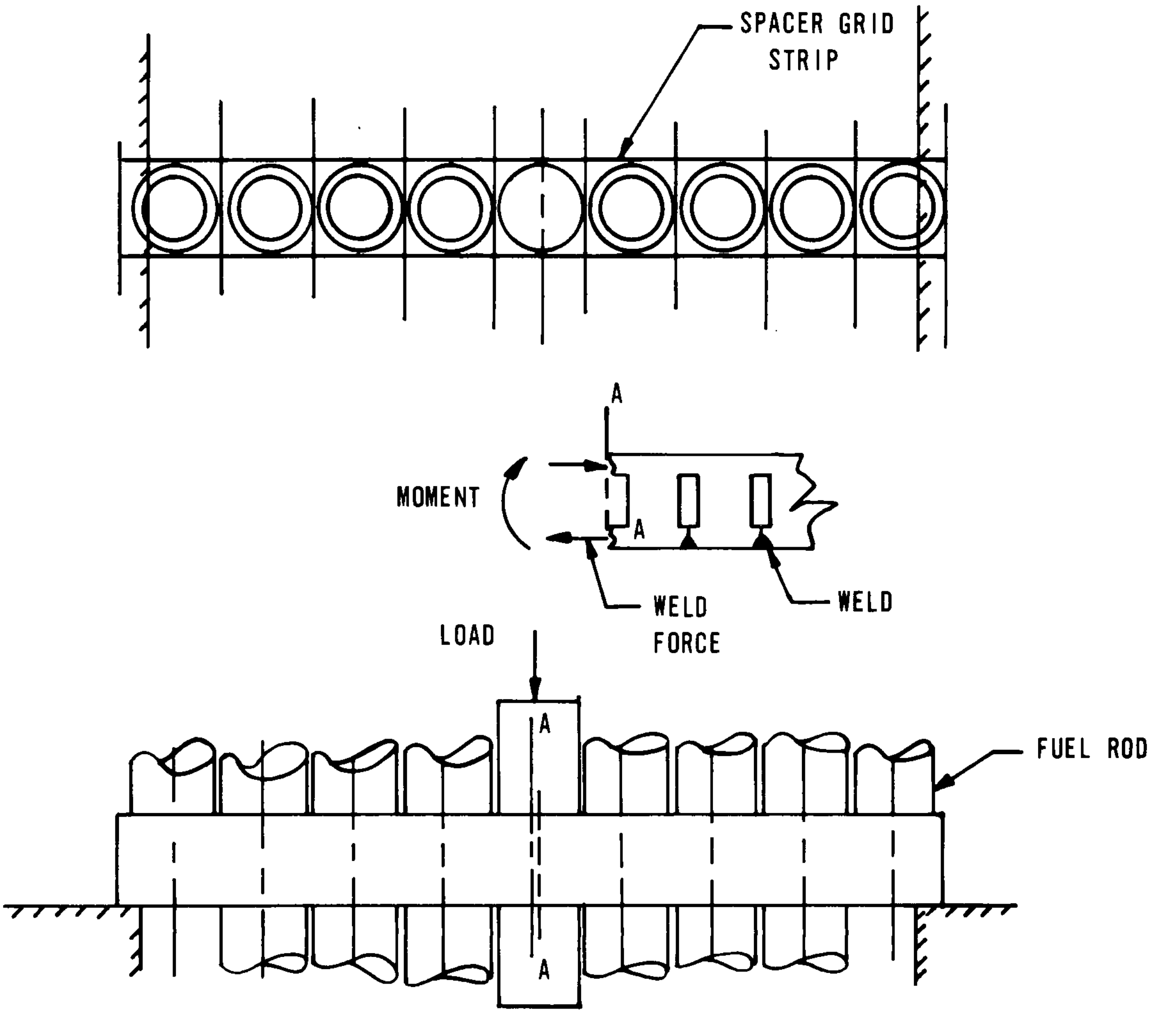


VERTICAL CONTACT ANALYSIS RESULTS

OCONEE NUCLEAR STATION

Figure 4C - 8

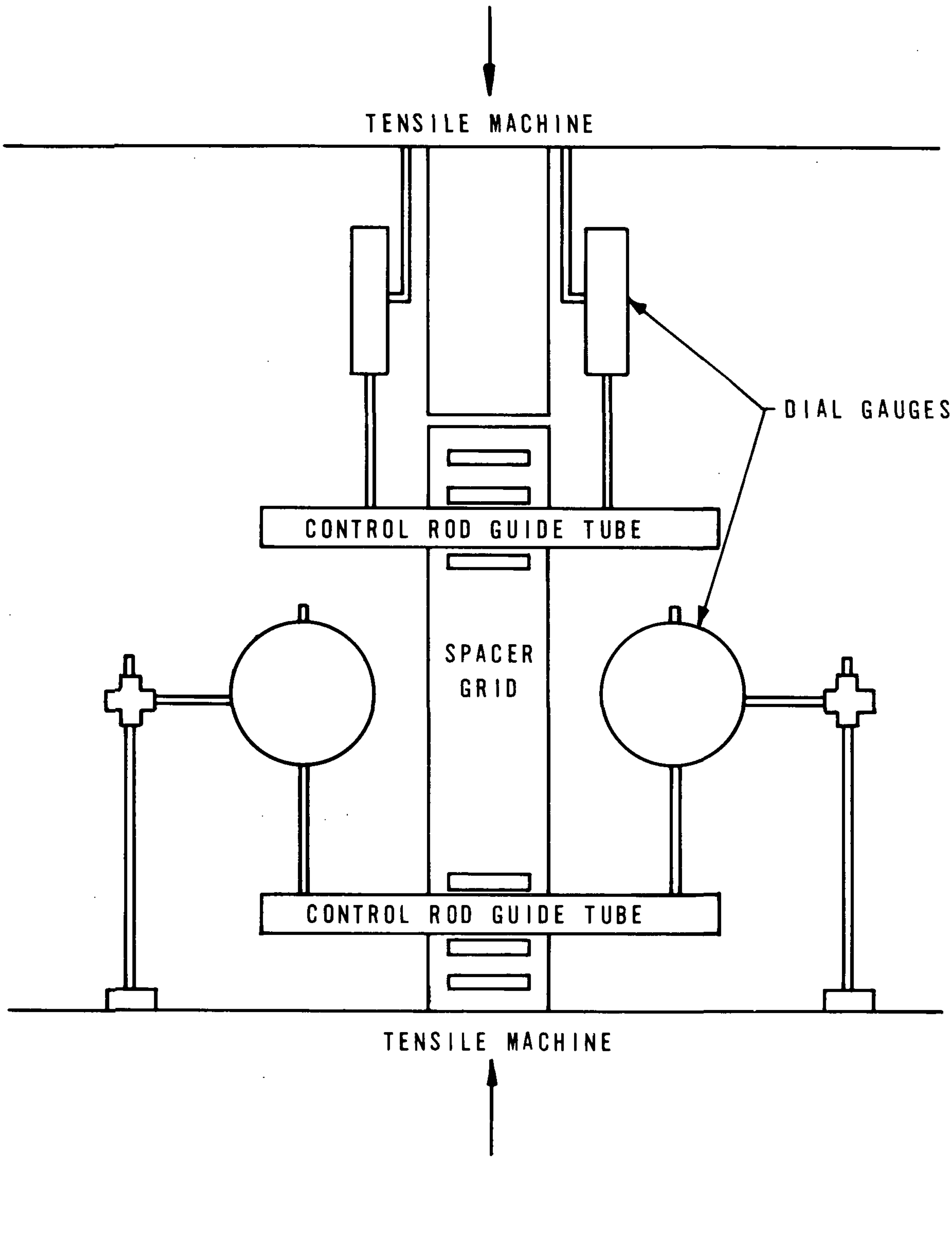
Rev. 16 7/30/71



SPACER GRID WELD TESTS

OCONEE NUCLEAR STATION

Figure 4C - 9



SPACER GRID COMPRESSION TEST MOUNT

OCONEE NUCLEAR STATION

Figure 4C - 10

LIST OF EFFECTIVE PAGES
FSAR SECTION 5

Structures

Page	Revision
LOEP-- 1 of 3	Rev. 37
LOEP-- 2 of 3	Rev. 33
LOEP-- 3 of 3	Rev. 37
5-i	Rev. 5
5-ii	Rev. 9
5-iii	Rev. 10
5-iv	Rev. 5
5-v	Rev. 21
5-vi	Original
5-1	Rev. 18
5-2	Original
5-3	Rev. 4
5-3a	Rev. 4
5-3b	Rev. 4
5-4	Original
5-5	Rev. 1
5-6	Original
5-7	Original
5-8	Original
5-9	Original
5-10	Original
5-11	Original
5-12	Rev. 4
5-12a	Rev. 4
5-13	Original
5-14	Rev. 4
5-14a	Rev. 5
5-14b	Rev. 5
5-14c	Rev. 5
5-14d	Rev. 5
5-15	Rev. 6
5-16	Original
5-17	Rev. 6
5-17a	Rev. 6
5-18	Rev. 6
5-19	Rev. 4
5-20	Original
5-21	Original
5-22	Original
5-23	Original
5-24	Original
5-25	Original
5-26	Original
5-27	Original
5-28	Original
5-29	Original
5-30	Original
5-31	Original

Structures

Page	Revision	Page	Revision
5-32	Rev. 5	5-47	Original
5-32a	Rev. 5	5-48	Original
5-32b	Rev. 5	5-49	Rev. 7
5-32c	Rev. 5	5-50	Rev. 21
5-32d	Rev. 5	5-51	Rev. 9
5-32e	Rev. 5	5-51a	Rev. 9
5-32f	Rev. 5	5-52	Rev. 5
5-32g	Rev. 5	5-53	Rev. 20
5-33	Rev. 1	5-54	Rev. 16
5-34	Rev. 4	5-54a	Rev. 20
5-34a	Rev. 4	5-54b	Rev. 20
5-34b	Rev. 4	5-55	Rev. 33
5-35	Original	5-56	Rev. 20
5-36	Original	5-57	Rev. 18
5-37	Original	5-58	Original
5-38	Original	5-59	Rev. 30
		5-59a	Rev. 30
5-39	Rev. 1	5-60	Rev. 1
5-40	Original	5-61	Original
5-41	Original	5-61a	Rev. 10
5-42	Original	5-62	Rev. 4
5-43	Original	5-62a	Rev. 4
5-44	Rev. 4	5-62b	Rev. 4
5-45	Rev. 4	5-62c	Rev. 10
5-46	Original	5-62d	Rev. 10

Structures

Page	Revision	Page	Revision
5-62e	Rev. 10	Fig. 5-9A	Rev. 6
5-63	Rev. 37	Fig. 5-9B	Rev. 6
5-64	Rev. 37	Fig. 5-10(1)	Rev. 5
5-65	Rev. 37	Fig. 5-10(2)	Rev. 5
5-66	Rev. 37	Fig. 5-11	Original
Fig. 5-1	Original	Fig. 5-12	Original
Fig. 5-2	Rev. 6	Fig. 5-13	Original
Fig. 5-3	Original	Fig. 5-14	Original
Fig. 5-4(1)	Rev. 4	Fig. 5-15	Original
Fig. 5-4(2)	Rev. 4	Fig. 5-16	Rev. 37
Fig. 5-5	Original	Fig. 5-17	Rev. 37
Fig. 5-6(1)	Original	Fig. 5-18	Rev. 37
Fig. 5-6(2)	Original	Fig. 5-19	Rev. 37
		Fig. 5-19a	Rev. 37
Fig. 5-6(3)	Original	Fig. 5-20	Original
Fig. 5-6(4)	Original	Fig. 5-21(1)	Rev. 5
Fig. 5-7(1)	Original	Fig. 5-21(2)	Rev. 5
Fig. 5-7(2)	Original	Fig. 5-21(3)	Rev. 5
Fig. 5-7(3)	Original	Fig. 5-21(4)	Rev. 15
Fig. 5-7(4)	Original	Fig. 5-22	Rev. 4
Fig. 5-7(5)	Original	Fig. 5-23	Rev. 21
Fig. 5-7(6)	Original		
Fig. 5-8	Original		
Fig. 5-9(1)	Original		
Fig. 5-9(2)	Original		

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

LIST OF TABLES

LIST OF FIGURES
(At Rear of Section)

Rev. 21. 7/26/72
Rev. 4. 4/20/70
Rev. 5. 5/25/70
Rev. 6. 6/22/70
Rev. 9. 8/11/70
Rev.15. 12/30/70

INTRODUCTION

This section applies to the Oconee Reactor Buildings and associated structures for Units 1, 2, and 3. This section is written in the singular tense for clarity but applies fully to each of the three Reactor Buildings unless noted otherwise.

5 STRUCTURES

5.1 REACTOR BUILDING

5.1.1 DESIGN BASIS

The Reactor Building completely encloses the reactor coolant system to minimize release of radioactive material to the environment should a serious failure of the reactor coolant system occur. The structure provides adequate biological shielding for both normal operation and accident situations. The Reactor Building is designed for an internal pressure of 59 psig. The
18. leakage rate will not exceed 0.25 percent by volume in 24 hours under the conditions of the maximum hypothetical accident as decribed in Section 14.

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss-of-coolant accident as defined in Section 14 with no loss of integrity. In this event, the total energy contained in the water of the reactor coolant system is assumed to be released into the Reactor Building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safeguards, and the combined influence of energy sources and heat sinks.

Energy is available for release into the containment structure from the following sources:

Reactor Coolant System Stored Heat
Reactor Stored Heat
Reactor Decay Heat
Metal-Water Reactions

The energy release and the containment pressure transient curve are shown in Section 14.

The design of the engineered safeguards systems and their operation is discussed more fully in Section 6; only their relation to the basis of Reactor Building design is discussed below. The engineered safeguards systems are provided to limit the consequences of an accident. Their energy removal capabilities limit the internal pressure after the initial peak so that Reactor Building design limits are not exceeded and the potential for release of fission products is minimized.

The emergency core cooling systems inject borated water into the reactor coolant system to remove core decay heat and to minimize metal-water reactions and the associated release of heat and fission products. Flashed primary coolant, reactor coolant system sensible heat, and core decay heat transferred to Reactor Building are removed by two engineered safeguards systems: the Reactor Building spray and/or the Reactor Building cooling systems.

The Reactor Building spray system removes heat directly from the Reactor Building atmosphere by cold water quenching of the Reactor Building steam.

The air recirculation and cooling systems remove heat directly from the Reactor Building atmosphere to the service water system with recirculating fans and cooling coils.

5.1.2 DESIGN CRITERIA

5.1.2.1 General Description

The Reactor Building houses the reactor coolant system. Its purpose is to contain any accidental release of radioactivity from the reactor coolant system. It is designated as a Class I Structure.

The basic design criteria is that the integrity of the liner plate be guaranteed under all loading conditions and the structure shall have a low-strain elastic response such that its behavior will be predictable under all design loadings.

The structure consists of a post-tensioned reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete foundation slab as shown in Figure 5-1. The entire interior surface of the structure is lined with a 1/4 inch thick welded ASTM A36 steel plate to assure a high degree of leak tightness. Numerous mechanical and electrical systems penetrate the Reactor Building wall through welded steel penetrations as shown in Figures 5-2 and 5-3. The mechanical penetrations and access openings are designed, fabricated, inspected, and installed in accordance with Subsection B, Section III, of the ASME Pressure Vessel Code.

Principal dimensions are as follows:

Inside Diameter	116 Ft
Inside Height (Including Dome)	208½ Ft
Vertical Wall Thickness	3-3/4 Ft
Dome Thickness	3-1/4 Ft
Foundation Slab Thickness	8-1/2 Ft
Liner Plate Thickness	1/4 Inch
Internal Free Volume	1,910,000 Cu Ft

The Reactor Building is shown in Figures 1-2 through 1-9.

In the concept of a post-tensioned Reactor Building, the internal pressure load is balanced by the application of an opposing external pressure type load on the structure. Sufficient post-tensioning is used on the cylinder and dome to more than balance the internal pressure so that a margin of external pressure exists beyond that required to resist the design accident pressure. Nominal, bonded reinforcing steel is also provided to distribute strains due to shrinkage and temperature. Additional bonded reinforcing steel is used at penetrations and discontinuities to resist local moments and shears.

The internal pressure loads on the foundation slab are resisted by both the external bearing pressure due to dead load and the strength of the reinforced concrete slab. Thus, post-tensioning is not required to exert an external pressure for this portion of the structure.

The post-tensioning system consists of:

a. Three groups of 54 dome tendons oriented at 120° to each other for a total of 162 tendons anchored at the vertical face of the dome ring girder.

b. 176 vertical tendons anchored at the top surface of the ring girder and at the bottom of the base slab.

c. Six groups of 105 hoop tendons plus two additional tendons enclosing 120° of arc for a total of 632 tendons anchored at the six vertical buttresses.

Each tendon consists of ninety 1/4 inch diameter wires with buttonheaded BBRV type anchorages, furnished by The Prescon Corporation. The tendons are housed in spiral wrapped corrugated thin wall sheathing. After fabrication, the tendon is shop dipped in a petrolatum corrosion protection material, bagged and shipped. After installation, the tendon sheathing is filled with a corrosion preventive grease.

Ends of all tendons are covered with pressure tight grease filled caps for corrosion protection.

ASTM A615, Grade 60 reinforcing steel, mechanically spliced with T-series CADWELDS, is used throughout the foundation slab and around the large penetrations. A615, Grade 40 steel is used for the bonded reinforcing throughout the cylinder and dome as crack control reinforcing. At areas of discontinuities where additional steel is used, such steel is generally A615, Grade 60 to provide an additional margin of elastic strain capability.

The 1/4 inch thick liner plate is attached to the concrete by means of an angle
grid system stitch welded to the liner plate and embedded in the concrete. The
details of the anchoring system are provided in Figure 5-1. The frequent anchor-
ing is designed to prevent significant distortion of the liner plate during accident
conditions and to insure that the liner maintains its leak tight integrity. The
design of the liner anchoring system also considers the various erection tolerances
and their effect on its performance. The liner plate is coated on the inside with
3 mils of inorganic zinc primer and 4 mils of Phenoline 305 for corrosion protec-
4. tion. See Table 5-0 for Reactor Building coatings. There is no paint on the side
in contact with concrete.

The concrete used in the structure is made with crushed marble aggregate obtained from Blacksburg, South Carolina. Such aggregate produces an excellent high strength, dense, sound concrete. The design strengths are 5000 psi at 28 days for the shell and foundation slab.

Personnel and equipment access to the structure is provided by a double door per-
sonnel hatch with double seals on the outer door and by a 19 ft. - 0 in. clear
diameter double gasketed single door equipment hatch as shown in Figure 5-3. A
double door emergency personnel escape hatch is also provided. These hatches are
designed and fabricated of A516, Grade 70 firebox quality steel made to A300 spec-
1 ification, Charpy V-notch impact tested to 0°F in accordance with Section

Table 5-0

Reactor Building Coatings

Coatings System/Materials Used		Physical/Chemical Characteristics	Location	Function	Performance Under Accident (LOCA) Conditions
Carbo Zinc II	3 mils	Self-curing ethyl silicate inorganic zinc, U S Patent 3,056,684			1. The decontamination factor for Phenoline 305 is 325. Test methods described in Oak Ridge National Laboratory Reports ORNL-3589, 3916 and others.
Phenoline 305 Finish	4 mils	Catalized modified phenolic			2. Carbo Zinc II withstands in excess of 3×10^9 Roentgens when irradiated in water. There is no serious damage to Phenoline 305 at 6×10^9 Roentgens when irradiated in air. Phenoline 305 withstands in excess of 2×10^9 Roentgens irradiation in water.
System #30	7 mils (Min. Dry)		Interior surfaces Reactor Building liner plate and appurtenances. Structural steel inside Reactor Building. Polar Crane.	To protect blast cleaned steel against corrosion under normal operation conditions. To aid in housekeeping and to provide surfaces that can be readily decontaminated in case of local or minor releases of radioactive materials.	3. System has satisfactorily withstood autoclave tests designed to simulate LOCA conditions as follow: a. Test specimens: Coating system applied to sandblasted steel coupons. b. Water chemistry: 3000 ppm boron as boric acid in water; also 3% boric acid. c. Temperature: For 3000 ppm boron - 3 hours at 285° F - 290° F 2 days at 200° F 6 days at 150° F 4 days at 130° F For 3% boric acid - 3 hours at 75° F to 300° F 3 hours at 300° F 3 hours at 300° F to 180° F 15 hours cooling to ambient Total 24 hour cycle repeated ten times System showed no loss of adherence or errosion of material from surface in autoclave exposure to steam and high temperature spray water solution. We understand testing performed by ANS subcommittee for Protective Coatings for Reactor Containment Facilities and by Dr C D Watson at Oak Ridge did not disclose any significant difference between results of static autoclave exposure and autoclave exposure using a spray of solution on panels. On this basis either static or dynamic exposure to spray solution is considered to be acceptable as basis for testing. 4. We do not have available test results on jet impingement effects; however, it is felt that there is no coating system available which would withstand a high temperature, high velocity steam jet. We believe that the assumption of large scale, rapid LOCA by means of a double-ended pipe failure or otherwise, negates the possibility of a concentrated local jet impinging on a coated steel area of substantial size. Therefore, we believe the autoclave tests in which specimens were subjected to steam and water at elevated temperatures more nearly approximate overall building environment under LOCA conditions than would a local steam jet application. We understand ANS subcommittee found no system for coating steel or concrete for resisting steam jet impingement and therefore has established no standards for this condition of exposure.

Rev. 4 4/20/70
New Page

Table 5-0 - Continued

Reactor Building Coatings

Coatings System/Materials Used	Physical/Chemical Characteristics	Location	Function	Performance Under Accident (LOCA) Conditions
Carboline 195 Surfacer 8 mils	Modified epoxy-polyamide filler			1. Decontamination factor for Phenoline 305 is 325 (see System #30).
Phenoline 305 Finish 4 - 8 mils	Catalized modified phenolid			2. Carboline 195 has performed well as a surfacer in irradiation tests of a number of top coats including Phenoline 305. See System #30 for irradiation characteristics of Phenoline 305.
Systems #31a, b, c 12 -16 mils (Min. Dry)		Reactor Building interior concrete surfaces.	To be used on concrete walls, ceilings, floors in moderate traffic areas to protect against penetration of radioactive materials into concrete, to prevent dusting, and to ease housekeeping and decontamination. Color improves light reflectance of concrete.	3. System has satisfactorily withstood autoclave tests designed to simulate LOCA accident conditions as follows: a. Test specimens: Prepared concrete coupons. b. Water chemistry: Same as for System #30. c. Temperature: 2 hours at 75° F - 300° F 14 hours at 300° F 2 hours at 75° F 4 hours cooling to ambient 4. See note 4, Systems #30.
Carboline 195 Surfacer 8 mils	Modified epoxy-polyamide filler			1. Decontamination factor for Phenoline 300 is 1700.
Phenoline 300 8-16 mils	Catalized epoxy			2. Phenoline 300 systems on concrete withstand irradiation up to 4.7×10^9 Roentgens when irradiated in water.
Systems #31d, e 16-24 mils (Min. Dry)		Reactor Building interior concrete floors subject to heavy traffic and in certain trenches and sumps.	Protects against penetrations of radioactive materials into concrete, dusting, physical damage and wear to concrete. Surface is easy to clean, decontaminate.	3. This system, as such, was not tested under autoclave conditions; however, Phenoline 300 is essentially heavy-duty Phenoline 305 modified for tank lining and heavy duty-floor coating service. It has superior resistance to penetration by chemicals and moisture over a wide range of temperatures and consistently performs better than Phenoline 305 when irradiated in water. 4. See note 4, Systems #30.

III of the ASME Pressure Vessel Code. All piping penetrations are furnished to the same requirements.

Structural brackets provided for the Reactor Building polar crane runway are fabricated of A36 steel shapes and A516, Grade 70 insert plates (Figure 5-1). Structural brackets and thickened plates are shop fabricated, stress relieved and shipped to the jobsite for welding into the 1/4 inch liner plate similar to the penetration assemblies.

The strength of the Reactor Building at working stress and overall yielding is compared to various loading combinations to assure safety. The Reactor Building is examined with respect to strength, the nature and the amount of cracking, the magnitude of deformation, and the extent of corrosion to assure proper performance. The structure is designed and constructed in accordance with design criteria based upon ACI 318-63, ACI 301, and ASME Pressure Vessel Code, Sections III, VIII and IX to meet the performance and strength requirements prior to prestressing, at transfer of prestress, under sustained prestress, at design loads and at yield loads.

The structure is analyzed using a finite element computer program for individual and various combinations of loading cases of dead load, live load, prestress, temperature and pressure. The computer output includes direct stresses, shear stresses, principal stresses and displacements of each nodal point.

Stress plots which show the total stresses from appropriate combinations of loading cases are made and areas of high stress are identified. The modulus of elasticity is corrected to account for the nonlinear stress-strain relationship at high compression, if necessary. Stresses then are recomputed if there are sufficient areas which require attention.

In order to consider creep deformation, the modulus of elasticity of concrete under sustained loads such as dead load and prestress is differentiated from the modulus of elasticity of concrete under instantaneous loads such as internal pressure and earthquake loads.

The forces and shears are added over the cross section and the total moment, axial force and shear are determined. From these values, the straight-line elastic stresses are computed and compared to the allowable values. The ACI 318-63 design methods and allowable stresses are used for concrete and prestressed and nonprestressed reinforcing steel except as noted in the design criteria.

It is the intent of the criteria to provide a structure of unquestionable integrity that will meet the postulated design conditions with a low strain elastic response. The Oconee Reactor Building meets these criteria because:

a. The design criteria are in general based on the proven stress, strain, and minimum proportioning requirements of the ACI or ASME

Codes. Where departures or additions from these codes have been made, they have been done in the following manner:

1. The environmental conditions of severity of load cycling, weather, corrosion conditions, maintenance, and inspection for this structure have been compared and evaluated with those for code structures to determine the appropriateness of the modifications.

2. The consultant firm of T. Y. Lin, Kulka, Yang and Associate was retained to assist in the development of the criteria. In addition to assisting with the criteria submitted in the PSAR, they have been involved in the continuing updating of the criteria and the review of design methods to assure that the criteria were being implemented as intended.

3. Dr. Alan H. Mattock of the University of Washington was retained to assist in developing the proper design criteria for combined shear, bending and axial load.

4. All criteria, specifications and details relating to liner plate and penetrations and corrosion protection have been referred to Bechtel's Metallurgy and Q[illegible] Control Department. This department maintains a staff to advise the corporation on problems of welding, quality control, metallurgy and corrosion protection.

5. The design of the Oconee Reactor Building was continually reviewed as the criteria were improved for successive license applications to assure that this structure does meet the latest criteria.

b. The primary membrane integrity of the structure is provided by the unbonded post-tensioning tendons, each one of which is stressed to 80 percent of ultimate strength during installation and performs
1. at approximately 50 percent - 60 percent during the life of the structure. Thus, the main strength elements are individually proof-tested prior to operation of the plant.

c. 970 such post-tensioning elements have been provided, 162 in the dome, and 176 vertical and 632 hoop tendons in the cylinder. Any three adjacent tendons in any of these groups can be lost without significantly affecting the strength of the structure due to the load redistribution capabilities of the shell structure. The bonded reinforcing steel provided for crack control assures that this redistribution capability exists.

d. The unbonded tendons are continuous from anchorage to anchorage, being deflected around penetrations and isolated from secondary strains of the shell. Thus, the membrane integrity of the shell can be assured regardless of conditions of high local strains.

e. The unbonded tendons exist in the structure at a slightly ever-decreasing stress due to relaxation of the tendon and creep of the concrete and, even during pressurization, are subject to a stress change of very small magnitude (2 percent to 3 percent of ultimate strength). Thus, the main structural system is never subject to large changes in load, even during accident conditions.

f. The concrete portion of the structure, similar to the tendons, is subject to the highest state of stress during the initial post tensioning. During pressurization, it is subject to a large change in load (or state of stress) but the change is, in general, a decrease in load. The large membrane compressive forces are diminished, and replaced, by relatively small radial pressures and stresses.

g. The deformations of the structure during plant operation, or due to accident conditions, are relatively minor due to the low strain behavior of the concrete. The largest deformations occur at the time of initial post-tensioning and shortly thereafter, prior to operation. This low strain behavior, and the inherent strength of the structure, permit the anchoring of all piping penetrating the structure directly to the shell. Such details (see Figure 5-2) eliminate the use of expansion bellow seals and significantly reduce the likelihood of leaks developing at the penetrations.

5.1.2.2 Loads Prior to Prestressing

Under this condition the structure is designed as a conventionally reinforced concrete structure. It is designed for dead load, live loads (including construction loads), and a reduced wind load. Allowable stresses are according to ACI 318-63.

5.1.2.3 Loads at Transfer of Prestress

The Reactor Building is checked for prestress loads and the stresses compared with those allowed by ACI 318-63 with the following exceptions: ACI 318-63, Chapter 26, allows concrete stress of $0.60f'_{ci}$ at initial transfer. In order to limit creep deformations, the membrane compression stress is limited to $0.30f'_{ci}$ whereas in combination with flexural compression the maximum allowable stress will be limited to $0.60f'_{ci}$ per ACI 318-63.

For local stress concentrations with nonlinear stress distribution as predicted by the finite element analysis, $0.75f'_{ci}$ is permitted when local reinforcing is included to distribute and control these localized strains. These high local stresses are present in every structure but they are seldom identified because of simplifications made in design analysis. These high stresses are allowed because they occur in a very small percentage of the cross section, are confined by material at lower stress and would have to be considerably greater than the values allowed before significant local plastic yielding would result. Bonded reinforcing is added to distribute and control these local strains.

Membrane tension and flexural tension are permitted provided they do not jeopardize the integrity of liner plate. Membrane tension is permitted to occur during the post-tensioning sequence but will be limited to $1.0\sqrt{f'_{ci}}$. When there is flexural tension but no membrane tension, the section is designed in accordance with Section 2605(a) of the ACI Code. The stress in the liner plate due to combined membrane tension and flexural tension is limited to 0.5 fy.

Shear criteria are in accordance with the ACI 318-63 Code, Chapter 26, as modified by the equations in 5.1.2.6 using a load factor of 1.5 for shear loads.

5.1.2.4 Loads Under Sustained Prestress

The conditions for design and the allowable stresses for this case are the same as above except that the allowable tensile stress in nonprestressed reinforcing is limited to 0.5 fy. ACI 318-63 limits the concrete compression to $0.45f'_c$ for sustained prestress load. Values of $0.30f'_c$ and $0.60f'_c$ are used as described above which bracket the ACI allowable value. However, with these same limits for concrete stress at transfer of prestress, the stresses under sustained load are reduced due to creep.

5.1.2.5 At Design Loads

This loading case is the basic "working stress" design. The Reactor Building is designed for the following loading cases:

a. $D + F + L + T_O$
b. $D + F + L + P + T_A + E$(or W)
c. $D + F + L + P'$

Where:

D = Dead Load
L = Appropriate Live Load
F = Appropriate Prestressing Load
P = Pressure Load (Varies with time from design pressure to zero pressure)
T_O = Thermal Loads Due to Operating Temperature
T_A = Thermal Loads Based on a Temperature Corresponding to a Pressure P
E = Design Earthquake
P' = Test Pressure = 1.15 P
W = Wind Load

Sufficient prestressing is provided in the cylindrical and dome portions of the vessel to eliminate membrane tensile stress (tensile stress across the entire wall thickness) under design loads. Flexural tensile cracking is permitted but is controlled by bonded reinforcing steel.

Under the design loads the same performance limits stated in 5.1.2.3 apply with the following exceptions:

a. If the net membrane compression is below 100 psi, it is neglected and a cracked section is assumed in the computation of flexural bonded reinforcing steel. The allowable tensile stresses in bonded reinforcing are 0.5 fy.

b. When the maximum flexural stress does not exceed 6 $\sqrt{f'_c}$ and the extent of the tension zone is not more than 1/3 the depth of the section, bonded reinforcing steel is provided to carry the entire tension in the tension block. Otherwise, the bonded reinforcing steel is designed assuming a cracked section. When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the bonded reinforcing steel is 0.5 fy minus the stress in reinforcing due to the thermal gradient as determined in accordance with the method of ACI-505.

c. The problem of shear and diagonal tension in a prestressed concrete structure should be considered in two parts: membrane principal tension and flexural principal tension. Since sufficient prestressing is used to eliminate membrane tensile stress, membrane principal tension is not critical at design loads. Membrane principal tension due to combined membrane tension and membrane shear is considered under 5.1.2.6.

   Flexural principal tension is the tension associated with bending in planes perpendicular to the surface of the shell and shear stress normal to the shell (radial shear stress). The present ACI 318-63 provisions of Chapter 26 for shear are adequate for design purposes with proper modifications as discussed under 5.1.2.6 using a load factor 1.5 for shear loads.

Crack control in the concrete is accomplished by adhering to the ACI-ASCE Code Committee standards for the use of reinforcing steel. These criteria are based upon a recommendation of the Prestressed Concrete Institute and are as follows:

0.25 percent reinforcing shall be provided at the tension face for small members

0.20 percent for medium size members

0.15 percent for large members

A minimum of 0.15 percent bonded steel reinforcing is provided in two perpendicular directions on the exterior faces of the wall and dome for proper crack control.

The liner plate is attached on the inside faces of the wall and dome. Since, in general, there is no tensile stress due to temperature on the inside faces, bonded reinforcing steel is not necessary at the inside faces.

5.1.2.6 Loads Necessary to Cause Structural Yielding

The structure is checked for the factored loads and load combinations that will cause structural yielding.

The load factors are the ratio by which loads will be multiplied for design purposes to assure that the load/deformation behavior of the structure is one of elastic, low-strain behavior. The load factor approach is being used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. It also places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads.

The final design of the structure satisfies the load combinations and factors shown in Appendix 5A, Section 2.2.

The load combinations, considering load factors referenced above, are less than the yield strength of the structure. The yield strength of the structure is defined as the upper limit of elastic behavior of the effective load carrying structural materials. For steels (both prestress and nonprestress) this limit is taken to be the guaranteed minimum yield given in the appropriate ASTM specification. For concrete, it is the ultimate values of shear (as a measure of diagonal tension) and bond per ACI 318-63 and the 28-day ultimate compressive strength for concrete in flexure (f'_c). The ultimate strength assumptions of the ACI Code for concrete beams in flexure are not allowed; that is, the concrete stress is not allowed to go beyond yield and redistribute at a strain of three or four times that which causes yielding.

The maximum strain due to secondary moments, membrane loads and local loads exclusive of thermal loads is limited to that corresponding to the ultimate stress divided by the modulus of elasticity (f'_c/E_c) and a straight-line distribution from there to the neutral axis assumed.

For the loads combined with thermal loads the peak strain is limited to 0.003 inch/inch. For concrete membrane compression, the yield strength is assumed to be $0.85f'_c$ to allow for local irregularities, in accordance with the ACI approach. The reinforcing steel forming part of the load carrying system is allowed to go to, but not to exceed, yield as is allowed for ACI ultimate strength design.

A further definition of yielding is the deformation of the structure which causes strains in the steel liner plate to exceed 0.005 inch/inch. The yielding of nonprestress reinforcing steel is allowed, either in tension or compression, if the above restrictions are not violated. Yielding of the prestress tendons is not allowed under any circumstances.

Principal concrete tension due to combined membrane tension and membrane shear, excluding flexural tension due to bending moments or thermal gradients, is limited to $3\sqrt{f'_c}$. Principal concrete tension due to combined membrane tension, membrane shear, and flexural tension due to bending moments or thermal gradients is limited to $6\sqrt{f'_c}$. When the principal concrete tension exceeds the limit of $6\sqrt{f'_c}$, bonded reinforcing steel is provided in the following manner:

a. Thermal Flexural Tension - Bonded reinforcing steel is provided in accordance with the methods of ACI-505. The minimum area of steel provided is 0.15 percent in each direction.

b. Bending Moment Tension - Sufficient bonded reinforcing steel is provided to resist the moment on the basis of cracked section theory using the yield stresses stated above with the following exception: When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the reinforcing steel is fy minus the stress in reinforcing due to the thermal gradient as determined in accordance with the methods of ACI-505.

Shear stress limits and shear reinforcing for radial shear are in accordance with Chapter 26 of ACI 318-63 with the following exceptions:

Formula 26-12 of the Code shall be replaced by

$$V_{ci} = Kb'd\sqrt{f'_c} + M_{cr}\left(\frac{V}{M'}\right) + V_i$$

Where:

$$K = \left[1.75 - \frac{0.036}{np'} + 4.0\ np'\right]$$

but not less than 0.6 for $p' \geq 0.003$.
For $p' < 0.003$, the value of K shall be zero.

$$M_{cr} = \frac{I}{Y}\left[6\sqrt{f'_c} + f_{pe} + f_n + f_i\right]$$

f_{pe} = Compressive stress in concrete due to prestress applied normal to the cross section after all losses (including the stress due to any secondary moment) at the extreme fiber of the section at which tension stresses are caused by live loads.

f_n = Stress due to axial applied loads (f_n shall be negative for tension stress and positive for compression stress).

f_i = Stress due to initial loads at the extreme fiber of a section at which tension stresses are caused by applied loads (including the stress due to any secondary moment. f_i shall be negative for tension stress and positive for compression stress).

$$n = \frac{505}{\sqrt{f'_c}}$$

$$p' = \frac{A'_s}{bd}$$

V = Shear at the section under consideration due to the applied loads.

M' = Moment at a distance d/2 from the section under consideration, measured in the direction of decreasing moment, due to applied loads.

V_i = Shear due to initial loads (positive when initial shear is in the same direction as the shear due to applied loads).

Lower limit placed by ACI 318-63 on V_{ci} as $1.7b'd\sqrt{f'_c}$ is not applied.

Formula 26-13 of the Code shall be replaced by

$$V_{cw} = 3.5b'd\sqrt{f'_c}\left(\sqrt{1 + \frac{f_{pc} + f_n}{3.5\sqrt{f'_c}}}\right)$$

The term f_n is as defined above. All other notations are in accordance with Chapter 26, ACI 318-63.

a. This formula is based on the recent tests and work done by Dr. A. H. Mattock of the University of Washington.

b. This formula is based on the commentary for proposal redraft of Section 2610, ACI-318, by Dr. A. H. Mattock, dated December 1962.

When the above-mentioned equations show that allowable shear in concrete is zero, radial horizontal shear ties are provided to resist all the calculated shear.

5.1.2.7 Other Design Loads

The Reactor Building shell is also designed for the following loads:

a. Dead load
b. Prestress forces
c. Live load including allowances for piping, ductwork and cable trays
d. Wind, including tornado
e. Earthquake
f. Thermal expansion of pipes attached to the Reactor Building wall

Transients resulting from the design basis accident and other lesser accidents are presented in Section 14 and serve as the basis for the Reactor Building design pressure of 59 psig and a design temperature of 286 F.

The external design pressure of the Reactor Building shell is 3 psig. This value is approximately 0.5 psig beyond the maximum external pressure that could be developed if the Reactor Building were sealed during a period of low barometric pressure and high temperature and, subsequently, the Reactor

Building atmosphere were cooled with a concurrent rise in barometric pressure. Vacuum breakers are not provided.

5.1.3 REACTOR BUILDING DESIGN ANALYSIS

The analysis for the Reactor Building falls into two parts, axisymmetric and nonaxisymmetric. The axisymmetric analysis is performed through the use of a finite element computer program for the individual loading cases of dead load, live load, temperature, prestress and pressure, as described in 5.1.3.1. The axisymmetric finite element approximation of the Reactor Building shell does not consider the buttresses, penetrations, brackets and anchors. These items of configuration, the lateral loads due to seismic or wind, and concentrated loads are considered in the nonaxisymmetric analysis described in 5.1.3.2.

This section discusses analytical techniques, references and design philosophy. The results of these analyses are shown in 5.1.4. The design criteria and analysis have been reviewed by Bechtel's consultants, T. Y. Lin, Kulka, Yang and Associate.

5.1.3.1 Axisymmetric Techniques

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints). Conventional analysis of frames and trusses can be considered to be examples of the finite element method. In the application of the method to an axisymmetric solid (eg, a concrete Reactor Building), the continuous structure is replaced by a system of rings of quadrilateral cross section which are interconnected along circumferential joints. Based on energy principals, work equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are unknowns of the system. The results of the solution of this set of equations are the deformation of the structure under the given loading conditions. For the output, the stresses are computed knowing the strain and stiffness of each element.

The finite element mesh used to describe the structure is shown in Figure 5-4. The upper portion and lower portion of the structure were analyzed independently to permit a greater number of elements to be used for those areas of the structure of major interest such as the ring girder area and the base of the cylinder. The finite element mesh of the structure base slab was extended down into the foundation material to take into consideration the elastic nature of the foundation material and its effect upon the behavior of the base slab. The tendon access gallery is separated from the Reactor Building base slab by 3" compressible material. No momemts or forces are transmitted from the base slab to the tendon access gallery. The maximum vertical elastic displacement of the base slab is one inch due to the maximum
4. loading combinations. The tendon access gallery was designed as a separate structure with no reactions being generated from the bedrock to the ring shaped gallery structure.

The finite element mesh for the reactor building does not include the interior structure. The interior structure was included in the finite element input as a lump weight. The finite elements provide stresses for axisymmetric loads. The stresses from the eccentric interior structure loads and earthquake loads are superimposed analytically to the finite element stresses. The final algebraic summation of all stresses was used to design the base slab.

4.

Stresses for Axisymmetric Loads	Stresses with Non-Axisymmetric Loads
11.0 kips/sq.ft.	26.0 kips/sq.ft.

The use of the finite element computer program permitted an accurate estimate of the stress pattern at various locations of the structure. The following material properties were used in the program for the various loading conditions:

	Load Conditions	
	D, F, T_0, T_A	P
$E_{concrete}$, Foundation (psi)	3.0×10^6	3.0×10^6
$E_{concrete}$, Shell (psi)	3.0×10^6	3.0×10^6
μ concrete (Poisson's Ratio)	0.17	0.17
α concrete (Coefficient of Expansion)	0.55×10^{-5}	-
$E_{subgrade}$ (psi)	4.5×10^6	4.5×10^6
E_{liner} (psi)	29×10^6	29×10^6
$f_{y\ liner}$ (psi)	36,000	36,000

The major benefit of the program is the capability to predict shears and moments due to internal restraint and the interaction of the foundation slab relative to the subgrade. The structure is analyzed assuming an uncracked homogeneous material. This is conservative because the decreased relative stiffness of a cracked section would result in smaller secondary shears and moments.

In arriving at the above-tabulated values of E, the effect of creep is included by using the following equation for long-term loads such as thermal load, dead load and prestress:

$$E_{cs} = E_{ci} \left(\varepsilon_i / (\varepsilon_s + \varepsilon_i)\right)$$

Where:

E_{cs} = sustained modulus of elasticity of concrete.

E_{ci} = instantaneous modulus of elasticity of concrete.

ε_i = instantaneous strain, inch/inch per psi.

ε_s = creep strain, inch/inch per psi.

The thermal gradients used for design are shown in Figure 5-5. The gradients for both the design accident condition and the factored load condition are based on the temperature associated with the factored pressure (factored loads are described in Appendix 5A). The design pressure and temperature of 59 psig and 286°F became 88.5 psig and 286°F at factored conditions.

The upper stress limit for a linear stress-strain relationship was assumed to be 3000 psi (0.6 f'_c) for use with analyses made by the use of the axisymmetric finite element analytical method. (The analyses referred to considered the concrete as uncracked and the analytical model is the entire containment.) However, the maximum predicted compressive stress was about 2559 psi. The load combination considered was $.95D+F+P+E'+T_A$ and the location for the predicted stress was for Section EF in ring girder. Therefore only the linear portion of the stress strain curve was used in the analyses that used the entire containment structure as a model.

The compressive stress and strain level is the highest (after the LOCA when temperature is still relatively high, 200°F, and pressure is dropping rapidly) at the inside face of the concrete at the edge of openings and also under the liner plate anchors. Neither concentration is a result of what may be considered a real load. In the case of an opening the real stress is a result of prestress, reduced pressure and dead load. Applying stress concentration factors to these loads still keeps the concrete in essentially the elastic range. When the strain and resulting stress from the thermal gradient are also multiplied by a stress concentration factor, the total strain and resulting stress will be above the linear stress range determined as by a uniaxial compression test. The relatively high stress level is not of real concern due to the following:

4.

1) The concrete affected is completely surrounded by either other concrete or the penetration nozzle and liner reinforcing plate. This confinement puts the concrete in triaxial compression and gives it the ability to resist forces far in excess of that indicated by a uniaxial compression test.

2) The high state of stress and strain exist at a very local area and really have no effect on the overall containment integrity.

However, to be conservative, reinforcing steel was placed in these areas and, also, the penetration nozzle will function as compressive reinforcement.

The concrete under the liner plate anchors has some limited yielding in order to get the necessary stress distribution required to resist the liner plate self-relieving loads.

The thermal loads are a result of the temperature differential within the structure. The design temperature stresses for this finite element analysis were prepared so that when temperatures are given at every nodal point, stresses are calculated at the center of each element.

Thus, the liner plate was handled as an integral part of the structure and was included in the finite element mesh of the Reactor Building, but having different material properties, and not as a mechanism which would act as an outside source to produce loading only on the concrete portion of the structure.

4. Figure 5-4, Sheet 1 of 2, shows the inclusion of the liner plate in the finite element mesh.

Under the design accident condition or factored load condition, cracking of the concrete at the outside face would be expected. The value of the sustained modulus of elasticity of concrete, E_{cs}, was used in ACI Code 505-54 to find the stresses in concrete, reinforcing steel and liner plate from the predicted design accident thermal loads and factored accident loads.

The isostress plots shown in Figures 5-6 and 5-7 do not consider the concrete cracked. The thermal stresses are combined from the individual isostress output for the cases of D + F + T and D + F + 1.5P + T. The first case is critical for concrete stresses and occurs after depressurization of the Reactor Building; the
4. second case is critical for the reinforcing stresses and it occurs when pressure and thermal loads are combined and cause cracking at the outside face. The loading cases for isostress plots shown in Figure 5-6 are D + F + 1.15P on Sheet 1, 0.95D + F + 1.5P + T on Sheet 2, D + F on Sheet 3, and T on Sheet 4. The loading cases for isostress plots shown in Figure 5-7 are D on Sheet 1, F on Sheet 2, T on Sheet 3, 0.95D + T on Sheet 4, F + 1.15P on Sheet 5, and F + 1.5P on Sheet 6.

The general approach of determining stresses in the concrete and reinforcement required the evaluation of the stress blocks of the cross section being analyzed.

The value of stresses was taken from the computer output in case of axisymmetric loading and from analytical solutions in case of nonaxisymmetric loading. Both computations were based on homogeneous materials; therefore, some adjustment was necessary to evaluate the true stress-strain conditions when cracks develop in the tensile zone of the concrete.

An equilibrium equation can be written considering the tension force in the reinforcement, the compressive force in the concrete and the axial force acting on the section. In this manner the neutral axis is shifted from the position defined by the computer analyses into a position which is the function of the amount of reinforcement, the modulus ratio, and the acting axial forces.

Large axial compressive force might prevent the existence of any tension stresses, as in the loading condition D + F + T; therefore, no self-relieving action exists; the stresses are taken directly from the computer output.

In the case of D + F + 1.5P + T, the development of cracks in the concrete decreases the thermal moment and this effect was considered; but the self-relieving properties of other loadings were not taken into account, even in places where they do exist, such as at discontinuities, e.g. the cylinder-base slab connection. This means that in analyzing the section, a reduced thermal moment was added to the unreduced moment caused by other loadings.

The thermal stresses in the containment are comparable to those developed in a reinforced concrete slab, which is restrained from rotation. The temperature varies linearly across the slab. The concrete will crack in tension and the neu-
5. tral axis will be shifted toward the compressive extreme fiber. The cracking will reduce the compression at the extreme fiber and increase the tensile stress in reinforcing steel.

The following analysis is based on the equilibrium of normal forces, therefore any normal force acting on the section must be added to the normal forces resulting from the stress diagram. The effects of Poisson's ratio are considered while the reinforcement is considered to be identical in both directions.

Stress - Strain relationship in compressed region of concrete:

$$E_c\Sigma_x = \sigma_x - \nu_c\sigma_y \qquad (1)$$

$$E_c\Sigma_y = -\nu_c\sigma_x + \sigma_y \qquad (2)$$

From the above equations (1) and (2):

$$\sigma_x = E_c\frac{\Sigma_x + \Sigma_y\nu}{1-\nu_c{}^2} \qquad (3)$$

$$\sigma_y = E_c\frac{\Sigma_y + \Sigma_x\nu}{1-\nu_c{}^2} \qquad (4)$$

Substituting,

$\sigma_x = \sigma_y = \sigma_c$ and $\Sigma_x = \Sigma_y = \Sigma_c$ into equations (3) and (4)

$$\sigma_c = E_c\Sigma_c\frac{1}{1-\nu_c} = 1.205\ E_c\Sigma_c \text{ (if } \nu_c = .17)$$

The reinforcement is acting in one direction, independently from the reinforcement in the perpendicular direction.

Example: If $E_c = 3 \times 10^6$ and $E_s = 29 \times 10^6$

$$n_R = \frac{29}{1.205 \times 3} = 8.02$$

The liner plate is acting in two directions, similar to the concrete except for the difference caused by the Poisson's ratios:

$$\sigma_L = E_s\Sigma_s\frac{1}{1-\nu_L} = 1.35\ E_s\Sigma_s \qquad \nu_L = .25$$

$$\nu_c = .17$$

$$n_L = \frac{1.35 \times 29}{1.205 \times 3} = 10.83$$

The following is an example taking Section GH of the use of the analytical method derived for $D + F + P + T_A + E$.

The concrete and reinforcement stresses are calculated by conventional methods, from the moment caused by loading other than thermal. The analyses assume homogeneous concrete sections. Those concrete and reinforcing steel stresses are then added to the thermal stresses as obtained by the method described.

Data:

$E_c = 3 \times 10^6$ psi $\qquad \nu_L = 0.25$

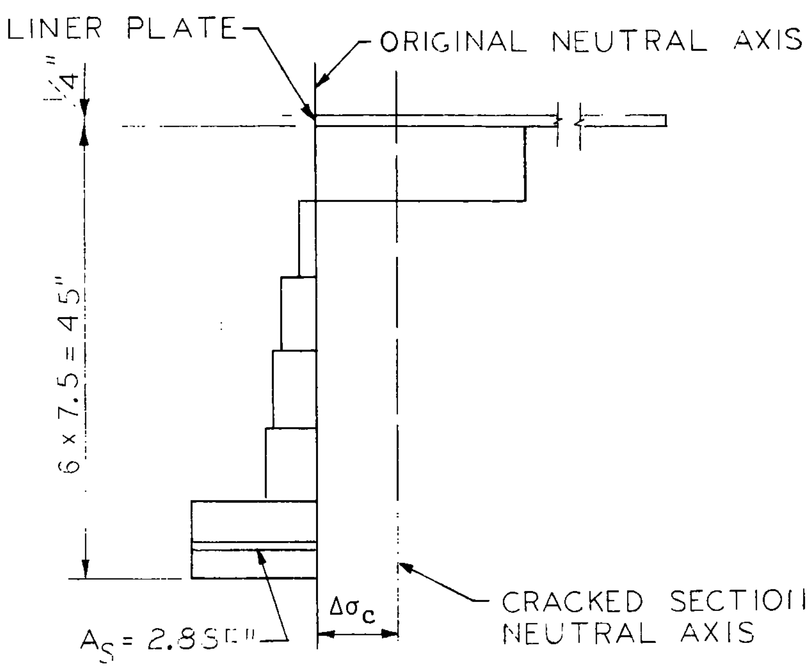
$E_s = 29 \times 10^6$ psi $\qquad n_R = 8.02$

$\nu_c = 0.17$ $\qquad n_L = 10.83$

Notation:

E_c	Modulus of elasticity of concrete.
E_s	Modulus of elasticity of steel.
n_L	Modular ratio of liner plate-concrete.
n_R	Modular ratio of reinforcement-concrete.
$\Delta\sigma_c$	Reduction of concrete compressive stress, considering cracking.
Σ_c	Concrete strain.
Σ_s	Steel strain.
Σ_x	Concrete strain in X direction.
Σ_y	Concrete strain in Y direction.
ν_c	Poisson's ratio of concrete.
ν_L	Poisson's ratio of liner plate.
σ_c	Stress in concrete.
σ_L	Stress in liner plate.
σ_R	Stress in reinforcement.
σ_x	Stress in concrete in direction X.

STRESS BLOCK FROM THE COMPUTER OUTPUT



(Thermal) Stresses (Psi)	(Thermal Forces (K/Ft.) Resultant
-36000	- 108.0
- 1997	- 179.7
+ 277	+ 24.9
+ 529	+ 47.6
+ 595	+ 53.5
+ 666	+ 60.0
+ 1467	+ 132.0

SECTION G-H
(ELEMENTS 333-339)

EQUILIBRIUM AFTER CRACKING

$$2.88\ (1467+\Delta\sigma_c)\ 8.02 - (179.7+108)\ 1000 + \Delta\sigma_c\ (12x7.5+3x10.83) = N = -\ 102{,}000$$

$$33884 + 23.1\ \Delta\sigma_c - 287700 + 124.5\ \Delta\sigma_c + 102{,}000 = 0$$

$$147.6\ \Delta\sigma_c = 151{,}816$$

$$\Delta\sigma_c = 1028.6$$

ASSUMED POSITION OF N. A. is O.K.

$$\Delta\sigma_c = 1029 \text{ Psi}$$

σ_s(After Cracking) $= (1467 + 1029)\ 8.02 = 20018$ Psi

σ_c $= -\ 1997 + 1029 = -\ 968$ Psi

σ_R $= \sigma_{D+F+P} + \sigma_T + \sigma_E$
$= -\ 503 + 20018 \pm 96 = 19611$ (Tensile)

σ_c $= \sigma_{D+F+P} + \sigma_T + \sigma_E$
$= 61 - 968 \pm 11 = -\ 918$ (Compression)

Rev. 5. 5/25/70
(New Page)

5. (Moved to page 5-14a)

5.1.3.2 Nonaxisymmetric Analysis

The nonaxisymmetric aspects of configuration or loading required various methods of analysis. The description of the methods used as applied to different parts of the containment is given below.

a. Buttresses

The buttresses and tendon anchorage zones are defined as Class I elements and were designed in accordance with the general design
6. criteria for the reactor building structure and with the applicable provisions of ACI 318-63, Chapter 26.

The buttresses were analyzed for two effects, nonaxisymmetric and anchorage zone stresses. Both effects are shown in the results of a two-dimensional plane strain finite element analysis with loads acting in the plane of the coordinate system (Figure 5-8).

At each buttress, the hoop tendons are alternately either continuous or spliced by being mutually anchored on the opposite faces of the buttress. Between the opposite anchorages, the compressive force exerted by the spliced tendon is twice as much as elsewhere. This value combined with the effect of the tendon which is not spliced will be 1.5 times the prestressing force acting outside of the buttresses. The cross-sectional area at the buttress is about 1.5 times that of the wall, so the hoop stresses as well as the hoop strains and radial displacements can be considered as being nearly constant all around the structure. Isostress plots of the plane strain analysis, Figure 5-9, confirm this.

The vertical stresses and strains, caused by the vertical post-tensioning, become constant at a short distance away from the anchorages because of the stiffness of the cylindrical shell. Since the stresses and strains remain nearly axisymmetric despite the presence of the buttresses, their effect on the overall analysis is negligible when the structure is under dead load or prestressing loads.

When an increasing internal pressure acts upon the structure, com-
6. bined with a thermal gradient (Figure 5-9A) such as at the design accident condition, the resultant forces being axisymmetric, the stiffness variation caused by the buttresses will decrease as the concrete develops cracks. The structure will then tend to shape itself to follow the direction of the acting axisymmetric resultant forces even more closely. Thus, the buttress effect is more axisymmetric at yield loads, which include factored pressure, than at design loads including pressure. This fact, combined with the design provision that alternate horizontal tendons terminate in a single buttress, indicates that the buttresses will not reduce the margins of safety available in the structure.

The analysis of the anchorage zone stresses at the buttresses has been determined to be the most critical of all the various types of anchorage areas of the shell. The local stress distribution in the immediate vicinity of the bearing plates has been derived by the

Rev. 5. 5/25/70
Rev. 6. 6/22/70

following three analysis procedures:

1. The Guyon equivalent prism method: This method is based on experimental photoelastic results as well as on equilibrium considerations of homogeneous and continuous media. It should be noted that the relative bearing plate dimensions are considered.

2. In order to include biaxial stress effects, use has been made of the experimental test results presented by S. J. Taylor at the March 1967 London Conference of the Institution of Civil Engineers (Group H, Paper 49). This paper compares test results with most of the currently used approaches (such as Guyon's equivalent prism method). It also investigates the effect of the rigid trumpet welded to the bearing plate.

3. The finite element method, assuming homogeneous and elastic material, was used in a plane strain analysis. The mesh and results are shown in Figures 5-8 and 5-9.

The Guyon method yields the following results for a loading ratio (a'/a)* = 0.9 Maximum compressive stress under the bearing plate:

σ_c = -2400 psi

Maximum tensile stress in spalling zone:

σ spalling = +2400 psi = - σ_c

Maximum tensile stress in bursting zones:

σ maximum bursting = 0.04 P = +96 psi

*Ratio of width of bearing plate to width of concrete under bearing plate.

S. J. Taylor's experimental results indicate that the anchor plate will give rise to a similar stress distribution pattern as Guyon's method; the main difference lies in the fact that the central bursting zone has a tensile stress peak of twice Guyon's value:

σ maximum bursting = +192 psi

By finite element analysis, the symmetric buttress loading yields a tensile peak stress in the bursting zone very close to S. J. Taylor's value:

σ maximum bursting = +114 psi

A state of biaxial tension in the concrete will appear on the outside face under the loading case 1.05D + 1.5P + 1.0T_A + 1.0F. The superposition of the corresponding state of stress with the local anchor stresses reduces the load carrying capacity of the anchorage unit and causes a reduction in the maximum tensile strain to cracking.

On the other hand, the uniform compressive state of stress (vertical prestress) applied to the anchorage zone increases the load carrying capacity of the anchorage unit, with the maximum tensile strain to cracking being increased.

The design of the buttress anchor zones considered such additional vertical stresses, leading to a state of pseudo biaxial stress, the second direction being radial through the thickness.

For the above-mentioned case, $1.05D + 1.5P + 1.0T_A + 1.0F$, the averaged vertical (meridional) stress component is:

$$f_a \simeq +400 \text{ psi}$$

The compressive bearing plate stress at 10 inches depth below the bearing plate is:

$$f_c \simeq -1500 \text{ psi}$$

(Note: The steel trumpet carries 7.2 percent of the prestress force.)

Thus, the two values introduced in the biaxial stress envelopes proposed in S. J. Taylor's article:

$$f_c/f'_c = 1500/5000 = 0.3$$

$$f_c/f'_c = 400/5000 = 0.08$$

show that failure could occur if vertical reinforcing were not provided. In fact, the maximum allowable vertical averaged tensile stress according to Taylor's interaction curve is $f_a/f'_c = 0.03$, therefore, $f_a = +150$ psi.

The three dimensional stress distribution in the anchor zones was analyzed in sufficient detail to permit the rational evaluation of stress concentrations. A conical wedge segment was used as the basic design element and the radial splitting tension was determined as a tangential distribution function. The summation of splitting stresses through the entire volume of the lead-in zone established the value of the splitting force. This force is a function of the a/b ratio and the cone angle and/or, a/b and h. Several different combinations of the values were analyzed and the most critical values selected. A system analysis for the vertical splitting force was carried out based on statics and the magnitude of vertical and
6. spalling forces were also determined.

The most unfavorable loads and load combinations were considered in the analysis of the anchorage zone and stresses based on transient thermal gradients were used in all cases where the use of a steady state gradient under-estimated the stresses and strains and were superimposed on the bursting stresses determined from the triaxial stress calculations. The computed stresses are less than the ACI allowable values. The design of the concrete reinforcement is based on this conservative analysis to provide a margin of safety similar to the other components of the reactor building structure and to control cracking in the anchorage zone. As a result, there is no danger of delayed rupture of the concrete under sustained load, due to local overstress and microcracking.

The reinforcing details, including the method for anchoring and splicing the reinforcing, are shown on Figure 5-9B.

The reinforcement required has been designed primarily to resist tensile forces, and has been located such that it will efficiently resist the tensile forces. The reinforcement was provided for load cases which create the maximum tensile forces and for other load cases the relevant shear forces or stresses were superimposed.

The possibility of the concrete breaking along shear plans was considered at the intersection of (1) the buttress with the cylinder and (2) the cylinder with the base slab.

1. <u>Buttress - Cylinder Intersection</u>

   6. An increase in the compression force at the buttress corresponds to an increase in the concrete area of the same magnitude.

2. <u>Cylinder - Base Slab Intersection</u>

   An analysis for the most critical radial shear conditions was performed. The difference in shear stiffness between the shell and the buttress and the remainder of the shell was included as a shear amplification factor. The reinforcing required was less than the reinforcing provided.

The possibility of concrete breaking along a shear plane is excluded by providing ample reinforcing. In other locations, breakage along the shear plane has been excluded by the opposition of prestressing and anchor forces.

The following three sources of information were also considered in the design of the anchorage zone reinforcing.

1. Full-scale load tests of the anchorage on the same concrete mix used in the structure and review of prior uses of the anchorage.

2. The post-tensioning supplier's recommendations of anchorage reinforcing requirements.

3. Review of the final details of the combined reinforcing by the consulting firm of T. Y. Lin, Kulka, Yang, and Associate.

b. Seismic or Wind Loading

6. Seismic loading of the structure controls in all cases over that of tornado or wind loading. The seismic analysis was conducted in the following manner. The loads on the Reactor Building caused by earthquake were determined by a dynamic analysis of the structure. The dynamic analysis was made on an idealized structure of lumped masses and weightless elastic columns acting as spring restraints. The analysis was performed in two stages: the determination of the natural frequencies of the structure and its mode shapes, and the response of these modes to the earthquake by the spectrum response method.

The natural frequencies and mode shapes were computed using the matrix equation of motion shown below for a lumped mass system. The form of the equation is: $(K)\ (\Delta) = \omega^2\ (M)\ (\Delta)$

K = matrix of stiffness coefficients including the combined effects of shear, flexure, rotation, and horizontal translation.

M = matrix of concentrated masses.

Δ = matrix of mode shape.

ω = angular frequency of vibration.

The results of this computation are the several values of ω_n and mode shapes Δ_n for n = 1, 2, 3, . . . m, where m is the number of degrees of freedom (i.e., lumped masses) assumed in an idealized structure.

The response of each mode of vibration to the design earthquake was then computed by the response spectrum technique, as follows:

1. The base shear contribution of the n^{th} mode $V_n = W_n S_{an}(\omega_n \gamma)$ where:

   W_n = effective weight of the structure in the n^{th} mode.

   $W_n = \dfrac{(\Sigma_x \Delta_{xn} w_x)^2}{\Sigma_x (\Delta_{xn})^2 w_x}$, where the subscript x refers to levels.

   throughout the height of the structure, and w_x is the weight of the lumped mass at level x.

   ω_n = angular frequency of the n^{th} mode.

   $S_{an}(\omega_n \gamma)$ = spectral acceleration of a single degree of freedom system with a damping coefficient of γ, obtained from the response spectrum.

2. The horizontal load distribution for the n^{th} mode was then computed as:

$$F_x = V_n \frac{(\Delta_{xn} w_x)}{\Sigma_x \Delta_{xn} w_x}$$

The several mode contributions were then combined to give the final response of the structure to the design earthquake.

3. The number of modes to be considered in the analysis was determined to adequately represent the structure being analyzed. The response of the modes of vibration was combined by taking the sum of the absolute modal values. The analytical model and results for 0.05g ground motion and 2 percent damping are shown in Figure 5-10.

4. Seismic and wind shears are transferred across construction
4. joints either by friction, by bond, by shear keys or by a combination of these.

c. <u>Large Opening (Equipment Hatch and Personnel Lock Opening)</u>

The primary loads considered in the design of the equipment hatch and personnel lock opening, as for any part of the structure, were dead load, prestress, pressure, earthquake, and thermal loads. The secondary loads considered were the following effects caused by the above primary loads:

1. The deflection of tendons around the opening.
2. The curvature of the shell at the opening.
3. The thickening around the opening.

The primary loads listed are mainly membrane loads with the exception of the thermal loads. In addition to membrane loads, accident pressure also produces punching shear around the edge of the opening. The values of these loads for design purposes were the magnitudes of these loads at the center of the opening. These are fairly simple to establish knowing the values of hoop and vertical prestressing, accident pressure, and the geometry and location of the opening.

Secondary loads were predicted by the following methods:

1. The membrane stress concentration factors and effect of the deflection of the tendons around the equipment hatch were analyzed for a flat plate by the finite element method. The stresses predicted by conventional stress concentration factors, compared with those values found from above-mentioned finite element computer program, demonstrated that the deflection of the tendons does not significantly affect the stress concentrations. This is a plane stress analysis and does not include the effect of the curvature of the shell. However, it gives an assurance of the correctness of the assumed membrane stress pattern caused by the prestressing around the opening. Results of this analysis are shown in Figure 5-11.

2. With the help of Reference 1, stress resultants around the large opening were found for various loading cases. Comparison of the results found from this reference, with the results of a flat plate of uniform thickness with a circular hole, showed the effect of the cylindrical curvature on stress concentrations around the opening.

Normal shear forces (relative to opening) were modified to account for the effect of twisting moments as shown in Reference 1. These modified shear forces are called Kirschoff's shear forces. Horizontal wall ties were provided to resist a portion of these shear forces.

3. The effect of the thickening on the outside face around the large opening was considered using several methods. Reference 2 was used to evaluate the effect of thickening on the stress concentration factors for membrane stress. A separate axisymmetric finite element computer analysis for a flat plate with anticipated thickening on the outside face was prepared to handle both axisymmetric and nonaxisymmetric loads to predict the effect of the concentration of hoop tendons, with respect to the Reactor Building at the top and bottom of the opening.

   For the analysis of the thermal stresses around the opening, the same method was used as for the other loadings. At the edge of the opening, a uniformly distributed moment, equal but opposite to the thermal moment existing on the rest of the shell, was applied and evaluated using the methods of the preceding Reference 1. The effects were then superimposed on the stresses calculated for the other loads and effects.

   In the case of accident temperature, after the accident presure has already been decreased, very little or no tension develops on the outside, so thermal strains will exist without the relieving effect of the cracks. However, the liner plate will reach a high strain level and so will the concrete at the inside corner of the penetration, thereby relieving the very high stresses, but still carrying a high moment in the state of redistribution stresses.

   In the case of 1.5P (prestress fully neutralized) + $1.0T_A$ (accident temperature), the cracked concrete with highly strained tension reinforcement constitutes a shell with stiffness decreased but still essentially constant in all directions. In order to control the increased hoop moment around the opening, the hoop reinforcement is about twice that of the radial reinforcement. See Figure 5-3.

   The equipment hatch opening was thickened for the following reasons:

   a. To reduce the larger than acceptable predicted membrane stresses around the opening.

   b. To accommodate tendon placement.

   c. To accommodate bonded steel reinforcing placement.

   d. To compensate for the reduction in the overall shell stiffness due to the opening.

The working stress method (elastic analysis) was applied to both the load combinations for design loads, as well as for yield loads, for the analytical procedures described above. The only difference is the higher allowable stresses under yield conditions. The various factored load combinations and capacity reduction factors are specified in Appendix 5A and were used for the yield load combinations using the working stress design method. The design assumption of straight line variation of stresses was maintained under yield conditions.

The governing design condition for the sides of the equipment hatch opening at the outside edge of the opening is the accident condition. Under this condition, approximately 60 percent of the total bonded reinforcing steel needed at the edge of the opening at the outside face is required for the thermal load.

Excluding thermal load, the remaining stress (equivalent to approximately 40 percent of the total load including thermal) at the edge of the outside face is the sum of the following stress resultants:

a. Normal stresses resulting from membrane forces, including the effect of thickening, contribute approximately minus 35 percent (minus 14 percent of total).

b. Flexural stresses resulting from the moments caused by thickening on the outside face contribute approximately 150 percent (60 percent of total).

c. Normal and flexural stresses resulting from membrane forces and moments caused by the effect of cylindrical curvature contribute approximately minus 15 percent (minus 6 percent of total).

d. <u>Penetrations</u>

Analysis of the Reactor Building penetrations falls into three parts: (1) the concrete shell, (2) the liner plate reinforcement and closure to the pipe, and (3) the thermal gradients and protection requirements at the high-temperature penetrations. The three categories will be discussed separately.

1. Concrete Shell

In general, special design consideration is given to all openings in the Reactor Building. Analysis of the various openings has indicated that the degree of attention required depends upon the penetration size. Small penetrations are considered to be those with a diameter smaller than 2-1/2 times the shell thickness: ie, approximately 8 feet in diameter or less. Reference 1 indicates that, for openings of 8-foot diameter or less, the curvature effect of the shell is negligible. In general, the typical concrete wall thickness has been found to be capable of taking the imposed stresses using bonded reinforcement, and the thickness is increased only as required to provide space requirements for radially deflected tendons. The induced stresses, due to normal thermal gradients and postulated rupture conditions, distribute rapidly and are of a minor nature compared to the numerous loading conditions for which the shell must be designed. The small penetrations are analyzed as holes in a plane sheet. Applied piping restraint loads due to thermal expansion or accident forces are assumed to distribute in the cylinder as stated in Reference 3. Typical details associated with these openings are indicated in Figure 5-2.

2. Liner Plate Closure

The stress concentrations around openings in the liner plate were calculated using the theory of elasticity. The stress concentrations were then reduced by the use of a thickened plate around the opening. In the case of a penetration with no appreciable external load, stud bolts are used to maintain strain compatibility between the liner plate and the concrete. Inward displacement of the liner plate at the penetration is also controlled by the stud bolts.

In the case of a pipe penetration in which significant external operating loads are imposed upon the penetration, the stress level from the external loads is limited to the design stress intensity values, S_m, given in the ASME Boiler and Pressure Vessel Code, Section III, Article 4. The stress level in the stud bolts from external loads is in accordance with the AISC Code.

The combining of stresses from all effects is performed using the methods outlined in the ASME Boiler and Pressure Vessel Code, Section III, Article 4, Figure N-414. The maximum stress intensity is the value from Figure N-415 (A) of the previously referenced code. Figure 5-12 shows a typical penetration and the applied loads.

Design stresses for the effects of pipe loads, pressure loads, dead load, and earthquake were calculated and the stress intensity kept below S_m.

The stresses from the remaining effects were combined with the above-calculated stresses and the stress intensity kept below S_a.

3. Thermal Gradient

The only high temperature lines penetrating the Reactor Building shell are the main steam and feedwater. Cooling fans and stacks designed to maintain the temperature in the penetration below 150 F are provided.

e. Liner Plate

There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure even though the liner will, at times, provide assistance in order to maintain deformation compatibility.

Loads are transmitted to the liner plate through the anchorage system and direct contact with the concrete and vice versa. Loads may be, at times, also transmitted by bond and/or friction with the concrete. These loads cause, or are caused by, liner strain. The liner is designed to withstand the predicted strains.

Possible cracking of concrete has been considered and reinforcing steel is provided to control the width and spacing of the cracks. In addition, the design is made such that total structural deformation remains small during the loading conditions, and that any cracking will be orders of magnitude less than that sustained in the repeated attempts to fail the prestressed concrete reactor vessel "Model 1," and even smaller than the concrete strains of overpressure tests of "Model 2" (both at General Atomic). See Reference 4 and Reference 5.

As described, the structural integrity consequences of concrete cracking are limited by the bonded reinforcing and unbonded tendons provided in accordance with the design criteria, 5.1.2. The effect os concrete cracking on the liner plate has also been considered. The anchor spacing and other design criteria are such that the liner will sustain orders of magnitude of strain, for example, less than did the liner of Model 1 at General Atomic (Reference 4) without tensile failure.

f. Liner Plate Anchors

The liner plate anchors were designed to preclude failure when subjected to the worst possible loading combinations. The anchors were also designed such that, in the event of a missing or failed anchor, the total integrity of the anchorage system would not be jeopardized by the failure of adjacent anchors.

The following loading conditions were considered in the design of the anchorage system:

1. Prestress
2. Internal Pressure
3. Shrinkage and Creep of Concrete
4. Thermal Gradients
5. Dead Load

6. Earthquake

7. Wind or Tornado

8. Vacuum

The following factors were considered in the design of the anchorage system:

1. Initial inward curvature of the liner plate between anchors due to fabrication and erection inaccuracies.

2. Variation of anchor spacing.

3. Misalignment of liner plate seams.

4. Variation of plate thickness.

5. Variation of liner plate material yield stress.

6. Variation of Poisson's ratio for liner plate material.

7. Cracking of concrete in anchor zone.

8. Variation of the anchor stiffness.

The anchorage system satisfies the following conditions:

1. The anchor has sufficient strength and ductility so that its energy absorbing capability is sufficient to restrain the maximum force and displacement resulting from the condition where a panel with initial outward curvature is adjacent to a panel with initial inward curvature.

2. The anchor has sufficient flexural strength to resist the bending moment which would result from Condition 1.

3. The anchor has sufficient strength to resist radial pull-out force.

When the liner plate moves inward radially as shown in Figure 5-13, the sections will develop membrane stress due to the fact that the anchors have moved closer together. Due to initial inward curvature, the section between 1 and 4 will deflect inward giving a longer length than adjacent sections and some relaxation of membrane stress will occur. It should be noted here that section 1-4 cannot reach an unstable condition due to the manner in which it is loaded.

The first part of the solution for the liner plate and anchorage system is to calculate the amount of relaxation that occurs in section 1-4, since this value is also the force across anchor 1 if it is infinitely stiff. This solution was obtained by solving the general differential equation for beams and the use of calculus to simulate relaxation or the lengthening of section 1-4. Figure 5-13 shows the symbols for the forces that result from the first step in the solution.

Using the model shown in Figure 5-14 and evaluating the necessary spring constants, the anchor was allowed to displace.

The solution yielded a force and displacement at anchor 1, but the force in section 1-2 was (N) - $K_{R(Plate)}S_1$ and anchor 2 was no longer in force equilibrium.

The model shown in Figure 5-14 was used to allow anchor 2 to displace and then to evaluate the effects on anchor 1.

The displacement of anchor 1 was $S_1 + S'_1$ and the force on anchor 1 was $K_c(S_1 + S'_1)$. Then anchor 3 is not in force equilibrium and the solution continued to the next anchor.

After the solution was found for displacing anchor 2 and anchor 3, the pattern was established with respect to the effect on anchor 1 and by inspection, the solution considering an infinite amount of anchors was obtained in the form of a series solution.

The preceding solution yielded all necessary results. The most important results were the displacement and force on anchor 1.

Various patterns of welds attaching the angle anchors to the liner plate have been tested for ductility and strength when subjected to a transverse shear load such as N and are shown in Figure 5-15.

Using the results from these tests together with data from tests made for the Fort St. Vrain PSAR, Amendment No. 2 and Oldbury vessels, Reference 6, a range of possible spring constants was evaluated for the Oconee liner. By using the solution previously obtained together with a chosen spring constant, the amount of energy required to be absorbed by the anchor was evaluated.

By dividing the amount of energy that the system will absorb by the most probable maximum energy, the result then yielded the factor of safety.

By considering the worst possible loading condition which resulted from the listed loading conditions and the conditions stated below, the results in Table 5-1 were obtained.

Case I - Simulates a plate with a yield stress of 36 ksi and no variation in any other parameters.

Case II - Simulates a 1.25 increase in yield stress and no variation in any other parameters.

Case III - Simulates a 1.25 increase in yield stress, a 1.16 increase in plate thickness and a 1.08 increase for all other parameters.

Case IV - Simulates a 1.88 increase in yield stress with no variation of any other parameters.

Case V - Is the same as Case III except the anchor spacing has been doubled to simulate what happens if an anchor is missing or has failed.

TABLE 5-1

Case	Nominal Plate Thickness (In.)	Initial Inward Displacement (In.)	Anchor Spacing L_1 (In.)	Anchor Spacing L_2 (In.)	Factor of Safety Against Failure
I	0.25	0.125	15	15	37.0
II	0.25	0.125	15	15	19.4
III	0.25	0.125	15	15	9.9
IV	0.25	0.125	15	15	6.28
V	0.25	0.25	30	15	4.25

g. Supports

In designing for structural bracket loads applied perpendicular to the plane of the liner plate, or loads transferred through the thickness of the liner plate, the following criteria and methods have been used:

1. The liner plate was thickened to reduce the predicted stress level in the plane of the liner plate. The thickened plate with the corresponding thicker weld attaching the bracket to the plate will also reduce the probability of the occurrence of a leak at this location.

2. Under the application of a real tensile load applied perpendicular to the plane of the liner plate, no yielding is to occur in the perpendicular direction. By limiting the predicted strain to 90 percent of the minimum guaranteed yield value, this criterion was satisfied.

3. The allowable stress in the perpendicular direction was calculated using the allowable predicted strain in the perpendicular direction together with the predicted stresses in the plane of the liner plate.

4. In setting the above criteria, the reduced strength and strain ability of the material perpendicular to the direction of rolling (in plane of plate) was also considered if the bracket did not penetrate the liner thickened plate. In this case, the major stress is normal to the plane of the liner plate. The allowable stresses were reduced to 75 percent of the stress permitted in Item (3) above.

5. The necessary plate characteristics were assured by ultrasonic examination of the thickened plates for lamination defects.

h. Missiles

The turbine-generator supplier has made a study of failure of rotating elements of steam turbines and generators. The postulated types of failures are: (1) failure of rotating components operating at or near normal operating speed and, (2) failure of components that control admission of steam to the turbine resulting in destructive shaft rotational speed.

1. Failure at or Near Operating Speed

All of the known turbine and generator rotor failures at near rated speed resulted from the combination of severe strain concentrations in relatively brittle materials. New alloys and processes have been developed and adopted to minimize the probability of brittle fracture in rotors, wheels, and shafts. Careful control of chemistry and detailed heat treating cycles have greatly improved the mechanical properties of all of these components. Transition temperatures (the temperature at which the character of the fracture in the steel changes from brittle to ductile, often identified as FATT) have been reduced on the low temperature wheel and rotor applications for nuclear units to well below startup temperatures. Improved steel mill practices in vacuum pouring and alloy addition have resulted in forgings which are much more uniform and defect free than ever before. More comprehensive vendor and manufacturer tests involving improved ultrasonic and magnetic particle testing techniques are better able to discover surface and internal defects than in the past. Laboratory investigation has revealed some of the basic relationships between structure strength, material strength, FATT and defect size, and location so that the reliability of the rotor as a structure has been significantly improved over the past few years.

New starting and loading instructions have been developed to reduce the severity of surface and bore thermal stress cycles incurred during service. The new practices include:

a) Better temperature sensors.

b) Better control devices for acceleration and loading.

c) Better guidance for station operators in the control speed, acceleration, and loading rates to minimize rotor stresses.

Progress in design, better materials and quality control, more rigorous acceptance criteria, and improved machine operation <u>have substantially reduced the likelihood of burst failures of turbine-generator rotors operating at or near rated speed</u>.

2. Failure at Destructive Shaft Rotational Speeds

Improvements of rotor quality discussed above, while reducing the chance of failures at operating speed, tend to increase the hazard level associated with unlimited overspeed because of higher bursting speed. Therefore, turbine overspeed protection systems have been evaluated as follows:

a) Main and secondary steam inlets have the following valves in series:

1) Control valves - controlled by the speed governor and tripped closed by emergency governor and backup overspeed trip, thus providing three levels of control redundancy.

2) Stop valves or trip throttle valve - actuated by the emergency governor and backup overspeed trip, thus providing two levels of control redundancy.

Since 1948 there have been over 650 turbines, of over 10,000 kw each, placed in service by the Oconee turbine supplier with no report of main stop valves failing to close when required to protect the turbine. Impending sticking has been disclosed by means of the fully closed test feature so that a planned shutdown could be made to make the necessary correction. This almost always involves the removal of the oxide layer which builds up on the stem and bushing and which would not occur on a low temperature nuclear application.

3) Combined stop and intercept valves in cross around systems - these are actuated by the speed governor, emergency, and backup overspeed trips. These valves also include the testing features described above.

The speed sensing devices for the governor and emergency governor are separate from each other, thus providing two independent lines of defense.

b) Uncontrolled Extraction Lines to Feedwater Heaters

If the energy stored in an uncontrolled extraction line is sufficient to cause a dangerous overspeed, two positive closing nonreturn valves are provided, to be actuated by the emergency governor and backup overspeed trip. These are designed for remote manual periodic tests to assure proper operation. The station piping, heater, and check valve system are reviewed during the design stages to make sure the entrained steam cannot overspeed the unit beyond safe limits.

Special field tests are made of new components to obtain design information and to confirm proper operation. These include the capability of controls to prevent excessive overspeed on loss of load.

Careful analysis of all past failures has led to design, inspection, and testing procedures to substantially eliminate destructive overspeed as a possible cause of failure in modern design units.

The study of postulated ruptures made by the turbine-generator supplier concludes that the missile having the highest combination of weight, size, and energy is the last stage wheel. The properties of this missile are summarized in Table 5-2. Initial velocities and energies shown below are based on 180 percent of the initial energy being absorbed in penetrating the casing.

TABLE 5-2

Weight	Impact Area
5944 lbs	Side On - 8.368 sq ft
	End On - 3.657 sq ft
Velocity	Kinetic Energy Ft-Lbs
Initial - 710 fps	Initial - 46.5×10^6
Impact	Impact
Cylinder - 502 fps	Cylinder - 23.25×10^6
Dome - 431 fps	Dome - 18.0×10^6

Analysis of the above missile is based on calculations using methods presented in Reference 7 to determine the depth to which this missile would penetrate the concrete Reactor Building. Conservatively, no reduction of missile energy was made for penetration of the Turbine Building and/or impact with intervening equipment and structural components after leaving turbine shell. The energy loss from 23.25×10^6 ft-lbs to 18.0×10^6 is caused by air friction. This effect has been calculated by using a drag coefficient of 1.0. Since the offset between the Turbine and Reactor Buildings is relatively short, about 170 feet, no account has been taken for air friction losses for the case in in which the missile is ejected nearly horizontally to strike the cylinder wall. Following are results of analysis:

Case I:

"Side on" impact. Missile could penetrate the concrete cylinder wall to a depth of approximately 6 inches and the dome to a depth of approximately 5-1/2 inches. The tendons will not be damaged since they are protected to a depth of 7-3/4 inches in the cylinder wall and 8 inches in the dome.

Case II:

"End on" impact. In this case the missile could penetrate the concrete cylinder wall to a depth of approximately 13-3/4 inches and the dome to a depth of approximately 12-1/4 inches. The tendon arrangement is such that the missile could strike two adjacent tendons in the dome or a maximum of three horizontal and one vertical tendons in the cylinder wall. The local effect on the tendons could be one of either partial deflection or possible severance. However, analysis of the structure indicates that the structure can withstand the loss of three horizontal and three vertical tendons in the cylinder wall or five adjacent tendons in the dome without loss of function and a greater number of tendons without building failure.

Case III:

As a final analysis, an extreme case was considered in which none of the initial kinetic energy of the missile is absorbed by its penetration through the turbine casing. The total initial energy of 46.5×10^6 ft-lbs is available for penetration of the cylinder wall and 29.3×10^6 ft-lbs for penetration of dome where the reduction is due to air friction only. The maximum depth of penetration of cylinder wall is 35-1/2 inches and the dome is 25 inches. The missile can strike five tendons in the dome or three horizontal and one vertical tendons in the cylinder wall. The local effect in the impact area would be as described in Case II above even though the depth of penetration is greater.

Depths of penetration of Reactor Building wall are summarized in Table 5-3.

TABLE 5-3

Depth of Penetration of Concrete

Case I		Case II		Case III	
Cylinder	Dome	Cylinder	Dome	Cylinder	Dome
6"	5-1/2"	12-3/4"	12-1/4"	35-1/2"	25"

Since the thicknesses of the cylinder wall and dome are 45 inches and 39 inches respectively, it can be seen that the turbine missile, even under extreme assumptions, does not penetrate the Reactor Building.

For an analysis of missiles created by a tornado having maximum wind speeds of 300 mph, two missiles were considered. One is a missile equivalent to a 12 foot long piece of wood 8 inches in diameter traveling end on at a speed of 250 mph. The second is a 2000 pound automobile with a minimum impact area of 20 square feet traveling at a speed of 100 mph.

For the wood missile, calculations based on energy principle indicate that because the impact pressure exceeds the ultimate compressive strength of wood by a factor of about four, the wood would crush due to impact. However, this could cause a secondary source of missiles if the impact force is sufficiently large to cause spalling of the free (inside) face. The compressive shock wave which propagates inward from the impact area generates a tensile pulse, if it is large enough, will cause spalling of concrete as it moves back from the free (inside) surface. This spalled piece moves off with some velocity due to energy trapped in the material. Successive pieces will spall until a plane is reached where the tensile pulse becomes smaller than the tensile strength of concrete. From the effects of impact of the 8 inch diameter by 12 foot long wood missile, this plane in a conventionally reinforced concrete section would be located approximately 3 inches from the free (inside) surface. However, since the Reactor Building is prestressed, there will be residual compression in the free face, as the tensile pulse moves out and spalling will not occur. Calculations indicate that in the impact area a 2 inch or 3 inch deep crushing of concrete should be expected due to excessive bearing stress due to impact.

For the automobile missile, using the same methods as in the turbine failure analysis, the calculated depth of penetration is 1/4 inch and for all practical purposes the effect of impact on the Reactor Building is negligible.

From the above, it can be seen that the tornado generated missiles neither penetrate the Reactor Building wall nor endanger the structural integrity of the Reactor Building or any components of the reactor coolant system.

5.1.4 IMPLEMENTATION OF CRITERIA

This section documents the manner in which the design criteria were met by the designer.

Section 5.1.4.1 consists of isostress plots and tabulations of predicted stresses for the various materials. The isostress plots of the homogeneous uncracked concrete structure indicate the general stress pattern for the structure as a whole, under various loading conditions. More specific documentation is made of the predicted stresses for all materials in the structure. In these tabulations, the predicted stress is compared with the allowable to permit an easy comparison and evaluation of the adequacy of the design.

Sections 5.1.4.3 and 5.1.4.4 illustrate the actual details used in the design to implement the criteria.

5.1.4.1 Results of Analysis

The isostress plots, Figures 5-6 and 5-7, show the three principal stresses and the direction of the principal stresses normal to the hoop direction. The principal stresses are the most significant information about the behavior of the structure under the various conditions and were a valuable aid for the final design.

The plots were prepared by a cathode-ray tube plotter. The data for plotting were taken from the stress output of the finite element computer program of the following design load cases:

$$D + F$$

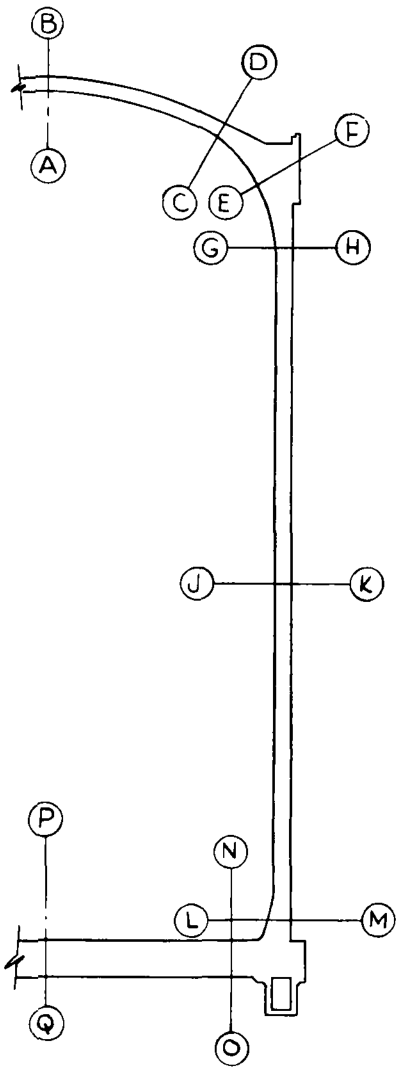
$$D + F + 1.15P$$

$$D + F + 1.5P + T_A$$

$$D + F + T_A$$

The above axisymmetric loading conditions have been found to be governing in the design since they result in highest stresses at various locations in the structure.

5. The containment stress analysis results for structural concrete and liner plate, including shear stresses, are shown in Table 5-3A.



KEY ELEVATION

(SHOWING LOCATION OF REFERENCE SECTIONS)

Table 5 - 3A
STRESS ANALYSIS RESULTS
Sheet 1 of 6

STRUCTURAL DATA

LOCATION	CONCRETE		REINFORCING STEEL		
	f'_c-psi	t-in	TYPE	p_m -%	p_h -%
A	5000	39	A615GR40	0.05	0.05
B	5000	39	A615GR40	0.23	0.23
C	5000	55	A615GR60	0.16	0.16
D	5000	55	A615GR60	0.30	0.30
E	5000	138	A615GR60	0.06	0.06
F	5000	138	A615GR60	0.18	0.18
G	5000	45	A615GR60	---	---
H	5000	45	A615GR60	0.53	0.53
J	5000	45	A615GR40	---	---
K	5000	45	A615GR40	0.25	0.25
L	5000	63	A615GR60	0.83	0.51
M	5000	63	A615GR60	0.74	0.72
N	5000	102	A615GR60	0.49	0.13
O	5000	102	A615GR60	0.87	0.31
P	5000	102	A615GR60	0.19	0.19
Q	5000	102	A615GR60	0.34	0.34

NOTES

1. LOADING CASES I, II, & III ARE WORKING STRESS ANALYSIS WHEREAS LOADING CASES IV, V, VI ARE YIELD STRESS ANALYSIS.
2. FOR NOTATION AND ALLOWABLE STRESSES SEE SHEET 2.
3. ALL CONCRETE EXTREME FIBER STRESS (σe) ARE SHOWN FOR THE INSIDE SURFACE. OUTSIDE SURFACE STRESSES ARE INDICATED BY ( ). THE STRESSES LISTED ARE THE CONTROLLING STRESSES FOR THAT SECTION.
4. COMPUTED VS. ALLOWABLE RATIOS FOR CASES IV, V, AND VI INCLUDE APPROPRIATE ϕ FACTORS, e.g. $\frac{\sigma a}{\phi f a}$
5. ALLOWABLE SHEAR STRESSES INCLUDE STIRRUPS WHEREVER APPLICABLE.
6. THE STRESSES SHOWN FOR THE LOAD CASES INCLUDING T_A ARE BASED ON CRACKED SECTION ANALYSIS UNLESS NOTED BY *.
7. DEVIATIONS IN ALLOWABLE STRESSES ARE IN ACCORDANCE WITH PSAR APPENDIX 5-C.

Rev. 5 5/25/70
New Page

Table 5 - 3A
STRESS ANALYSIS RESULTS
Sheet 2 of 6

NOTATION

D	DEAD LOAD
F	PRESTRESS
P	INTERNAL PRESSURE
E	EARTHQUAKE (DESIGN)
E'	EARTHQUAKE (HYPOTHETICAL)
T_A	ACCIDENT TEMPERATURE
fc	ULTIMATE CONCRETE STRESS
fy	STEEL RE-BAR YIELD STRESS
fa	ALLOWABLE CONCRETE AXIAL STRESS
fce	ALLOWABLE CONCRETE AXIAL & FLEXURE STRESS
v	ALLOWABLE CONCRETE SHEAR STRESS INCLUDING STIRRUPS IF APPLICABLE
fs	ALLOWABLE STEEL STRESS
σa	NOMINAL MEMBRANE STRESS
σe	COMBINED AXIAL & FLEXURE NOMINAL STRESS
τ	ACTUAL SHEAR STRESS
h	SUBSCRIPT INDICATING HOOP DIRECTION
m	SUBSCRIPT INDICATING MERIDIONAL DIRECTION
p_h	HOOP STEEL PERCENTAGE
p_m	MERIDIONAL STEEL PERCENTAGE
+	TENSILE STRESSES
–	COMPRESSIVE STRESSES
*	UNCRACKED SECTION ANALYSIS

ALLOWABLE STRESSES

	WORKING STRESS DESIGN	YIELD STRESS DESIGN
SHELL CONCRETE:	fa = 1500 psi	fa = ϕa (fc) = (0.85) (5000) = 4,250 psi
	fce = 3000 psi	fce = ϕce (fc) = (0.90) (5000) = 4,500 psi
BASE CONCRETE:	f_{ce} = 2250 psi	fa = ϕa (fc) = (0.85) (5000) = 4,250 psi
		fce = ϕce (fc) = (0.90) (5000) = 4,500 psi
STEEL: A615GR40	fs = 20,000 psi	fs = ϕ (fy) = (0.90) (40,000) = 36,000 psi
A615GR60	fs = 30,000 psi	fs = ϕ (fy) = (0.90) (60,000) = 54,000 psi

Table 5 - 3A
STRESS ANALYSIS RESULTS
Sheet 3 of 6

"D + F INITIAL" (Stresses in psi) Case I*

SECTION		MERIDIONAL σ OUTSIDE	MERIDIONAL σ INSIDE	MERIDIONAL σ AXIAL	HOOP σ OUTSIDE	HOOP σ INSIDE	HOOP σ AXIAL	SHEAR τ	SHEAR V_{ci}	SHEAR V_{cw}
SHELL	A - B	-1,340	-1,140	-1,250	-1,283	-1,096	-1,178	-14	111	605
	C - D	-218	-1,280	-668	-312	-460	-362	74	100+33	473+33
	E - F	-471	-581	-441	-353	-428	-386	47	185+145	411+145
	G - H	-667	-515	-584	-860	-872	-864	34	666	451
	J - K	-729	-673	-708	-1,205	-1,272	-1,251	-4	—	484
	L - M	-319	-881	-566	-211	-349	-273	-7	101+81	446+81
BASE	N - O	141	-84	23	10	-60	-36	-16	105+257	235+257
	P - Q	-27	-24	-26	-26	-20	-26	7	—	260

ALLOWABLE CONCRETE STRESSES:

Shell: fa = 1500 psi
fce = 3000 psi

Table 5 - 3A
STRESS ANALYSIS RESULTS
Sheet 4 of 6

REACTOR BUILDING — SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES

SECTION	LOAD CASE	CONCRETE								REINFORCING STEEL			
		COMPUTED					COMPUTED VS. ALLOWABLE			COMPUTED		COMPUTED VS. ALLOWABLE	
		σem	σeh	σam	σah	τ	$\frac{\sigma e}{fce}$	$\frac{\sigma a}{fa}$	$\frac{\tau}{v}$	σm	σh	$\frac{\sigma m}{fs}$	$\frac{\sigma h}{fs}$
A-B	II - D+F+1.15P	(-313)*	(-303)*	-282*	-271*	-5*	0.104*	0.188*	0.036*	—	—	—	—
	III - D+F+P+T_A+E	-1,632	-1,422	-412	-368	-22	.544	.275	0.089	11,608	12,408	.580	.620
	IV - 0.95D+F+1.5P+T_A	-328	-223	+16	+58	0	0.073	0	0.000	26,723	22,326	.742	.620
	V - 0.95D+F+1.25P+1.25E+T_A	-1029	-823	-198	-154	-23	0.229	.047	0.920	17,462	17,513	.485	.486
	VI - 0.95D+F+P+E'+T_A	-1,632	-1422	-412	-368	-26	0.363	.097	0.105	11,608	12,408	.322	.345
C-D	II - D+F+1.15P	-310*	-329*	-259*	-278*	83*	0.110*	0.185*	0.216*	—	—	—	—
	III - D+F+P+T_A+E	-2040	-1638	-312	-296	103	0.680	.208	0.896	26,361	21,360	.879	.712
	IV - 0.95D+F+1.5P+T_A	-502	-1493	-131	-240	79	0.329	.056	0.642	15,587	26,135	.288	.484
	V - 0.95D+F+1.25P+1.25E+T_A	-920	-1482	-216	-268	109	0.329	.063	0.380	15,283	21,552	.283	.400
	VI - 0.95D+F+P+E'+T_A	-2232	-1683	-312	-296	109	0.496	.073	0.752	28,762	21,360	.533	.396
E-F	II - D+F+1.15P	(-551)*	(-374)*	-328*	-363*	49*	0.183*	0.242*	0.245*	—	—	—	—
	III - D+F+P+T_A+E	-298	-1520	-274	-358	83	0.507	0.239	0.170	—	12,800	—	.427
	IV - 0.95D+F+1.5P+T_A	-(305)	-1450	-233	-343	52	0.322	0.081	0.107	2,532	12,213	0.047	.226
	V - 0.95D+F+1.25P+1.25E+T_A	-170	-2367	-230	-350	91	0.526	0.082	0.153	—	36,328	—	.673
	VI - 0.95D+F+P+E'+T_A	-295	-1520	-271	-358	91	0.338	0.084	0.151	—	12,800	—	.237

Table 5 - 3A
STRESS ANALYSIS RESULTS
Sheet 5 of 6

REACTOR BUILDING — SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES

SECTION	LOAD CASE	CONCRETE								REINFORCING STEEL			
		COMPUTED					COMPUTED VS. ALLOWABLE			COMPUTED		COMPUTED VS. ALLOWABLE	
		σ_{em}	σ_{eh}	σ_{am}	σ_{ah}	τ	$\frac{\sigma_e}{f_{ce}}$	$\frac{\sigma_a}{f_a}$	$\frac{\tau}{v}$	σ_m	σ_h	$\frac{\sigma_m}{f_s}$	$\frac{\sigma_h}{f_s}$
G-H	II - D+F+1.15P	(-147)*	(-299)*	-111*	-298*	-10*	0.100*	0.199*	0.089*	—	—	—	—
	III - D+F+P+T_A+E	-744	-1594	-143	-386	42	0.531	.257	0.500	21,517	12,890	.717	.430
	IV - 0.95D+F+1.5P+T_A	-50	-616	+50	-124	-17	0.137	.029	0.233	27,548	20,924	.5101	.388
	V - 0.95D+F+1.25P+1.25E+T_A	-319	-1108	-25	-255	-57	0.246	.060	0.626	25,157	16,924	.466	.313
	VI - 0.95D+F+P+E'+T_A	-724	-1594	-137	-386	52	0.354	.091	0.584	21,696	12,890	.402	.239
J-K	II - D+F+1.15P	(-255)*	-250*	-228*	-254*	-4*	0.085*	0.169*	0.082*	—	—	—	—
	III - D+F+P+T_A+E	-944	-2606	-229	-474	-61	0.869	.316	0.735	16,118	9,344	.806	.467
	IV - 0.95D+F+1.5P+T_A	-21	-719	+111	+46	-11	0.160	0	0.647	29,315	19,273	.814	.535
	V - 0.95D+F+1.25P+1.25E+T_A	-478	-1319	-98	-186	-73	0.293	.044	0.948	19,986	15,116	.555	.420
	VI - 0.95D+F+P+E'+T_A	-918	-2606	-222	-474	-73	0.579	.111	0.737	16,334	9,344	.454	.260
L-M	II - D+F+1.15P	(-567)*	(-289)*	-335*	-267*	78*	0.189*	0.223*	0.294*	1,900*	—	0.063*	—
	III - D+F+P+T_A+E	-727	-595	-237	-166	159	0.242	.158	0.503	7760	12,325	.259	.411
	IV - 0.95D+F+1.5P+T_A	—	-850	-154	-249	140	0.190	.059	0.356	(16,440)	8,997	.304	.167
	V - 0.95D+F+1.25P+1.25E+T_A	-449	-340	-141	-144	198	0.100	.034	0.504	10,068	12,367	.186	.229
	VI - 0.95D+F+P+E'+T_A	-630	-528	-203	-145	173	0.140	.048	0.456	8,595	12,881	.159	.238

Table 5 - 3A
STRESS ANALYSIS RESULTS
Sheet 6 of 6

REACTOR BUILDING — SUMMARY OF CONCRETE
AND REINFORCING STEEL STRESSES

SECTION	LOAD CASE	CONCRETE								REINFORCING STEEL			
		COMPUTED					COMPUTED VS. ALLOWABLE			COMPUTED		COMPUTED VS. ALLOWABLE	
		σ_{em}	σ_{eh}	σ_{am}	σ_{ah}	τ	$\frac{\sigma_e}{f_{ce}}$	$\frac{\sigma_a}{f_a}$	$\frac{\tau}{v}$	σ_m	σ_h	$\frac{\sigma_m}{f_s}$	$\frac{\sigma_h}{f_s}$
N-O	II - D+F+1.15P	-215*	-183*	223*	-35*	-182*	0.095*	$f_a = .3f'_c$ NOT APPLICABLE TO SLAB	0.361*	7,755*	7,236*	0.259*	0.241*
	III - D+F+P+T_A+E	-24	—	+151	+73	284	0.011		0.563	25,130	27,666	.838	.922
	IV - 0.95D+F+1.5P+T_A	-50	-627	+188	+15	323	0.139		0.455	27,386	24,980	.507	.463
	V - 0.95D+F+1.25P+1.25E+T_A	—	—	+188	+98	330	—		0.482	30,034	36,326	.556	.673
	VI - 0.95D+F+P+E'+ T_A	-28	—	+157	+90	293	—		0.442	26,379	32,501	.490	.602
P-Q	II - D+F+1.15P	-13*	(-14)*	-14*	-15*	9*	0.006*	$f_a = .3f'_c$ NOT APPLICABLE TO SLAB	0.035*	—	—	—	—
	III - D+F+P+T_A+E	-1000	-960	-16	-15	29	0.444		0.492	31,000	29,475	1.03	.9825
	IV - 0.95D+F+1.5P+T_A	-811	-501	-11	-10	6	0.360		0.130	32,600	34,097	.604	.631
	V - 0.95D+F+1.25P+1.25E+ T_A	-944	-581	-13	-13	36	0.420		0.563	36,502	36,162	.676	.670
	VI - 0.95D+F+P+E'+ T_A	-1026	-632	-16	-15	35	0.456		0.556	38,844	37,315	.719	.691

5.1.4.2 Prestress Losses

In accordance with the ACI Code 318-63, the design provides for prestress losses caused by the following effects:

a. Seating of anchorage.

b. Elastic shortening of concrete.

c. Creep of concrete.

d. Shrinkage of concrete.

e. Relaxation of prestressing steel stress.

f. Frictional loss due to intended or unintended curvature in the tendons.

All of the above losses can be predicted with sufficient accuracy.

The environment of the prestress system and concrete is not appreciably different in this case, from that found in numerous bridge and building applications. Considerable research has been done to evaluate the above items and is available to designers in assigning the allowances. Building code authorities consider it acceptable practice to develop permanent designs based on these allowances.

The following categories and values of prestress losses have been considered in the design:

Type of Loss	Assumed Value
Seating of Anchorage	None
Elastic Shortening of Concrete	$\frac{f_{cpi}}{3.0 \times 10^6}$ Inch/Inch
1. Creep of Concrete	0.280×10^{-6} Inch/Inch/psi
Shrinkage of Concrete	100×10^{-6} Inch/Inch
Relaxation of Prestressing Steel	8% of $0.65f_s$ = 12.5 Ksi
Frictional Loss	$K = 0.0003$, $\mu = 0.156$

There is no allowance for the seating of the BBRV anchor since no slippage occurs in the anchor during transfer of the tendon load into the structure. Sample lift-off readings will be taken to confirm that any seating loss is negligible.

The loss of tendon stress due to elastic shortening was based on the change in the initial tendon relative to the last tendon stressed.

The concrete properties study conducted at Clemson University indicated an actual
1. creep value of 0.280×10^{-6} inch/inch/psi. Conversion of the unit creep data to
hoop, vertical and dome stress gives these values of stress loss in the tendons:

Hoop	-16.1 Ksi
Vertical	- 8.05 Ksi
Dome	-16.1 Ksi

1. A single creep loss figure of 420×10^{-6} inch/inch at 1500 psi (f_{cpi}) was used throughout the structure. This results in a prestress loss of 12.6 ksi.

Rev 1. 9/15/69

The value used for shrinkage loss represents only that shrinkage that could occur after stressing. Since the concrete is, in general, well aged at the time of stress, little shrinkage is left to occur and add to prestress loss.

The value of relaxation loss is based on the information furnished by the tendon system vendor, The Prescon Corporation.

Frictional loss parameters for unintentional curvature (K) and intentional curvature (μ) are based on full-scale friction test data. This data indicates actual values of K = 0.0003 and μ = 0.125 versus the design values of K = 0.0003 and μ = 0.156.

Assuming that the jacking stress for the tendons is 0.80 f_s' or 192,000 psi and using the above prestress loss parameters, the following tabulation shows the magnitude of the design losses and the final effective prestress at end of 40 years for a typical dome, hoop and vertical tendon.

	Dome (Ksi)	Hoop (Ksi)	Vertical (Ksi)
Jacking Stress	192	192	192
Friction Loss	19	21.3 (1)	21
Seating Loss	0	0	0
1.			
Elastic Loss	14.5	14.3	7.2
Creep Loss	12.6	12.6	12.6
Shrinkage Loss	2.9	2.9	2.9
Relaxation Loss	12.5	12.5	12.5
Final Effective Stress (2)	130.5	128.4	135.8

(1) Average of crossing tendons.

(2) This force does not include the effect of pressurization which increases the prestress force.

To provide assurance, of achievement of the desired level of Final Effective Prestress and that ACI 318-63 requirements are met, a written procedure was prepared for guidance of post-tensioning work. The procedures provided nominal
4. values for end anchor forces in terms of pressure gage readings for calibrated jack-gage combinations. Force measurements were made at the end anchor, of course, since that is the only practical location for such measurements.

The procedure required the measured temporary jacking force, for a single tendon, to approach but not exceed 850 kips. ($0.8f_s'$). Thus the limits set by AC 318-63 2606 (a) 1, and of the prestressing system supplier, were observed. Additionally, benefits were obtained by in place testing of the tendon to provide final assurance that the force capability exceeded that required by design. During the increase in force, measurements were required of elongation changes and force changes in order to allow documentation of compliance with ACI 318-63 2621 (a). The procedures required that the prestressing steel be installed in the sheath before stressing for a sufficient time period that the temperatures of the prestressing steel and concrete reach essential equilibrium, to establish conformance with ACI 318-63 2621 (e). The jacking force of $0.8f_s'$ further provided for a means of equalizing the force in individual wires of a tendon to establish compliance with ACI 318-63 2621 (b). The procedures required compliance with ACI 318-63 such that, if broken wires resulted from the post-tensioning sequence, compliance with section 2621 (d) was documented. Each of the above procedures contributed to assurance that the desired level of Final Effective Prestress would be achieved.

The requirements of ACI 318-63 2606 (a) 2 state that f_s should not exceed $0.7f_s'$ for "post-tensioning tendons immediately after anchoring".

Industry has been considering rewriting that requirement such that it has only one interpretation rather than the several now possible. Consideration is also being given to raising the value of $0.7f_s'$ or eliminating the requirement entirely and, instead, retaining the $0.8f_s'$ or some other limitation on temporary jacking force.

Paragraph 2606 (a) 2 of ACI 318-63 refers to "tendons" rather than to an individual tendon. Further, the paragraph does not refer to the location to be considered for the determination of f_s in the manner, for example, of the "temporary jacking force" referred to in paragraph 2606 (a) 1.

Two interpretations were therefore required. Both interpretations had to consider the effect of the resultant actions on both the prestressing system and structure.

The first interpretation was that the location for measurement of the seating force, used in calculating f_s' was at the end anchor and just subsequent to the measurement of the "temporary jacking force" referred to in ACI 2606 (a) 1. The advantages of this location are several. One is that it is a practical one and thus the possibility for achieving valid measurements is greater. The second is that it is the same location used for measuring the "temporary jacking force" and measurements could be made without the added complexity of additional measuring devices. The third advantage is that measurements at this location provide assurance that the calculated f_s' does not anywhere exceed the maximum f_s' to which that tendon has been subjected.

Several possible cases were considered for the second interpretation so as to allow anchoring of an individual tendon without exceeding the requirement stated for "tendons" collectively in ACI 318-63 2606 (a) 2. One such case assumed that the anchoring force for the typical tendon was that for a tendon anchored midway through the prestressing sequence. It further assumed that the losses to be assumed were one-half of the sum of elastic losses, and of the creep, shrinkage and relaxation predicted to occur during the entire prestressing sequence. This interpretation however was not considered to be practical nor enforceable since it resulted in changing the seating forces as the actual, (as compared to the schedule), time length of the prestressing period was dictated by weather, and manpower availability.

Another case considered was that of anchoring each tendon at a measured force of 850 kips ($0.8f_s'$). Although there was no apparent detrimental effect to the prestressing system or structure, insertion of shims would be almost impossible. Further, it was concluded that this case would not establish compliance with ACI 318-63.

The case adopted was to seat each tendon with a measured "pressure" reading for the jack, at "lift-off" of the end anchor, of 775 kips (between 0.72 and 0.73 f_s'). This procedure had several advantages.

One advantage was that the force on the containment and the tendon was within the bounds of those for which it had been tested and resulted in no known detrimental effects. The second advantage was that the stressing procedure was simplified, since the stressing crews did not have to accommodate a large number of different anchoring force requirements. The third advantage was that, at the completion of stressing the last tendon, the expected losses were such that the average f_s' at the end anchors of the tendons would be less than 0.7 f_s', thus establishing compliance with ACI 318-63 2606 (a) 1 and 2. The fourth advantage was that the percentage loss of prestressing force was less than would be the case if the tendons were anchored in such a manner the calculated value of f_s' nowhere exceeded 0.7 f_s'.

The latter advantage deserves special mention since it plays a strong role in assuring that the Final Effective Prestress equalled or exceeded the desired value. For example, if the f_s' at anchorage of the tendons were 0.1 f_s', creep and shrinkage of concrete could result in the loss of almost all of the prestressing force. Assuming that the total losses due to creep, shrinkage and elastic shortening equals 0.1 f_s', then the Final Effective Prestress would be 20% of an initial prestress equivalent to 0.5 f_s'. If the initial prestress were equivalent to 0.7 f_s', the Final Effective Prestress, neglecting relaxation for the moment, would be about 86% of the initial prestress. Clearly, the assurance (that the concrete creep and shrinkage losses have been properly accounted for) increases as the f_s' for the anchored tendons and tendon increases. However, this design was committed to meeting the ACI 318-63 requirement and the anchorage force for the tendons was kept at or below 0.7 f_s' in accordance with the interpretation described.

5.1.4.3 Liner Plate

The design criteria which are applied to the Reactor Building liner to assure that the specified leak rate is not exceed under accident conditions are as follows:

a. That the liner be protected against damage by missiles (see 5.1.5.3).

b. That the liner plate strains be limited to allowable values that have been shown to result in leak tight vessels or pressure piping.

c. That the liner plate be prevented from developing significant distortion.

d. That all discontinuities and openings be well anchored to accommodate the forces exerted by the restrained liner plate, and that careful attention be paid to details of corners and connections to minimize the effects of discontinuities.

The most appropriate basis for establishing allowable liner plate strains is considered to be the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Article 4. Specifically the following sections have been adopted as guides in establishing allowable strain limits:

Paragraph N-412 (m)	Thermal Stress (2)
Paragraph N-414.5	Peak Stress Intensity Table N-413 Figure N-414, N-415 (A)
Paragraph N-412 (n)	
Paragraph N-415.1	

Implementation of the ASME design criteria requires that the liner material be prevented from experiencing significant distortion due to thermal load and that the stresses be considered from a fatigue standpoint (Paragraph N-412 (m) (2)).

The following fatigue loads are considered in the design of the liner plate:

a. Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 40 cycles for the plant life of 40 years.

b. Thermal cycling due to Reactor Building interior temperature varying during the startup and shutdown of the reactor system. The number of cycles for this loading is assumed to be 500 cycles.

c. Thermal cycling due to the loss-of-coolant accident will be assumed to be one cycle. Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by the concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Section III fatigue considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the plant life.

The thermal stresses in the liner plate fall into the categories considered in Article 4, Section III, Nuclear Vessels of the ASME Boiler and Pressure Vessel Code. The allowable stresses in Figure N-415 (A) are for alternating stress intensity for carbon steel and temperatures not exceeding 700°F.

In accordance with ASME Code, Paragraph 412 (m) (2), the liner plate is restrained against significant distortion by continuous angle anchors and never exceeds the temperature limitation of 700°F and also satisfies the criteria for limiting strains on the basis of fatigue consideration.

Paragraph 412 (n), Figure N-415 (A) of the ASME Code has been developed as a result of research, industry experience, and the proven performance of code vessels, and it is a part of a recognized design code. Figure N-415 (A) and its appropriate limitations have been used as a basis for establishing allowable liner plate strains. Since the graph in Figure N-415 (A) does not extend below ten cycles, ten cycles are being used for a loss of coolant accident instead of one cycle.

The maximum compressive strains are caused by accident pressure, thermal loading prestress, shrinkage and creep. The maximum strains do not exceed .0025 inch/inch and the liner plate always remains in a stable condition.

At all penetrations the liner plate is thickened to reduce stress concentrations in accordance with the ASME Boiler and Pressure Vessel Code 1965, Section III, Nuclear Vessels.

The liner plate is anchored as shown in Figure 5-1 with anchorage in both the longitudinal and hoop direction. The anchor spacing and welds were designed to preclude failure of an individual anchor. The load deformation tests referred to in 5.1.3.2 indicate that the alternate stitch fillet weld used to secure the anchor to the liner plate would first fail in the weld and not jeopardize the liner plate leak tight integrity.

Offsets at liner plate seams are controlled in accordance with ASME Section III Code, which allows 1/16 inch misalignment for 1/4 inch plate. The flexural strains due to the moment resulting from the misalignment were added to calculate the total strain in the liner plate.

The liner plate plus structural shapes to support the liner are ASTM A36 or ASTM A516 steel. The selection of this material complies with "Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors" prepared by Subcommittee N6.2, Containment, of ASA Sectional Committee N6, Reactor Safety Standards.

5.1.4.4 Penetrations

Penetrations conform to the applicable sections of ASA N6.2-1965, "Safety Standard for the Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors." All personnel locks and any portion of the equipment access door extending beyond the concrete shell conform in all respects to the requirements of ASME Section III, Nuclear Vessels Code.

The basis for limiting strains in the penetration steel is the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, Article 4, 1965, and therefore, the penetration structural and leak tightness integrity are maintained. Local heating of the concrete immediately around the penetration will develop compressive stress in the concrete adjacent to the penetration and a negligible amount of tensile stress over a large area. The mild steel reinforcing added around penetrations distributes local compressive stresses for overall structural integrity.

Horizontal and vertical bonded reinforcement is provided to help resist membrane and flexural loads at the penetrations. This reinforcement was located on both the inside and outside face of the concrete. Stirrups were also used to assist in resisting shear loads.

Local crushing of the concrete due to deflection of the reinforcing or tendons is precluded by the following details:

a. The surface reinforcements either have a very large radius such as hoop bars concentric with the penetration or are practically straight, having only standard hooks as anchorages where necessary.

b. The tendons are bent around penetrations at a mimimum radius of approximately 20 feet. Maximum tendon force at initial prestress is 850 kips, which results in a bearing stress of about 880 psi on the concrete.

It is also important to note that the deflected tendons are continuous past the openings and are isolated from the local effects of stress concentrations by virtue of being unbonded.

In accordance with ASME Section III, piping penetration reinforcing plates and the weldment of the pipe closure to it are stress relieved. This code requirement and the grouping of penetrations into large shop assemblies permit a minimum of field welding at penetrations.

The personnel hatch consists of a steel cylinder with 3 ft.-6 in. x 6 ft.-8 in. doors at each end interlocked so that only one door can be open at any time. The hatch is designed to withstand all Reactor Building design conditions with either or both doors closed and locked. Doors open toward the center of the Reactor Building and are thus sealed under Reactor Building pressure. Design live load on the hatch floor is 200 psf.

Operation of the hatch is normally manual, that is, without power assist. Interlocks will prevent opening both doors at once.

Double gaskets are provided on the outer door to permit periodic pressurizing of the space between the gaskets from outside the Reactor Building. The hatch barrel may be pressurized to demonstrate its leak tightness without pressurizing the Reactor Building. Auxiliary restraint beams are attached to the inner door in this case to help the locking bars to resist internal lock pressure, which is greatly in excess of the Reactor Building design external pressure of 3 psig. The personnel hatch was pneumatically shop tested for pressure and leakage.

Figure 5-3 shows the principal features of the personnel hatch.

An emergency hatch is provided with 30 inch diameter doors. Its features are identical to the personnel hatch.

A 19-foot diameter equipment hatch opening to the outside provides for movement of large items into and out of the Reactor Building. The door is secured by bolts on the inside of the Reactor Building wall and can be opened only from inside the Reactor Building. It is opened only when the reactor is subcritical. Double gaskets on the door permit the seals to be pressurized from outside the Reactor Building to check the integrity of the seals. During operation, the space between the double gaskets is vented to the penetration room.

Figure 5-3 shows the principal features of the equipment hatch.

a. Piping and Ventilation Penetrations

All piping and ventilation penetrations are of the rigid welded type and are solidly anchored to the Reactor Building wall or foundation slab, thus precluding any requirements for expansion bellows. All penetrations and anchorages are designed for the forces and moments resulting from operating conditions. External guides and stops are provided as required to limit motions, bending and torsional moments to prevent rupture of the penetrations and the adjacent liner plate for postulated pipe rupture. Piping and ventilation penetrations have no provision for individual testing since they are of all-welded construction.

For typical details of piping penetrations, see Figure 5-2.

b. Electrical Penetrations

Medium voltage penetrations for reactor coolant pump power shown on Figure 5-2 are canister type using glass sealed bushings for conductor seals. The canisters are filled to a positive pressure with an inert gas. The assemblies are bolted to mating flanges which incorporate double "O" ring seals with a test port between as a means of verifying seal integrity.

Low voltage power, control and instrumentation assemblies are shown on Figure 5-2. These assemblies are designed to bolt to mating flanges mounted inside the Reactor Building. Each assembly includes two header plates to which are welded glass to metal sealed conductors. The space between the seal headers is piped to a pressure gage and a charging valve located outside of the Reactor Building. This test volume is pressurized with an inert gas. Dual "O" rings with a test port between are used to complete the seal to the mating flange, which is welded to the penetration nozzle.

5.1.4.5 Miscellaneous Considerations

In various cases, it has been the designer's decision to provide structural adequacy beyond that required by the design criteria. Those cases are as follows:

a. Section 5.1.2.5 requires a mimimum of 0.15 percent bonded reinforcing steel in two perpendicular directions on the exterior faces of the wall and dome for proper crack control. Due to the weather exposure, a mimimum of approximately 0.5 percent was provided.

b. Section 5.1.2.5 requires a minimum of 0.15 percent bonded steel reinforcing (as stated above) for any location. At the base of the cylinder, the controlling design case requires 0.25 percent vertical reinforcing. As a result of pursuing the recommendation of the AEC Staff to further investigate current research on shear in concrete, several steps were taken:

1. The work of Dr. Alan H. Mattock was reviewed and he was retained as a consultant on the implementation of the current research being conducted under his direction. The criteria has been updated in accordance with his recommendations.

2. Concurrently with reviewing Dr. Mattock's work, the firm of T. Y. Lin, Kulka, Yang and Associates was consulted to review the detailed design of the cylinder to slab connection. It was their recommendation to use approximately 0.5 percent reinforcing rather than the 0.25 percent reinforcing indicated by the detailed design analysis for the vertical wall dowels. This increase would assure that there was sufficient flexural steel to place the section within the lower limits of Mattock's test data (approximately 0.3 percent) to prevent flexural cracking from adversely affecting the shear capability of the section.

5.1.5 INTERIOR STRUCTURE

5.1.5.1 Design Bases

The Reactor Building interior structure (comprising all elements inside the Reactor Building shell) is a Seismic Class I structure and is designed on the following bases:

a. The stresses in any portion of the structure under the action of dead
load, live load and design seismic load will be below the allowable
1. stresses given by either the ACI Building Code, ACI 318-1963 except as
noted in 5.1.2.6, AISC Manual of Steel Construction, 6th Edition.

b. The stresses in any portion of the structure under the action of dead load, and thermal load will be below 133 percent of the allowable stresses given in (a).

c. The capability to safely shut down the plant will be maintained under the combined action of dead load, maximum seismic load, pressure and jet impingement load. The latter two loads are based on the rupture of one pipe in the primary loop. The deflections of structures and supports under these combined loads would be such that the functioning of engineered safeguards equipment would not be impaired. The yield load equations in Appendix 5A are adhered to except that local yielding is permitted for pipe, jet or missile barriers provided there is no general failure.

Rev. 1. 9/15/69

5.1.5.2 Design Loads and Materials

The Reactor Building interior structure consists of (1) the reactor cavity, (2) two steam generator compartments, and (3) a refueling pool which is located between the steam generator compartments and above the reactor cavity.

The reactor cavity houses the reactor vessel and serves as a biological shield wall. The reactor cavity is also designed to contain core flooding water up to the level of the reactor nozzles.

The primary functions of the steam generator compartment walls are to serve as secondary shield walls and to resist the pressure and jet loads described below.

The foundations for all NSSS equipment including the reactor vessel, the steam generators, and the pressurizer are designed to remain within the elastic range during rupture of any pipe combined with the "maximum earthquake."

The design pressure differential across walls and slabs of enclosed compartments in the internal structure are as follows:

Reactor Cavity	- 208 psi
East Steam Generator Compartment	- 11.1 psi
West Steam Generator Compartment	- 11.1 psi

In addition to the peak pressure differentials, the steam generator compartment walls are designed for simultaneous action of a single jet impingement load and the safe shutdown earthquake. Design of structures was done using conventional structural analytical techniques.

Pipe whipping restraints are provided for the main steam, feedwater and other high-pressure piping in accordance with criteria in Section 5.4.

The materials used for the above structural elements are as follows:

Structural Steel	-ASTM A36
Concrete	$-f'_c$ = 4000 psi at 28 days.
	$-f'_c$ = 5000 psi at 28 days (for steam generator bases, reactor foundation, and primary shield wall).
Reinforcing Bars	-ASTM A615, Grade 40 for Bars #11 and under ASTM A615, Grade 60 for Bars larger than #11.

5.1.5.3 Missile Protection

High-pressure reactor coolant system equipment which could be the source of missiles is suitably screened by the concrete shield wall enclosing the reactor coolant loops and by special missile shields to block any passage of missiles to the Reactor Building walls. Potential missile sources are oriented so that the potential missile will be intercepted by the shields and structures provided. A structure is provided over the control rod drive mechanisms to block any missiles generated from the unlikely fracture of the mechanisms.

Missile protection is provided to comply with the following criteria:

a. The Reactor Building and liner are protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe.

b. The engineered safeguards system and components required to maintain Reactor Building integrity are protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria was developed and implemented using the following methods:

a. Components of the reactor coolant system were examined to identify and to classify missiles according to size, shape and kinetic energy for purposes of analyzing their effects.

b. Missile velocities were calculated considering both fluid and mechanical driving forces which can act during missile generation.

c. The reactor coolant system is surrounded by reinforced concrete and steel structures designed to withstand the forces associated with double-ended rupture of a main coolant pipe and designed to stop missiles.

d. The structural design of the missile shielding takes into account both static and impact loads and is based upon the state of the art of missile penetration data.

The types of missiles for which missile protection is provided are:

a. Valve stems.

b. Valve bonnets.

c. Instrument thimbles.

d. Various types and sizes of nuts and bolts.

Protection is not provided for certain types of missiles for which postulated accidents are considered incredible because of the material characteristics, inspections, quality control during fabrication, and conservative design as applied to the particular component. Included in this category are missiles caused by massive, rapid failure of the reactor vessel, steam generator, pressurizer, main coolant pump casings and drives.

5.2 ISOLATION SYSTEM

5.2.1 DESIGN BASES

The general design basis governing isolation valve requirements is:

> Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Reactor Building, and various types of isolation valves.

Reactor Building isolation occurs on a signal of approximately 4 psig in the Reactor Building. Valves which isolate penetrations that are directly open to the Reactor Building, such as the Reactor Building purge valves and sump drain valves, will also be closed on a high radiation signal. (11.1.2.4.2)

The isolation system closes all fluid penetrations, not required for operation of the engineered safeguards systems, to prevent the leakage of radioactive materials to the environment.

All remotely operated Reactor Building isolation valves are provided with position limit indicators in the control room.

5.2.2 SYSTEM DESIGN

The fluid penetrations which require isolation after an accident may be classed as follows:

Type I. Each line connecting directly to the reactor coolant system has two Reactor Building isolation valves. One valve is inside and the other valve is outside the Reactor Building. These valves may be either a check valve and a remotely operated valve, or two remotely operated valves, depending upon the direction of normal flow.

Type II. Each line connecting directly to the Reactor Building atmosphere has two isolation valves. At least one valve is outside and the other may be inside or outside the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves, depending upon the direction of normal flow.

Type III. Each line not directly connected to the reactor coolant system or not open to the Reactor Building atmosphere has at least one valve, either a check valve or a remotely operated valve. This valve is located outside the Reactor Building.

Type IV. Lines which penetrate the Reactor Building and are connected to either the building or the reactor coolant system, but which are not normally open during reactor operation, may have manual valves with provisions for locking in a closed position.

There are additional subdivisions in each of these major groups. The individual system flow diagrams show the manner in which each Reactor Building isolation valve arrangement fits into its respective system. For convenience, each different valve arrangement is shown in Table 5-4 and Figures 5-16 through 5-19 of this section. The symbols on these figures are identified on Figure 9-1. This table lists the mode of actuation, the type of valve, its normal position and its position under Reactor Building isolation conditions. The specific system penetrations to which each of the arrangements is applied is also presented. It may be noted that only electric motor-operated or check valves are used inside the Reactor Building. Each valve will be tested periodically during normal operation or during shutdown conditions to assure its operability when needed.

The accident analysis for failure or malfunction of each valve is presented with the respective system evaluation of which that valve is a part, eg, chemical addition and sampling system, etc. in Sections 6 and 9.

There is sufficient redundancy in the instrumentation circuits of the engineered safeguards protective system to minimize the possibility of inadvertent tripping of the isolation system. Further discussion of this redundancy and the instrumentation signals which trip the isolation system is presented in Section 7.

The system abbreviations which are used in column three of Table 5-4 are defined as follows:

HP	High Pressure Injection System
LP	Low Pressure Injection System
CC	Component Cooling System (Reactor Building)
SF	Spent Fuel Cooling System
WD	Waste Disposal System
CA	Chemical Addition and Sampling System
BS	Reactor Building Spray System
LPSW	Low Pressure Service Water System
CF	Core Flooding System
SS	Steam Supply System

Table 5-4 can be found at the end of Section 5.

5.3 VENTILATION SYSTEM

5.3.1 DESIGN BASES

5.3.1.1 Governing Conditions

The Reactor Building normal ventilation system is composed of the normal cooling system and the purge system and accomplishes two functions. One function is the removal of normal heat loss from equipment and piping in the Reactor Building, and the other is to purge the Reactor Building with fresh air whenever desired.

The Reactor Building normal and emergency cooling units are combined into one system. The Reactor Building's ventilation system is described in Section 6, "Engineered Safeguards."

5.3.1.2 Sizing

To provide for access to the Reactor Building, the normal ventilation system is sized to control the interior air temperature to 104 F in accessible areas during operation and a minimum of 60 F during shutdown.

4. The purge system equipment is sized for a flow rate of 50,000 scfm, providing approximately 1.5 air changes per hour in the Reactor Building. The normal cooling units will be utilized and are sized to distribute adequate air over and around all heat producing or releasing equipment.

5.3.2 SYSTEM DESIGN

A flow diagram of the normal ventilation and purge systems is shown in Figure 5-20.

The normal cooling system consists of fan-cooler units located outside the secondary shielding. These units recirculate and cool the Reactor Building atmosphere. The coolers use low pressure service water as the heat removal medium. The fan units discharge the cooled air through ducts to provide adequate distribution for the equipment and areas including the reactor cavity.

The purge system consists of a heater and filters and a discharge fan-filter unit. All of the purge system, except interior ducts, and two isolation valves are located outside the Reactor Building. Ducts are provided inside the Reactor Building for adequate distribution.

The purge system discharge to the unit vent is monitored and alarmed to prevent release exceeding acceptable limits.

5.3.2.1 Isolation Valves

Since the normal cooling system is contained completely within the Reactor Building, it does not include provisions for any isolation valves other than on cooling water lines. The purge system is provided with double automatic isolation valves (or dampers) in both the supply and discharge ducts. These valves are normally closed and will be opened only for the purging operation. They are electrically actuated inside the Reactor Building and pneumatically actuated outside the Reactor Building. Refer to 6.4 for a discussion of the penetration room ventilation system.

The isolation signal and controls are discussed in 5.2. Operability testing of the isolation valves is accomplished each time the purge system is put into operation. Analysis of effect on LOCA dose in the event a LOCA occurred
4. while the Reactor Building is being purged is reported in 14.2.2.3.7.

5.4 LEAKAGE MONITORING SYSTEM

No continuous leakage monitoring system will be provided.

The barrier to leakage in the Reactor Building is the one-quarter inch steel liner plate. All penetrations are continuously welded to the liner plate before the concrete in which they are embedded is placed. The penetrations, shown on Figure 5-2 and 5-3, become an integral part of the liner and are so designed, installed, and tested.

The steel liner plate is securely attached to the prestressed concrete Reactor Building and is an integral part of this structure. This Reactor Building is conservatively designed and rigorously analyzed for the extreme loading conditions of a highly improbably hypothetical accident, as well as for all other types of loading conditions which could be experienced. Thorough control is maintained over the quality of all materials and workmanship during all stages of fabrication and erection of the liner plate and penetrations and during construction of the entire Reactor Building.

The comprehensive program for preoperational testing, inspection, and postoperational surveillance is described in detail in 5.6, "Tests and Inspection," and is summarized in the following paragraphs.

During construction, the entire length of every seam weld in the liner plate is leak tested. Individual penetration assemblies are shop tested. Welded connections between penetration assemblies and the liner plate are individually leak tested after installation. Following completion of construction, the entire Reactor Building, the liner, and all its penetrations are tested at 115 percent of the design pressure to establish structural integrity. The initial leak rate tests of the entire Reactor Building are conducted at the maximum calculated peak accident pressure and at one-half this pressure to demonstrate vapor tightness and to establish a reference for periodic leak testing for the life of the station. Multiple and redundant systems based on different engineering principles are provided as described in Section 6, "Engineered Safeguards," to provide a very high degree of assurance that the accident conditions will never be exceeded and that the vapor barrier of the containment will never be jeopardized.

Under all normal operating conditions and under accidental conditions short of the worst loss-of-coolant accident, virtually no possibility exists that any leakage could occur or that the integrity of the vapor barrier could be violated in any way that would be significant to the public health and safety or to that of the station personnel. Adequate administrative controls will be enforced to minimize the possibility of human error. Station operators will be trained and licensed in accordance with regulations. Safety analyses are presented in Section 14.

Penetrations such as the personnel access and emergency hatches cannot be opened except by deliberate action and are interlocked and alarmed by fail-safe devices such that the Reactor Building will not be breached unintentionally. The liner plate over the foundation slab is protected by cover concrete. Wherever access to the liner plate is blocked by interior concrete, means are provided so that weld seams can be tested for leakage. The liner plate is protected against corrosion by suitable coatings. Walls and floors for biological and missile shielding, and for access and operating purposes, also provide compartmentation which constitutes protection for the liner during operating as well as accident conditions.

Once the adequacy of the liner has been established initially, there is no reason to anticipate progressive deterioration during the life of the station which would reduce the effectiveness of the liner as a vapor barrier. Inside the Reactor Building, the atmosphere is subject to a high degree of temperature control. The outside of the liner is protected by 3-3/4 feet of prestressed concrete which is exceptionally resistant to all weather conditions.

Inspection on a periodic basis, as necessary, will be conducted in all spaces accessible under full power operation. Biological shielding is provided to reduce radiation to limits which make occupancy of spaces adjacent to the liner permissible.

All penetrations except the following are grouped within or vented to the penetration room. Any leakage that might occur from these penetrations will be collected and discharged through high efficiency particulate air (HEPA) filters and charcoal filters to the unit vent as described in 6.4. In this manner, leakage which might occur from these penetrations will be isolated from leakage which might occur through the Reactor Building itself.

a. Main Steam Lines

b. Sump Drain Lines

c. Decay Heat Removal Lines

d. Reactor Building Equipment Drain Lines

The above lines are not considered a source of significant leakage because they are welded to the liner plate.

Individual major penetrations or groups of penetrations will be tested by means of permanently installed pressure connections or temporarily installed pressure or vacuum boxes. If necessary, liner plate weld seams will be tested by the vacuum box soap bubble method where accessible, or by means of the permanently installed backup channels and angles where inaccessible.

In any event, sources of excessive leakage will be located and such corrective action as necessary will be taken. This will consist of repair or replacement. Appropriate action will also be taken to minimize the possibility of recurrence of excessive leakage, including such redesign as might prove to be necessary to protect public health and safety. Leak testing will be continued until a satisfactory leak rate has again been demonstrated.

A considerable background of operating experience is being accumulated on containments and penetrations. Full advantage of this knowledge has been taken in all phases of design, fabrication, installation, inspection and testing. Practical improvements in design and details have been incorporated as they are developed, where applicable.

The steel-lined Reactor Building is self-sufficient, and other than valves and hatch doors, there are no operating parts. The containment boundary is extended only by listed penetrations and further described and tabulated in 5.2, "Isolation System" and 5.3, "Ventilation System."

5.5 SYSTEM DESIGN EVALUATION

The penetration room ventilation system described in 6.4 provides a partial double containment system and is an additional engineered safeguard.

A full evaluation of the containment system which is provided is included in 5.4, "Leakage Monitoring System," in justification of not providing such a monitoring system. The Reactor Building with the appurtenant engineered safeguards systems will prevent uncontrolled release of radioactivity to plant and surrounding areas during normal operating and accident conditions, as well as for lesser accidental conditions. Containment integrity is maintained whenever, simultaneously, the reactor coolant system is pressurized above 300 psig, the reactor coolant temperature is 200 F or above and there is nuclear fuel in the core.

5.6 TESTS AND INSPECTION

5.6.1 PREOPERATIONAL TESTING AND INSPECTION

5.6.1.1 During Construction

Test, code, and cleanliness requirements accompanied each specification or purchase order for materials and equipment. Hydrostatic, leak, metallurgical, electrical, and other tests to be performed by the supplying manufacturers are enumerated in the specifications together with the requirements, if any, for test witnessing by an inspector. Fabrication and cleanliness standards, including final cleaning and sealing, are described together with shipping procedures. Standards and tests are specified in accordance with applicable regulations, recognized technical society codes and current industrial practices.

Inspection is performed in the shops of vendors and subcontractors as necessary to verify compliance with specifications.

The following codes of practice are used to establish standards of construction procedure:

ACI 301	-	Specification for Structural Concrete for Buildings (Proposed)
ACI 318	-	Building Code Requirements for Reinforced Concrete
ACI 347	-	Recommended Practice for Concrete Framework
ACI 605	-	Recommended Practice for Hot Weather Concreting
ACI 613	-	Recommended Practice for Selecting Proportions for Concrete
ACI 614	-	Recommended Practice for Measuring, Mixing and Placing Concrete
ACI 315	-	Manual of Standard Practice for Detailing Reinforced Concrete Structures
ASME	-	Boiler and Pressure Vessel Code, Sections III, VIII, and IX
AISC	-	Steel Construction Manual
PCI	-	Inspection Manual

5.6.1.1.1 Concrete

Testing of concrete materials and concrete as placed is described in Appendix 1B. An experienced full-time concrete inspector continuously checked concrete batching and placing operations.

Concrete mixes were designed and the associated tests run by the concrete testing laboratory at Clemson University in accordance with ACI 613. During construction, the field inspection personnel made minor modifications that were necessitated by variations in aggregate gradation or moisture content.

In determining the design mixes, air content, slump and bleeding tests were run in accordance with the appropriate ASTM Specifications.

The concrete ingredients consist of Type II Cement (ASTM C-150), Solar 25 air entraining agent (ASTM C-260), Plastiment water reducing agent (ASTM C-494), Aggregate (ASTM C-33) and water that was free from injurious amounts of chlorides, sulphates, oil, acid, alkali, organic matter, or other deleterious substances.

Fine aggregate consists of clean, sharp, washed sand of uniform gradation from Becker County Hagood Quarry. Coarse aggregate consists of washed crushed rock having hard, strong, durable pieces of Gaffney marble from Campbell Limestone Company. The acceptability of the aggregate was based on Los Angeles Abrasion, Clay Lumps Natural Aggregates, Material Finer number 200 sieve, Organic impurities effect on Mortar, Organic impurities - Sands, Potential Reactivity, Seive Analysis, Soundness, Specific Gravity and Absorption, and Petrographic tests based on the appropriate ASTM Specificiations.

Cast-in-place concrete was used to construct the Reactor Building shell. The base slab construction was performed in seven pours utilizing large block pours. After the completion of the base slab steel liner erection and testing, an additional concrete slab was placed to provide protection for the floor liner.

The concrete placement in the walls was done in 10 ft high lifts with vertical joints at the radial center line of each of six buttresses. Cantilevered jump forms on the exterior face and interior steel wall liner served as the forms for the wall concrete.

The dome liner plate, temporarily supported by 18 radial steel trusses and purlins, served as an inner form for the initial 8 inch thick pour in the dome. The weight of the subsequent pour was supported in turn by the initial 8 inch pour. The trusses were lowered away from the liner plate after the initial 8 inches of concrete had reached design strength, but prior to the placing of the balance of the dome concrete.

The horizontal and the vertical construction joints were prepared by blasting
1. with compressed air. Horizontal surfaces were covered with approximately 1/4
inch thick mortar of the same cement-sand ratio as used in the concrete immediately before concrete placing.

5.6.1.1.2 Prestressing

Testing and inspection of all prestressing materials and special installation
5. equipment is described in Appendix 5B. Full-time supervision of the pre-stressing
operation was provided.

The BBRV post-tensioning system furnished by The Prescon Corporation was used.

Each tendon consists of ninety 1/4 inch diameter wires conforming with ASTM
3. A-421-65T, two anchor heads and two sets of shims conforming with AISI C1045HR.
The tendon sheathing system consists of spirally wound carbon steel tubing connecting to a trumplate (bearing plate and trumpet) at each end. The bearing plates were fabricated from steel plate conforming with AISI C1045HR and the trumpets from AISI C1010HREW material.

The C-1045 HR material used for the stressing washers, deadend washers, shims and
bearing plates was modified by the addition of silicon to obtain a finer grain
7. structure and cleaner steel than unmodified C-1045. The average depth of the heat
affected zone resulting from flame cutting is approximately 1/16 inch and the improved general ductility of modified C-1045 material should increase resistance to cracks starting in heat affected zones and decrease the probability of crack propagation. However, a cracked plate could continue to perform its function without loss of structural integrity and should be evaluated in terms of actual functional ability.

Flame cutting is limited to sizing the bearing plate and making the center hole. 7. All other holes in the bearing plate are drilled. The deadend washer is flame cut to size and drilled for the tendon wires. No flame cutting is performed on the stressing washer.

Tendons were delivered to the site coated with a rust preventive and encased in polyethylene bags. Each tendon came precut to exact length, with one end unfinished and the other end shop buttonheaded, and with its anchor head attached.

The tendon installation prestressing procedure was carried out as follows:

a. To assure a clear passage for the tendons, a "sheathing rabbit" was run through the sheathing both prior to and following placement of the concrete.

b. Tendons were uncoiled and pulled through the sheathing unfinished end first.

c. The unfinished end of the tendons was pulled out with enough length exposed so that field attachment of the anchor head and buttonheading could be performed. To allow this operation, trumplates on the opposite end had an enlarged diameter to permit pulling in the shop finished ends with their anchor heads.

d. The anchor heads were attached and the tendon wires buttonheaded.

e. The shop finished end of the tendon was pulled back and the stressing jack attached.

f. The post-tensioning was done by jacking to the permissible overstressing force to compensate for friction and placing the shims precut to lengths corresponding to the calculated elongation. Proper tendon stress was achieved by comparing both jack pressure and tendon elongation against previously calculated values. The 5. vertical tendons were prestressed from either one or both ends, while the horizontal and dome tendons were prestressed from both ends.

g. The grease caps were bolted onto anchorages at both ends and made ready for pumping the tendon sheathing filler material.

h. The tendon sheaths and grease caps were filled with sheathing filler and sealed. The sheathing filler material had limitations specified for deleterious water soluble salts.

Corrosion protection of the tendons and interior surface of sheathing was applied prior to shipment.

Tendon sheaths mark 24H34, 13H34 and 34V14 on Unit 1 and 13H21 on Unit 2 were plugged. The location of the plugged sheaths are shown 21. 9. in Figure 5-23.

The Reactor Building has been analyzed based on the above missing tendons for the various loading conditions including missiles. The

stresses for the various loading conditions were within the allowable design stresses. The missing tendons will not have any affect on the structure to withstand turbine and tornado generated missiles without loss of function. The missing tendons are located on the northwest face and shielded by location from a direct turbine missile strike.
9. However, as stated in Section 5.1.3.2, the structure can withstand the loss of three horizontal and three vertical tendons in the cylinder wall without loss of function. The depth of penetration from tornado generated missiles as stated in Section 5.1.3.2 is less than the tendon concrete cover and will not endanger the structural integrity of the Reactor Building.

5.6.1.1.3 Reinforcing Steel

Testing and inspection of reinforcing steel is described in Appendix 1B.

The concrete inspector visually inspected the shop fabricated reinforcing steel for compliance with drawings and specifications. Intermediate grade reinforcing steel conformed with ASTM A615, Grade 40 and high strength reinforcing steel conformed with ASTM A615, Grade 60.

Whenever required, mechanical splices were made by the CADWELD process using clamping devices, sleeves, charges, etc., as specified by the manufacturer for "T" series connections. All personnel engaged in making the splices were trained and supervised by the manufacturer's representative and had passed all the necessary qualification tests and procedures before production splicing. Prior to splicing operations, bar ends were inspected for damaged deformations and were power brushed to remove all loose mill scale, rust, and other foreign material. Immediately before the splice sleeve positioning, bar ends were preheated to assure complete absence of moisture.

All completed splices were visually inspected, at both ends of the splice,
1. for sound and nonporous filler material. The strength of 95 percent of the CADWELD joints, as verified by tests, was greater than 125 percent of the ASTM specified minimum yield strength of the reinforcing bars used. The average strength of all test splices exceeded the ASTM ultimate strength of the reinforcing bar used.

5.6.1.1.4 Liner Plate

Testing and inspection of liner plate is described in Appendix 1B.

Construction of the liner plate conformed to the applicable portions of Part UW of Section VIII of the ASME Code. Specifically, Paragraphs UW-26 through UW-38, inclusive, applied in their entirety. In addition, the qualification of all welding procedures and welders was performed in accordance with Part A of Section IX of the ASME Code. All liner angle welding was visually inspected prior to, during, and after welding to insure that quality and general workmanship met the requirements of the applicable welding procedure specification.

The erection of the liner plate was as follows:

After the floor plate embedments in the foundation slab had been placed and
1. welded, and concrete was poured flush, the wall liner plates were erected in
60 degree segments and 10 feet high courses. This pattern was followed to the
dome spring line and then the steel dome erection trusses were placed. During
1. the period of erection of wall liner plates, the floor liner plate was placed
and welded.

The tolerances for liner plate erection were as follows:

1. a. The location of any point on the liner plate shall not vary from the design diameter by more than ± 3 inches.

b. Maximum inward deflection (toward the center of the structure) of the
2. 1/4 inch liner plate between the angle stiffeners of 1/8 inch, when
measured with a 15 inch straightedge placed horizontally.

5.6.1.2 Structural Test

Each of the three Reactor Buildings will be pressurized to 115 percent of design pressure for one hour following completion of construction to establish the structural integrity of the building. The structural integrity test of each building will be conducted in accordance with a written procedure. Operating units will remain in operation during the structural test of another unit. Personnel access limitations included in the written procedures will designate areas of limited access during specific periods of the test. Except for per-
5. sonnel access restrictions, the operation of one unit will not be affected by a building being tested.

The structural integrity test of each building will verify the workmanship involved; in addition, the test of the Unit 1 Reactor Building will verify the design and workmanship. The response of the Unit 1 building will be compared with the calculated behavior to confirm the design by means of instrumentation.

5.6.1.2.1 Test Objectives

a. To provide direct verification that the structural integrity as a whole is equal to or greater than necessary to sustain the forces imposed by two different and large loading conditions.

b. To provide direct verification that the in-place tendons (the major strength elements) have a strength of at least 80 percent of guaranteed ultimate tensile strength and that the concrete has the strength needed to sustain a strain range from high initial average concrete compression when unpressurized to low average concrete compression when pressurized.

c. To acquire detailed strain data which will be compared with the analytical predictions.

To achieve objectives, data will be acquired and evaluated to determine the response of the structure during and immediately after post-tensioning to determine any indication of unanticipated and continued deformation under load. A quality assurance program was instituted as described in Appendix 1B. In addition, each individual tendon is tensioned in place to 80 percent of the guaranteed ultimate tensile strength and then anchored at a lower load that is still in excess of those predicted to exist at test pressure levels. During pressurization of the structure, the structure's response will be measured at selected pressure levels with the highest being 1.15 times the design pressure. An
1. indication that the structure is capable of withstanding internal pressure will result from these tests. The strain measuring program is described in 5.6.1.2.2.

Individual test values which fall outside the predicted ranges will not be considered as necessarily indicative of a lack of adequate structural integrity.
1. Structural integrity cannot be judged on the data acquired from only one sensor since such precise devices may malfunction.

5.6.1.2.2 Instrumentation

The structural response of the building will be assessed by comparing the theoretical analysis to test results of strains and deformations at boundaries, points of stress concentration, openings, areas of maximum creep, and at sections representing typical stress conditions.

20. 16. The following instruments were installed in the first Reactor Building.

118 Two element strain rosette, waterproofed BLH Company designation FAET-12-12-56, to be attached to the reinforcing bars.

9 Linear element, electric resistance strain gages, BLH designation AS9-1 (Valore Type) to be attached to the surface of the concrete.

16. Taut wire system for measuring building deformation.

6 Electric resistance strain gage, Budd Company designation CP-1101 EX to be attached to the surface of the concrete for measuring crack propagation.

5. 1 Cement Paint (Figure 5-21) to observe cracks in concrete.

7 Load cells each containing strain gages to be attached to the tendons.

18 Three element rosette, electric resistance strain gages BLH Company designation FAER-25-12-(60)56, to be attached to the inside and outside face of the liner and penetration nozzles.

26 Two element strain rosette, BLH Company designation FAET-25-12-5, to be attached to the inside face and outside face of the liner and penetration nozzles.

The instrument layout is shown on Figure 5-21, sheets 1, 2, and 3. The types and locations of the gages are described in the legend on the figure.

Because of the well-known vunerability of the bonded resistance gages to moisture, special care is taken in bonding and waterproofing of the gages.

In order to reduce the possibilities of faulty preparation of the gages in the field, the gages are encapsulated and the wires soldered to the gage leads and then waterproofed in the shop.

Bonding and waterproofing materials such as BLH EPY150 Cement, Epoxylite 222 and Microcrystalline Wax are used to install the gages.

Gages were calibrated in accordance with the manufacturer's instructions and set at zero reading during installation.

15.

The final procedures in sequence of structural proof testing are as follows:

a. Test strain gages immediately after installation.
b. Test strain gages immediately after pouring concrete.
c. Record strains and deflections and observe cracking at three intervals suitably spaced during prestressing and immediately after all prestressing is completed.
5. d. After prestressing and before testing, a certain number of readings will be taken to determine the effects of creep and shrinkage.
e. Record measurements at increments of 10 psi up to 40 psi and then at increments of 5 psi up to proof-test pressure.
f. Record measurements at 15 psi increments during depressurization.
g. Observe the development of cracks during load application. Measurement of cracks with mechanical dial gages will be made when deemed pertinent by the test engineer.

The Reactor Building air temperature is monitored by resistance thermometers and the dewpoint temperature is monitored by a dewpoint sensor. Using the Reactor Building coolers and electric heaters, the temperature is maintained between 60 and 100 F and above the dewpoint temperature.

The status of gages on November 28, 1970 was as follows:

	Gage Mark	Number Inoperative	Number Operative	Number Being Replaced
	SGA-1	114	4	(See b below)
	SGE-2	7	2	(See b below)
	SGC-3	0	6	-
15.	SGR-4	7	11	6
	SFT-5	7	19	6
	LC (Load Cell)	1	6	(See d below)
16.	Taut Wire System	0	-	-

Since a significant number of embedded gages are inoperative, we believe it prudent to verify the design by (a) utilizing test results from Palisades and, (b) continuing with the Oconee Structural Test, as noted below:

a) The design and construction of Palisades and Oconee Reactor Buildings are very similar. The Palisades' structural instrumentation program was successful and permitted a detailed comparison between design calculations and observed response.

Rev. 5. 5/25/70
Rev. 10. 8/28/70
Rev. 15. 12/30/70
Rev. 16. 7/30/71

16. b) At Oconee, the taut wire system (building deformation) will permit verification that the structural response is consistent with the predicted behavior. In addition, twenty-six Carlson SAIOS strain gages will be surface mounted on the Reactor Building to obtain concrete strains for comparison with Palisades and those predicted for Oconee as shown on Figure 5-21, Sheet 4.

15. c) Six inoperative gages mark SGR-4 and SFT-5 are accessible and will be replaced to obtain data for comparison with Palisades and predicted strains for Oconee.

d) Load cells that are inoperative will be repaired or supplemented with prestress rams that have been modified with 20 psi division gages to measure tendon forces. Prestress rams were used at Palisades and performed satisfactory. Results of measured forces can then be compared with those predicted.

The taut wire system consists of linear potentiometers (infinite resolution type) as the transducer element. Movement of the linear potentiometers will
16. be actuated by invar wires attached at one end to the point of measurement and at the other end to a reference point. Approximately 35 linear potentiometers will be used to measure building deformations during the structural test.

Units 2 and 3 Reactor Buildings are instrumented with the taut wire system for measuring building deformations as described above for Unit 1. Displacement measurements are made at the following locations:

Dome	- Four points
Cylinder Wall	- Seven elevations at approximately 20 foot intervals at a buttress section and a wall section
20. Equipment Hatch	- Nine points with six of the points on the horizontal centerline and three of the points on the vertical centerline above the hatch
Vertical	- Two points

The above locations were selected so that deformation measurements could be compared with Unit 1 measurements.

Concrete crack patterns are recorded at the base-wall intersection, cylinder wall mid-height, springline, equipment hatch opening, buttress-cylinder wall intersection, cylinder wall-ring girder intersection, and top of ring girder. Each inspection area consisted of approximately 40 square feet. Cracks that exceed 0.01 inch in width are mapped.

5.6.1.3 Initial Leakage Tests

Following completion of the Reactor Buildings and prior to the hot functional tests and fueling of the reactors, integrated leakage rate tests will be per-

Rev. 10. 8/28/70
(Carry-over)
Rev. 15. 12/30/70
Rev. 16. 7/30/71
Rev. 20. 5/25/72

formed on the containment systems. One test will be performed at or above the maximum calculated peak accident pressure. A second test will be performed at a pressure of not less than 50 percent of maximum calculated peak accident pressure.

The absolute pressure-temperature and/or the reference vessel method will be used for these tests. The objectives of these tests are:

a. To determine the initial integrated leakage rate for comparison with the design leakage rate.

b. To establish representative leakage characteristics of the containment system to permit retesting at reduced pressures.

c. To establish a performance history summary of the integrated leakage rate tests.

d. To establish a test method and the equipment to be used for subsequent retesting.

The leakage rate will be measured by integrating the leakage rate for a period
33. of not less than 10 hours. This integrated leakage will be verified by the "pump-back" method and/or introduction of a known leak rate. The necessary instrumentation will be installed to provide accurate data for calculating the leakage rate. It will be demonstrated that the total Reactor Building leakage rate to the environment will maintain public exposure below 10CFR100 limits in the event of an accident.

5.6.2 POSTOPERATIONAL SURVEILLANCE

5.6.2.1 Leakage Monitoring

A program of testing and surveillance of each of the three duplicate Reactor Buildings has been developed to provide assurance, during service, of the capability of each containment system to perform its intended safety function. This program consists of tests defined as follows:

Overall integrated leak rate tests of the Reactor Buildings and systems which under post accident conditions become an extension of the containment boundary.

Local leak detection tests of components having resilient seals, gaskets, or sealant compounds that penetrate or seal the boundary of the containment system. Components included in this category are:

a. Personnel Hatches
b. Emergency Hatches
c. Equipment Hatches
d. Reactor Building Purge Penetrations
e. Fuel Transfer Tube Covers
f. Electrical Penetrations
g. Reactor Building Atmosphere Sampling Penetrations

Local leak detection and operability tests of containment isolation valves in systems that vent directly to the Reactor Building atmosphere or the Reactor Coolant system that must close upon receiving an isolation signal and seal the containment under accident conditions. Valves and containment penetrations which during operation are normally valved closed and which if open could be immediately closed, will not require testing.

Operability tests of engineered safeguards systems which under post accident conditions are relied upon to limit or reduce leakage from the containment. Included in these tests are:

a. Reactor Building Spray Systems
b. Reactor Building Penetration Room Ventilation Systems
c. Reactor Building Cooling Systems
d. Reactor Building Isolation Valves not covered above.

Following the integrated leakage rate tests, performed as a part of the pre-operational testing, subsequent tests will be performed at a pressure of 50 percent of the maximum calculated peak accident pressure or greater. The tests will be performed on a schedule based on the following considerations:

a. There are three Reactor Buildings each having the same design. Information pertaining to deterioration in performance obtained in the testing of one Reactor Building is therefore applicable to the other Reactor Buildings.

b. Local leak detection tests will be performed on a more frequent basis than the integrated tests to detect and correct excessive leakage at containment penetrations. Where feasible, these tests will be performed during operation; otherwise, they will be performed during refueling outages and/or major maintenance outages. These tests will be performed at or above the maximum calculated peak accident pressure.

c. The engineered safeguards tests will also be performed at more frequent intervals than the integrated leak rate tests to verify the functional capability of these systems which are relied upon to limit or reduce leakage from the containment buildings in the case their service is required. These tests will be performed during outages for refueling and/or major maintenance outages.

The schedule of testing, type of test, and components to be tested are as follows:

Integrated Leak Rate Tests

Integrated leak rate tests shall be performed as follows:

a. Each reactor building shall be tested at the calculated peak
18. accident pressure of 59 psig and at one-half this pressure prior to the initial fuel loading.

b. After the initial preoperational leakage rate test, two integrated leakage rate tests shall be performed on each reactor building at
8. approximately equal intervals between each major shutdown for in-service inspection to be performed at 10 year intervals. In addi-
20 tion, an integrated test shall be performed at each 10 year interval, coinciding with the inservice inspection shutdown. The test shall coincide with a shutdown for major fuel reloading. These tests shall be conducted at or above one-half peak accident pressure (P_t).

Local Leak Detection and Operability Tests (Resilient Seals)

18. Local leak detection and operability tests shall be performed as required by the Technical Specifications.

18.

Engineered Safeguards Tests

The Reactor Building spray, penetration room ventilation, Reactor Building cooling systems, and the Reactor Building isolation valves will be tested during refueling or extended maintenance outages to provide approximately annual tests. These tests will include:

a. Reactor Building Cooling and Reactor Building Penetration Room Ventilation Systems

   Each of these systems is operated periodically during normal operating periods to maintain satisfactory temperatures within the Reactor Buildings and penetration rooms respectively. This normal operator initiated operation of these systems provides verification of the operability. In addition to this normal operation, an annual test of these systems in the engineered safeguards mode will also be performed. This test will be initiated by inserting a simulated engineered safeguards signal as would occur during an accident situation. Verification of the proper operation of the components of these systems will be determined and a record of the test results made a part of the permanent plant records.

b. Reactor Building Spray System

   The Reactor Building Spray System will be tested in the same manner as the systems above with the exception that the Reactor Building spray headers will be isolated to prevent spray water from entering the spray headers. A special test connection is provided ahead of the Reactor Building isolation valves so that the portion of the system outside the Reactor Building will be in normal operation. When the test of that portion of the system outside the Reactor Building has been completed, compressed air will be blown through each of the spray headers in the Reactor Building through special test connections to verify that spray water would be directed into the Reactor Building under accident conditions. Proper operation of the various components of this system will be verified and a record of the test results made a part of the plant records.

c. Reactor Building Isolation Valves

   Proper operability of the Reactor Building isolation valves not covered in the other tests will be verified by inserting a simulated engineered safeguards signal to initiate operation of these valves.

5.6.2.2 Surveillance of Structural Integrity

Provisions have been made for an in-service surveillance program, throughout the life of the plant, intended to provide sufficient in-service historical evidence to maintain confidence that the integrity of the Reactor Building is being preserved. The requirements of this program are detailed in the Technical Specifications.

Earthquake instrumentation being provided is a strong-motion accelerograph designated RMT-280 and manufactured by Geotech Corporation of Garland, Texas. This is a completely self-contained magnetic tape recorder that will provide acceleration data in all three planes from torsion-type accelerometers having a natural frequency of 12 to 20 Hertz. A reference frequency and timing track is also recorded. The recorder is started by an actuating pendulum having a normal setting of 0.01 g and an external contact is available to indicate the recorder being in operation. The recorder continues to run until seven seconds after the pendulum contact opens. Recording time available is one hour.

Data available are as follows:

Photographic reproduction of the original accelerogram
Digitized accelerogram on punched cards or magnetic tape
Acceleration, velocity and displacement analog plots
Response spectra plots of pseudo-relative velocity

Also being provided is an additional actuating pendulum set to close contacts at 0.05 g to alarm when design conditions occur.

The equipment will be located in the tendon access gallery of Unit 1 Reactor Building. Also, a second strong motion accelerograph will be located directly above at elevation 797' + 6" in Unit 1 Reactor Building. Orientation of the sensors of the two accelerographs are identical.

Peak recording accelerometers are also installed at various locations within Unit 1 Reactor Building as follows:

1. Adjacent to the strong motion accelerograph located in Tendon Access Gallery.
2. Support of the pressurizer vessel.
3. Support of Core Flood Tank 1A.
4. On the main steam line pipe hanger.
5. On the feedwater line pipe hanger.
6. On the core flood injection line pipe hanger.

The major Class I structures, Reactor Building and Auxiliary Buildings, will be founded on a common rock foundation and will have similar base motions. The dynamic structural properties and responses of these structures are generated using similar assumptions and analytical techniques. Therefore, the response of these structures can be determined based upon the instrumentation in one structure.

Top of soil (free field) responses will not provide useful analytical data for the evaluation of major Class I structures founded on rock. Therefore, it is felt that free field instrumentation will not contribute to the evaluation of these structures.

In the event of an earthquake, the data will be analyzed to determine the magnitude of the earthquake. If the design earthquake is exceeded, the units would be shut down and structures, systems, and equipment thoroughly investigated. Responses from instruments located on selected structures, systems and components will be compared to calculated responses for those structures, systems and components at the respective location when subjected to the same base response.

The recorded seismic data will be used for comparison and verification of seismic analysis assumptions, damping characteristics and the analytical model used for the plant seismic design.

5.7 OTHER STRUCTURES

5.7.1 AUXILIARY BUILDING

5.7.1.1 General Description

The building was constructed on a reinforced concrete mat foundation. Below grade, the building consists of reinforced concrete walls and slabs. Above grade, the building consists principally of reinforced concrete columns, beams, and slabs.

The following facilities related to the nuclear steam supply system are located in the Auxiliary Building:

a. New and spent fuel handling, storage, and shipment
b. Control room
c. Waste disposal system
d. Chemical addition and sampling system
e. Component cooling system
f. Reactor Building spray system
g. High and low pressure injection system
h. Spent fuel cooling system
i. Electrical distribution system

Figures 1-2 through 1-9 are plans and elevations showing the Auxiliary Building.

5.7.1.2 Design

The areas of the building housing the above facilities have been designed for the loads and conditions as shown in Table 5-5 with maximum allowable stresses as follows:

Loading Condition	Maximum Allowable Stress
A	Stresses in accordance with ACI and AISC Codes
B, D	$fc = 0.85\ f'_c$ for Flexure
	$fc = 0.70\ f'_c$ for tied compression members
1.	Shear $= 1.1\ \sqrt{f'_c} \times 1.33$ for beams with no web reinforcing
	$fs = 0.90fy$ for Flexure
	$fs = 0.85fy$ for reinforcing steel with lap or mechanical splices
	$fs = 0.90fy$ for web reinforcing
	Bond $= \dfrac{3.4\ \sqrt{f'_c}}{D} \times 1.33$ for top bars
	$= \dfrac{4.8\ \sqrt{f'_c}}{D} \times 1.33$ other than top bars
C, E	Anlayzed on basis of Reference 7

Rev. 1. 9/15/69

TABLE 5-5

Auxiliary Building Loads and Conditions

AREA	CONDITIONS	
Control Room	A,B,C,D,E	Blow out panels designed to relieve 3 psi differential pressure
Cable Room	A,B,C,D,E	
Electrical Equipment Room	A,B,C,D,E	
Spent Fuel Pool	A,B,C,D,E	Blow out panels designed to relieve 3 psi differential pressure
Spent Fuel Storage Racks	A,D	Inherently resistant to wind loads
Spent Fuel Handling Crane	A,D,E	Inherently resistant to wind loads. Hold down device provided
Penetration Room Frames	A,B,D	Physical separation provided for missile protection
Cable Shaft	A,B,C,D,E	
Elevator Steel Shaft	A,D	
Main Steam Pipe Supports	A,B,D	
Hot Machine Shop	A,D	
Balance of Auxiliary Building	A,B,D	Frame designed for B, but not external walls above grade. Areas below grade are inherently protected against missiles in C and E.

A = All normal dead, equipment, live, and wind loads due to 95 mph wind.

B = Normal dead and equipment loads plus tornado wind load due to 300 mph wind.

C = Tornado missiles of (1) 8 in. diameter x 12 ft. long piece of wood, 200 pounds, 250 mph, and (2) 2,000 pound automobile, 100 mph, 20 sq. ft. impact area, for 25 ft. above grade.

D = Normal dead and equipment loads plus maximum hypothetical earthquake loads.

E = Turbine-generator missile, 5,944 pounds, 502 fps, kinetic energy of 23.25 x 10^6 ft.-lbs., side on impact area of 8.368 sq. ft. and end on impact area of 3.657 sq. ft.

The Spent Fuel Pool Walls were analyzed for thermal loads in accordance with methods presented in ACI 505. The exterior wall temperature was assumed to be 60 F for areas enclosed by the Auxiliary Building and 0 F for exposed areas.

Under normal conditions, the interior wall temperature was 150 F and the maximum calculated thermal stress was 996 psi for concrete and 11,410 psi for reinforcing steel.

After prolonged outage of the cooling system, the interior wall temperature could reach 212 F and the maximum calculated thermal stress was 1681 psi for concrete and 25,600 psi for reinforcing steel. Reinforcing steel conforming with ASTM A516, Grade 60, was used.

A minimum of 0.30 percent reinforcing was used in the spent fuel pool walls to control concrete cracking. Also, a 1/4 inch thick steel liner was used on the inside face of the pool for leak tightness.

The fuel storage racks support the fuel elements at the top. The bottom of the fuel elements is supported by attachments to the fuel pool slab.

The racks were designed for seismic loadings by considering the peak spectral acceleration associated with a single mass system for one percent damping. In addition, the drag forces associated with maximum velocities of the water relative to the racks was considered acting simultaneous with the peak acceleration.

The Spent Fuel Pool Slab was designed for the postulated cask drop accident. Fill concrete was placed from sound rock to the bottom of the fuel pool slab in the area covered by the cask crane to prevent the shearing of a large plug from the pool slab in the event the cask was accidently dropped.

10. The height of the cask drop is the maximum vertical travel of the crane and is 45 feet of which 40 feet is through water. The penetration of the cask into the slab is calculated to be 1.75 inches. No credit was taken for the water resistance nor the resistance of the linear plate.

The geometry and strength characteristics (edge radius, base material, etc.) of the cask will be specified to assure that the calculated penetration can occur without rupturing the liner plate. The analysis considers local concrete crushing and liner yielding; however, the strains in the liner plate will not exceed ultimate.

5.7.2 TURBINE BUILDING

5.7.2.1 General Description

The building was constructed of reinforced concrete below grade consisting of substructure walls and a mat foundation. Above grade, the building consists of structural steel with metal siding.

5.7.2.2 Design

Prior to design, the Principal Civil Engineer issued a Civil Design Memorandum that stated the basic design criteria.

Based on the basic criteria and general arrangement drawings of the Turbine Building, design studies were made to determine building dimensions, type of steel, member sizes and shapes.

Transverse Analysis

Each bent consisted of the three main crane columns, on lines D, J and M, the roof girders, the columns of lines K and L and the operating and mezzanine floor framing. Where continuity of framing was not interrupted by the turbine-generator support, the short columns and operating and mezzaine floor framing were included as a part of the rigid frame. See Figure 5-22 for typical Turbine Building Cross-Section.

A computer program, "Stress", was used in the analysis of the bents.

The loadings were applied as follows:

Dead Loads - Roof - 50 psf, reduced to 25 psf when the type of roof construction was finalized.
Floors - Grating Areas - 20 psf.
Concrete Areas - Operating Floor - 11-1/2" masonry -
4. 170 psf.
- Mezzanine Floor - 8" masonry - 115 psf.
- Upper Surge Tank Floor - 4" masonry - 65 psf plus tank at normal operating condition,
Crane Columns and Girders - Calculated weights.

Live Loads - Roof - 50 psf.
Grating Areas - 100 psf.
Operating Floor - Turbine Bay - 600 psf.
Heater Bay - 400 psf.
Mezzanine Floor - 250 psf.
Cranes - 180 Ton and 80 Ton Cranes fully loaded, lifted load and lateral force arranged to produce maximum stresses. The lateral forces were reduced to 15 percent of the sum of the weights of the lifted load and the crane trolleys.

Wind Load - 30 psf.

Seismic Loading No. 1 - (Load Combinations)
Critical Damping - 2%
Maximum Ground Motion Acceleration - 5% of gravity
Maximum Acceleration for Design - 12% of gravity (This is the maximum value of the acceleration response curve for 2% damping.)
Loadings - Roof - 50 psf, reduced to 25 psf when the type of roof construction was finalized.
Operating Floor - dead load of floor plus equipment load. (Equipment load estimated at 250 psf.)
Mezzanine Floor - dead load of floor plus equipment load. (Equipment load estimated at 150 psf.)
Upper Surge Tank and Floor - 65 psf plus tank at normal operating condition.
Crane - 180 Ton Crane, fully loaded, at center of bay.
Crane Columns and Girders - Calculated weights.

Seismic Loading No. 2 - (Load Combinations)
Critical Damping = 2%
Maximum Ground Motion Acceleration - 10% of gravity
Maximum Acceleration for Design - 22% of gravity (This is the maximum value of the acceleration response curve for 2% damping.)
Loadings - Roof - 25 psf.
Operating Floor - dead load of floor plus equipment load. (Equipment load estimated at 200 psf.)
Mezzanine Floor - dead load of floor plus equipment load. (Equipment load estimated at 125 psf.)
Upper Surge Tank and Floor - 65 psf plus tank at normal operating condition.
Cranes - 180 Ton Crane and 80 Ton Crane at rest and in unloaded condition.
Crane Columns and Girders - Calculated weights.

Seismic Loading No. 2 was introduced approximately six months after the building was analyzed for Seismic Loading No. 1. With more complete information, it was apparent that the equipment loads assumed for the Operating and Mezzanine Floors were too conservative. Therefore, the equipment loads were reduced for the analysis for Seismic Loading No. 2.

Longitudinal Analysis

Column lines B, D, J and M were braced with diagonal members. For lines D, J and M, this bracing took the form of two members for each brace with batten plates and angle lacing tying them together.

The loadings were applied as follows:

Wind Load - 30 psf.
Crane Load - 10% of Maximum wheel load.

Seismic Loading No. 1 - Same as Seismic Loading No. 1 for Transverse Analysis with the following exceptions.

Loadings - Operating Floor - Equipment load estimated at 130 psf.
Mezzanine Floor - Equipment load estimated at 110 psf.

Seismic Loading No. 2 - Same as Seismic Loading No. 2 for Transverse Analysis with the following exceptions:

Loadings - Operating Floor - Equipment load estimated at 130 psf.
Mezzanine Floor - Equipment load estimated at 110 psf.

Loading Combinations and Factors

$$S = 1.0\ D + 1.0\ L$$
$$1.33S = 1.0\ D + 1.0\ L + 1.0\ W$$
$$1.33S = 1.0\ D + 1.0\ E$$
$$1.64S = 1.0\ D + 1.0\ E'$$

S = Allowable stress due to normal loading - from AISC specifications.
D = Dead Loads (Equipment loads included in the case of seismic loadings)
L = Live Loads
W = Wind Loads
E = Loads from Seismic Loading No. 1
E'= Loads from Seismic Loading No. 2

A dynamic seismic analysis of the building was performed consisting of a three mass system. Seismic loading conditions for Seismic Loading No. 2 were applied. Maximum accelerations consisting of absolute sums for the three modes were .47 g for the roof, .20 g for the operating floor and .16 g for the mezzanine floor. It is considered that the absolute sum is a conservative value. The structure was analyzed using these accelerations and stresses were found to be within design criteria. Typical stress values, shown as percentage of allowable, are as follows:

Location	Normal Load	Seismic Load #2 (Static Analysis)	Seismic Load (Dynamic Analysis)
Col. D at basement	83%	68%	44%
Col. D at roof	94%	36%	28%
Col. J below oper. floor	81%	51%	67%
Col. J above oper. floor	78%	37%	48%
Col. J at roof	88%	54%	59%
Col. M below oper. floor	89%	72%	75%
Col. M at oper. floor	84%	36%	47%
Col. M at roof	90%	56%	48%

5.7.3 KEOWEE STRUCTURES

A review of the Keowee structural design, including seismic loadings, has been made as follows (all structures utilize 3000 psi concrete, 40,000 psi reinforcing steel and A36 structural steel).

5.7.3.1 Powerhouse

A typical reinforced concrete frame was investigated for the following loading conditions using a static type analysis:

a. Dead load plus live load (1000 lbs per square foot) using allowable stresses in accordance with ACI Code. The maximum calculated stresses were f_s = 18,590 psi and f_c = 1122 psi.

b. Dead load plus live load (1000 lbs per square foot) plus seismic load equal to .10g times the dead load. The maximum calculated stresses were f_s = 19,120 psi and f_c = 1189 psi. Allowable stresses were $f_s = .9\ f_y$ = 36,000 psi and $f_c = .85\ f'c$ = 2550 psi.

c. Dead load plus live load (1000 lbs per square foot) plus seismic load equal to .20g times the dead load. The maximum calculated stresses were f_s = 19,700 psi and f_c = 1229 psi.

The large live loading of 1000 lbs per square foot was included to allow for heavy equipment loads expected during construction and maintenance. Therefore, to be conservative, the 1000 lbs per square foot was included to b and c above but with seismic loadings added as a function of dead load only.

5.7.3.2 Spillway

A typical spillway pier was investigated for the following loading conditions:

a. Dead load plus hydrostatic load with allowable stresses in accordance with ACI Code. The maximum calculated stresses were f_s = 0 and f_c = 61.7 psi.

b. Dead load plus hydrostatic load plus seismic load equal to .10 times dead load. The maximum calculated stresses were f_s = 7760 psi and f_c = 173 psi. The allowable stresses were $f_s = .9\ f_y$ = 36,000 psi and $f_c = .85\ f'c^*$ = 3400 psi.

c. Same as b except seismic load equal to .20 times dead load. The maximum calculated stresses were f_s = 16,350 psi and f_c = 227 psi.

* f'c = 4000 psi in piers.

In addition, the taintor gate thrust girder was investigated for the following loading conditions:

a. Dead load plus hydrostatic load with allowable stresses in accordance with AISC Code. The maximum calculated stress was f_s = 23,300 psi.

b. Dead load plus hydrostatic load plus seismic load equal to .10 times dead load with allowable stress = $.9\ f_y$ = 32,500 psi. The maximum calculated stress was f_s = 25,000 psi.

c. Same as b except seismic load equal to .20 times dead load. The maximum calculated stress was f_s = 28,800 psi.

5.7.3.3 Service Bay Substructure

The Service Bay substructure contains the Control Room, Cable Room, Equipment Room and Battery Room areas. The substructure was investigated for the following loading conditions:

a. Dead load plus live load with allowable stresses in accordance with ACI Code. The maximum calculated stresses were f_s = 19,700 psi and f_c = 1160 psi.

b. Dead load plus live load plus seismic load equal to .15 times the combined dead-live load. The allowable stresses were f_s = .9 f_y = 36,000 psi and f_c = .85 $f'c$ = 2550 psi. The maximum calculated stresses were f_s = 24,000 psi and f_c = 1410 psi. It is apparent that the seismic loads could be substantially increased with resulting stresses being well below those allowable.

5.7.3.4 Breaker Vault

The Breaker Vault is located on the Operating Floor level of the Keowee Powerhouse and was designed primarily to afford tornado protection for electrical equipment. The controlling case was dead load plus equipment loads plus tornado wind and missile. Resulting stresses for this case were f_s = 38,000 psi and f_c = 2190 psi.

These compare to the allowable f_s = .9 f_y = 36,000 psi and f_c = .85 $f'c$ = 2550 psi. The actual steel stresses were about 5-1/2 percent over the allowable but 5-1/2 percent below the guaranteed minimum yield and are considered satisfactory for this severe loading combination.

A second case considered dead load plus seismic loads equal to .15 times the combined dead-live loads plus normal wind. By inspection, it was found that this would result in substantially lower stresses than the loading combination above. Therefore, a detailed design check was not made.

5.7.3.5 Intake Structure

Three design cases were considered:

a. Construction condition (dead load plus wind load) with no water and allowable stresses being within the ACI and AISC Code. The resulting stresses were extremely low.

b. Cylinder gate closed and structure unwatered. Allowable stresses were based on ACI and AISC Code. Calculated stresses were found to be well within the code limits.

c. The third case considered the cylinder gate open, dead loads and seismic loads equal to .15 times the dead load. Maximum calculated stresses were f_s = 39,700 psi and f_c = 2050 psi.

The resulting steel stresses are marginally below the guaranteed minimum yield and are considered satisfactory for the severe loading combination.

5.8 REFERENCES

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(2) Levy, Samuel, McPherson, A. E., and Smith, F. C., "Reinforcement of a Small Circular Hole in a Plane Sheet Under Tension," Journal of Applied Mechanics, June 1948.

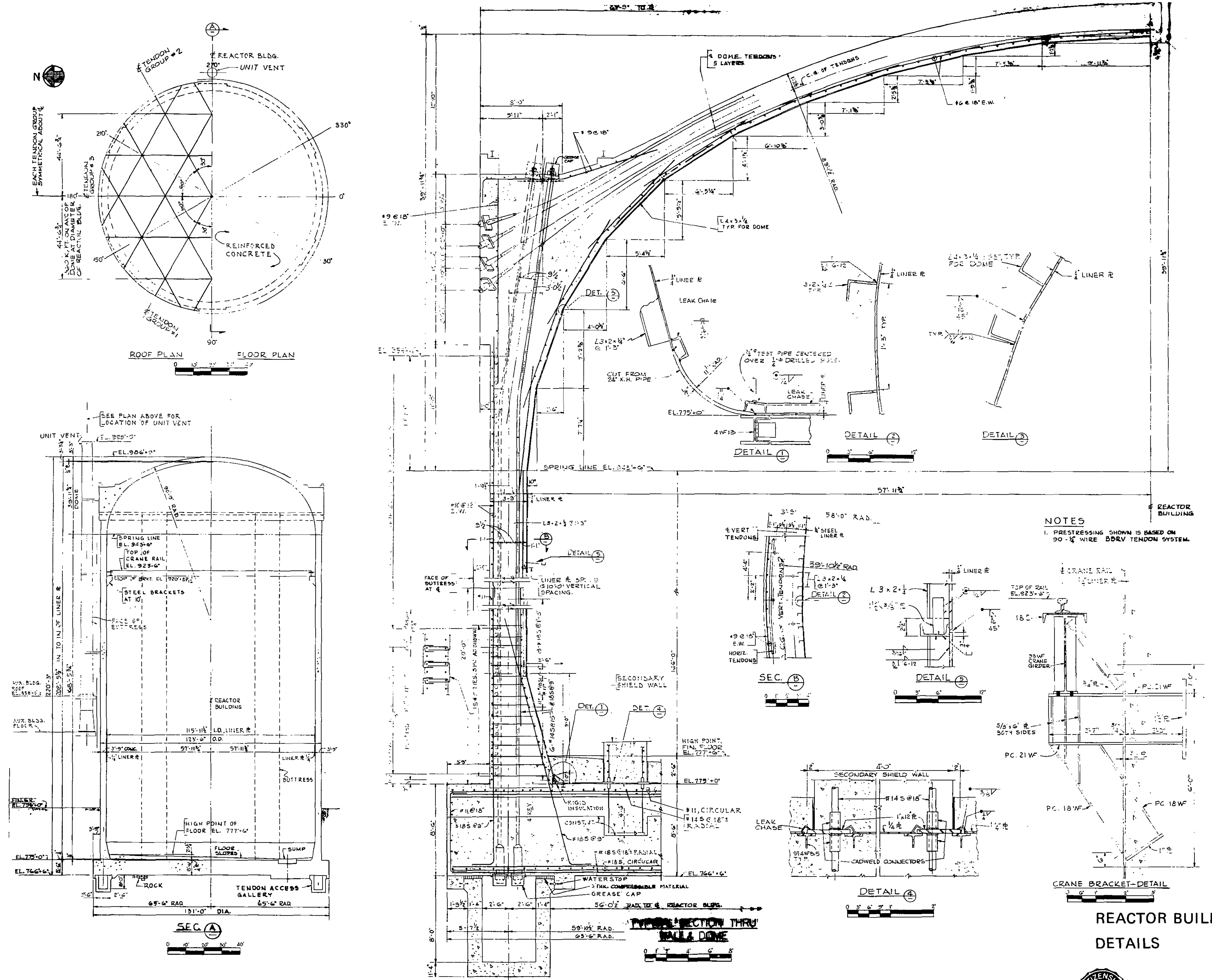
(3) Wichman, K. R., Hopper, A. G., and Mershon, J. L., "Local Stress in Spherical and Cylindrical Shells due to External Loadings," Welding Research Council Bulletin No. 107, August 1965.

(4) HTGR and Laboratory Staff, Prestressed Concrete Reactor Vessel, Model 1, GA7097.

(5) Advance HTGR Staff, Prestressed Concrete Reactor Vessel, Model 2, GA7150.

(6) Hardingham, R. P., Parker, J. V., and Spruce, T. W., Liner Design and Development for the Oldbury Vessels, London Conference on Prestressed Concrete Pressure Vessels, Group J, Paper 56.

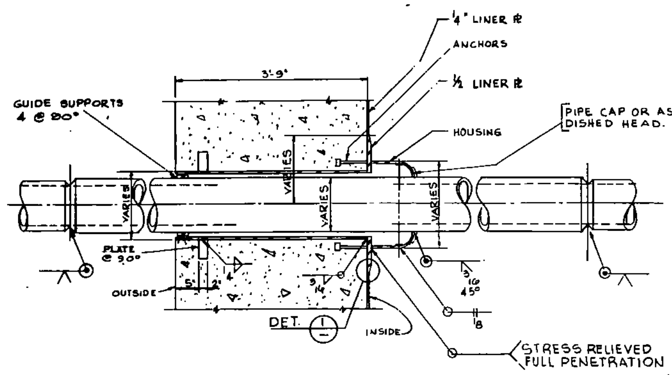
(7) Amirikian, A., Design of Protective Structures, Bureau of Yards and Docks, Department of the Navy, NAVDOCKS P-51, 1950.



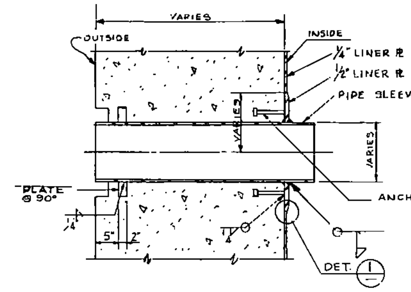
REACTOR BUILDING TYPICAL DETAILS

OCONEE NUCLEAR STATION

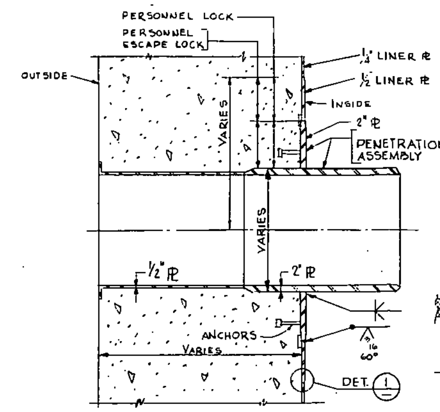
Figure 5 - 1



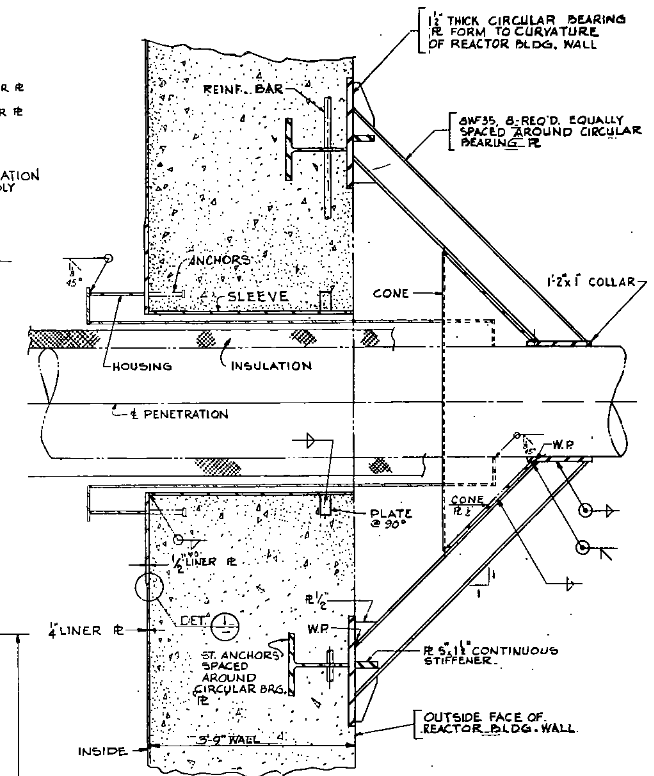
TYPE - I
TYP. COLD PIPE PENETRATION
LESS THAN 150° F
NO SCALE



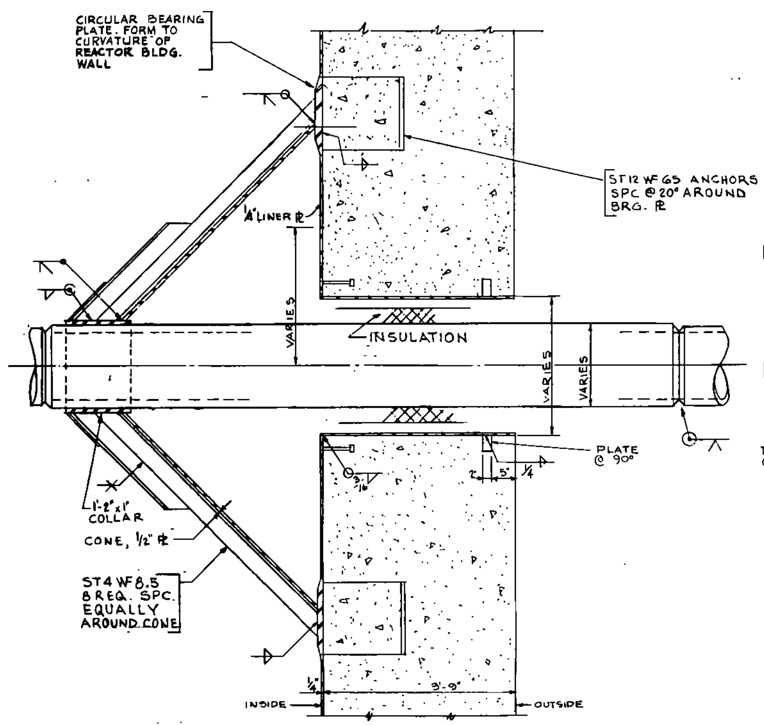
TYPE - III
TYP. ELECTRICAL PENETRATION
NO SCALE



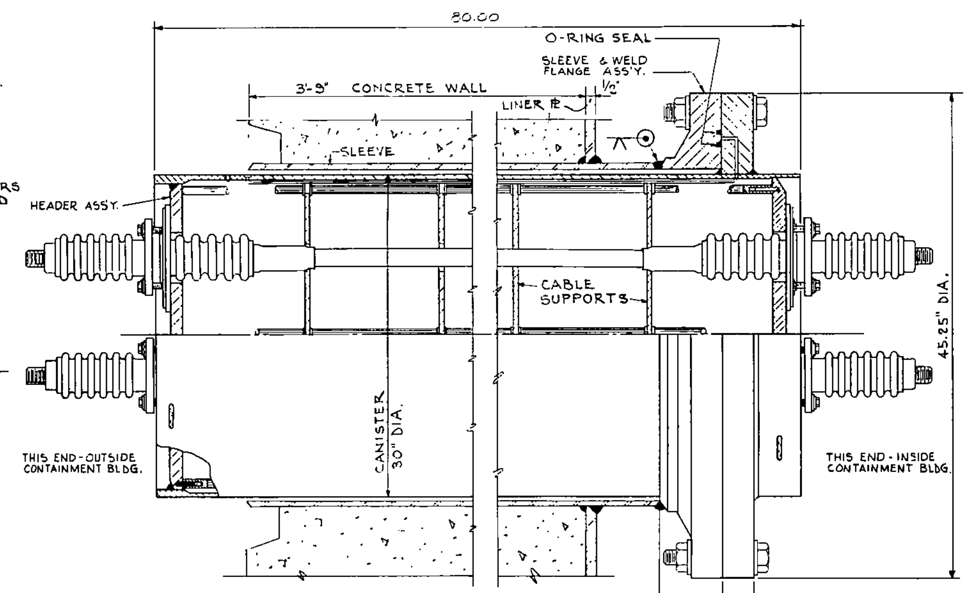
TYPE - IV
LOCK PENETRATION
NO SCALE



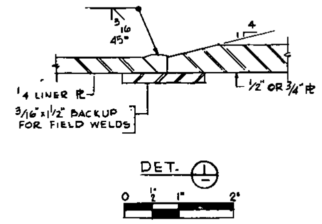
TYPICAL MAIN FEEDWATER LINE PENETRATION
NO SCALE

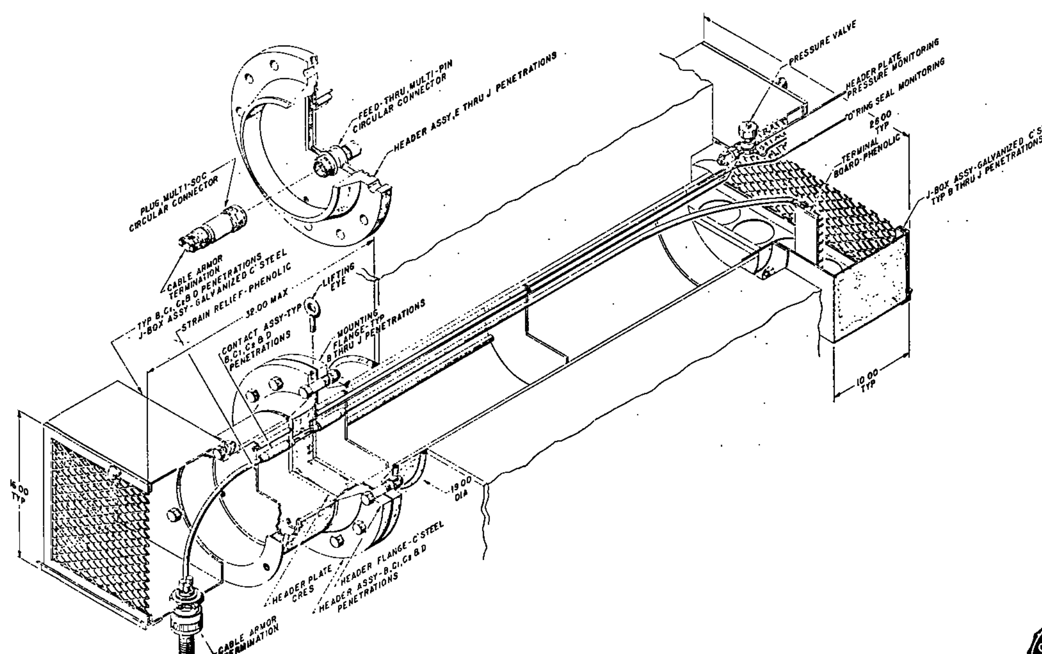


TYPICAL MAIN STEAM LINE PENETRATION
NO SCALE



TYPICAL MEDIUM VOLTAGE POWER 6900 VOLT PENETRATION



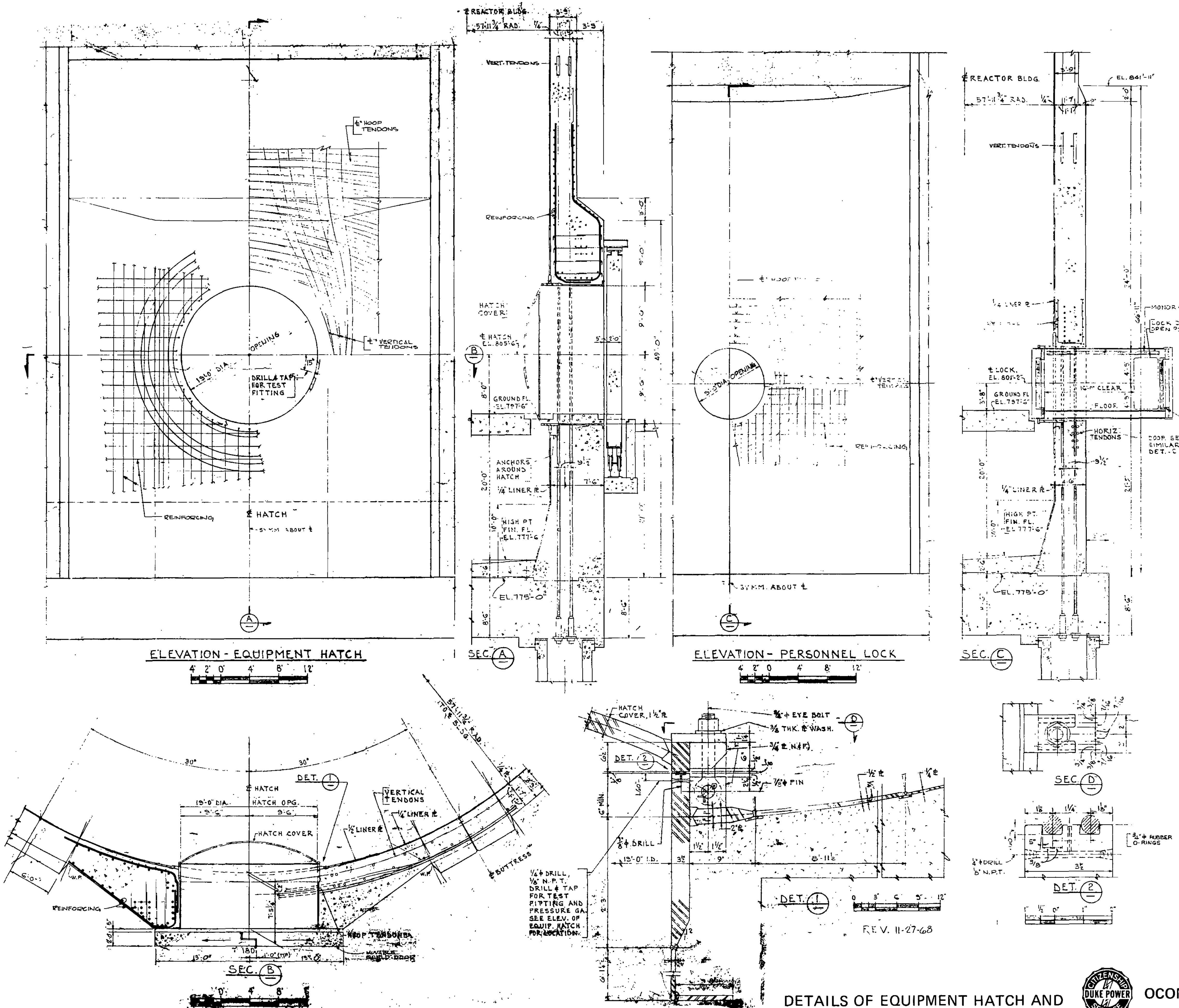


TYPICAL LOW VOLTAGE POWER CONTROL & INSTRUMENTATION PENETRATION
NO SCALE

TYPICAL ELECTRICAL AND PIPING PENETRATIONS

DUKE POWER

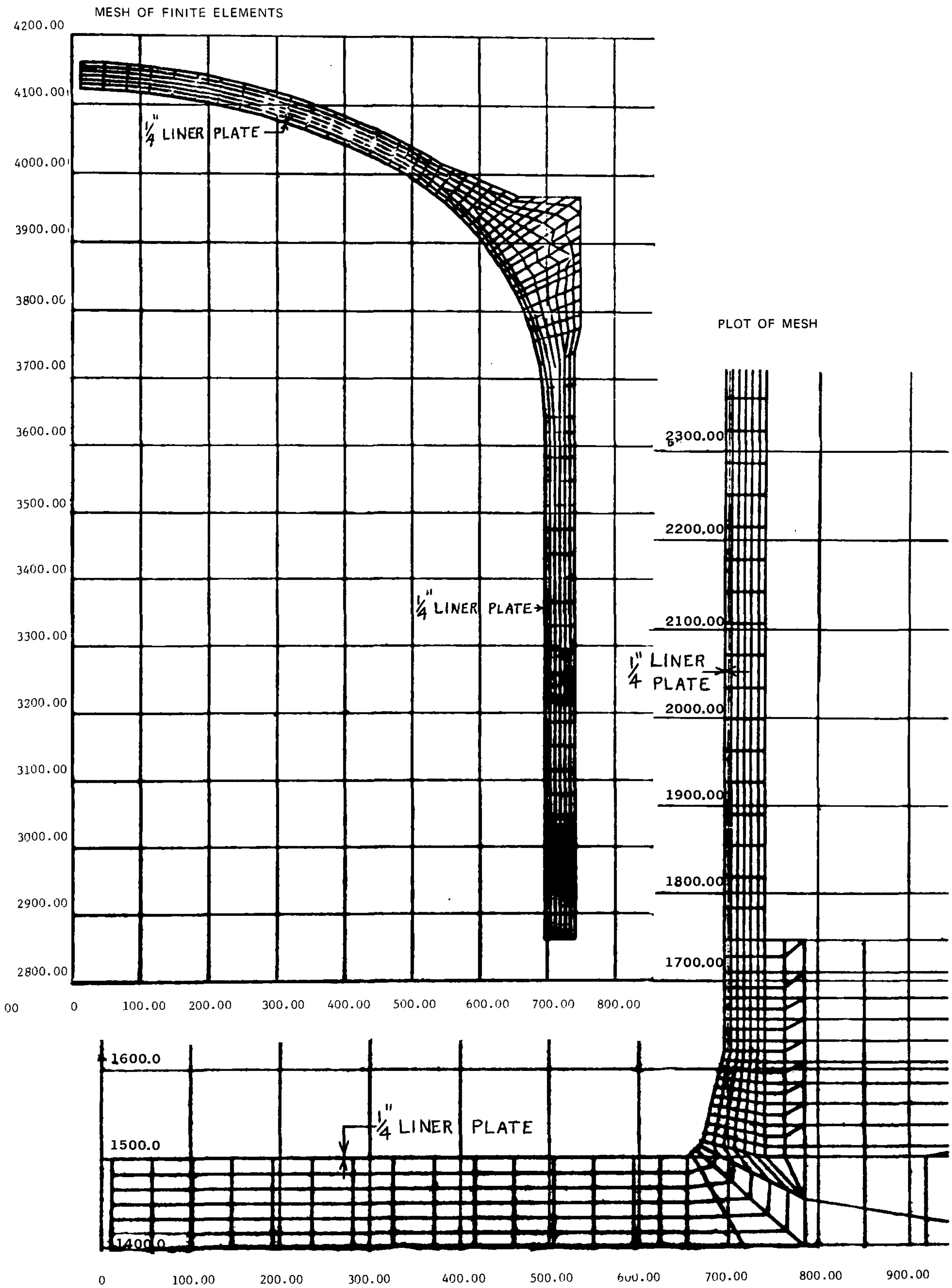
OCONEE NUCLEAR STATION
Figure 5 - 2
Rev. 6 6/22/70



DETAILS OF EQUIPMENT HATCH AND PERSONNEL HATCH

OCONEE NUCLEAR STATION

Figure 5 - 3



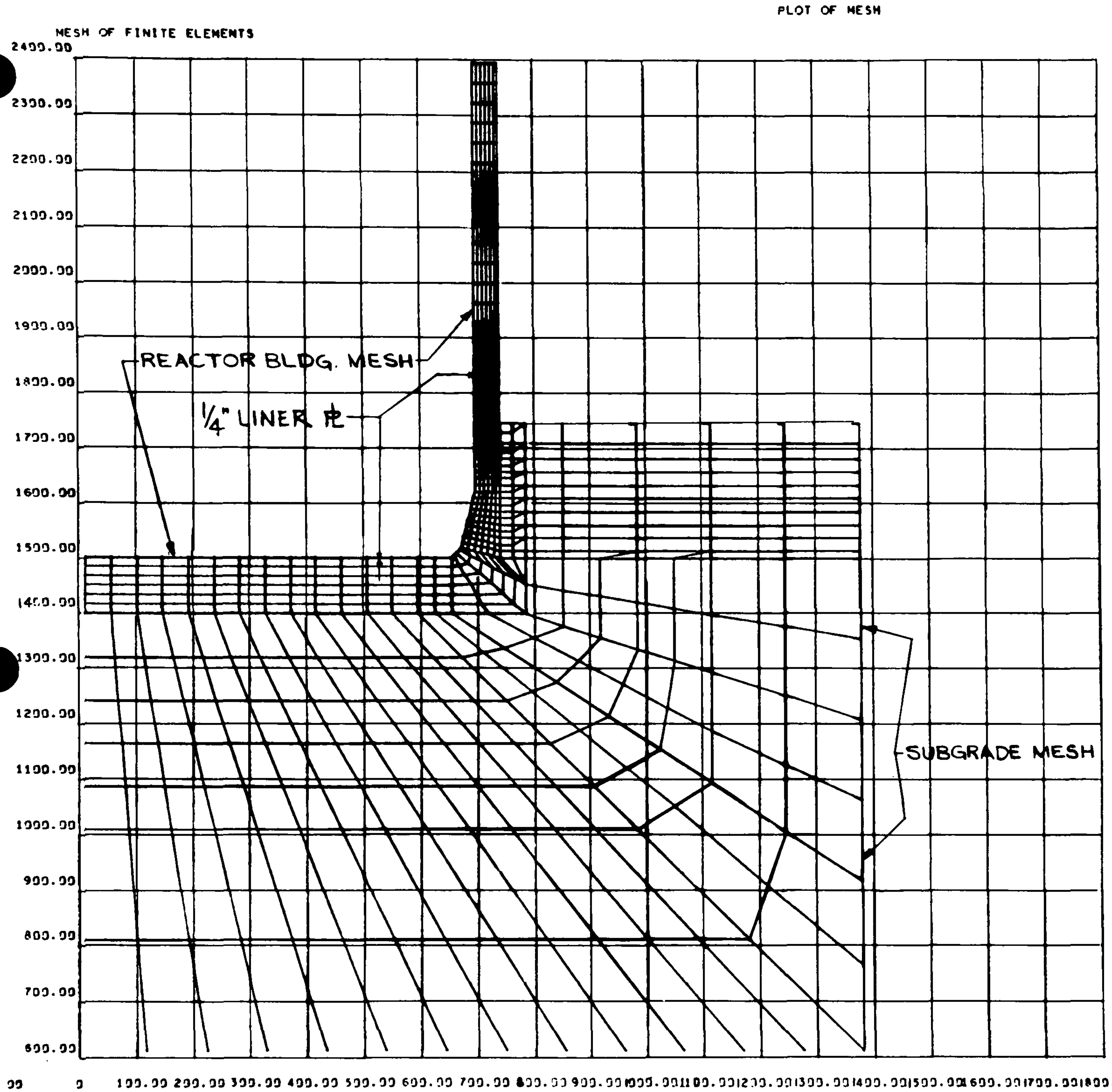
REACTOR BUILDING FINITE ELEMENT MESH

OCONEE NUCLEAR STATION

Figure 5 - 4

(Sheet 1 of 2)

Rev. 4 4/20/70



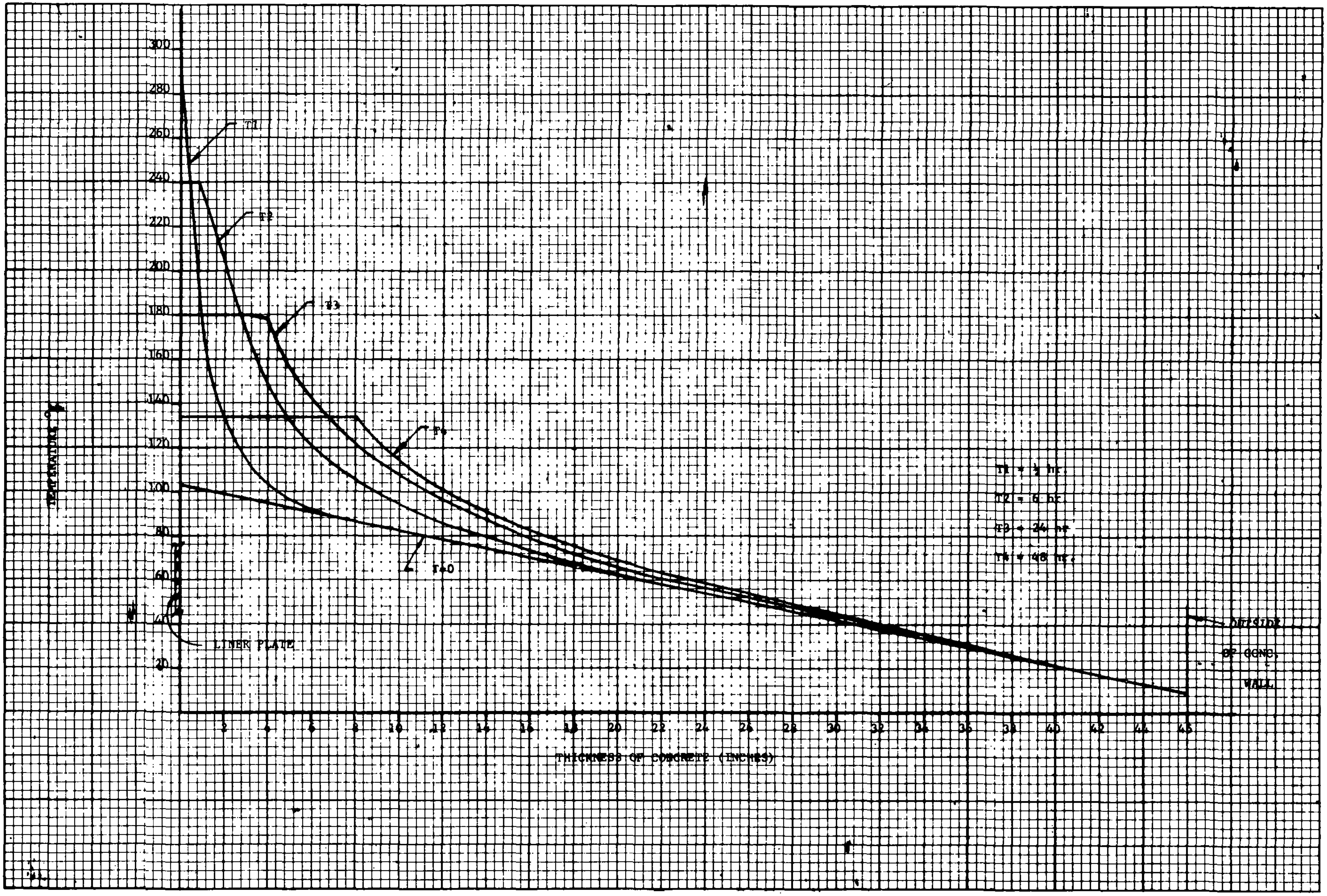
REACTOR BUILDING FINITE ELEMENT MESH

OCONEE NUCLEAR STATION

Figure 5 - 4

(Sheet 2 of 2)

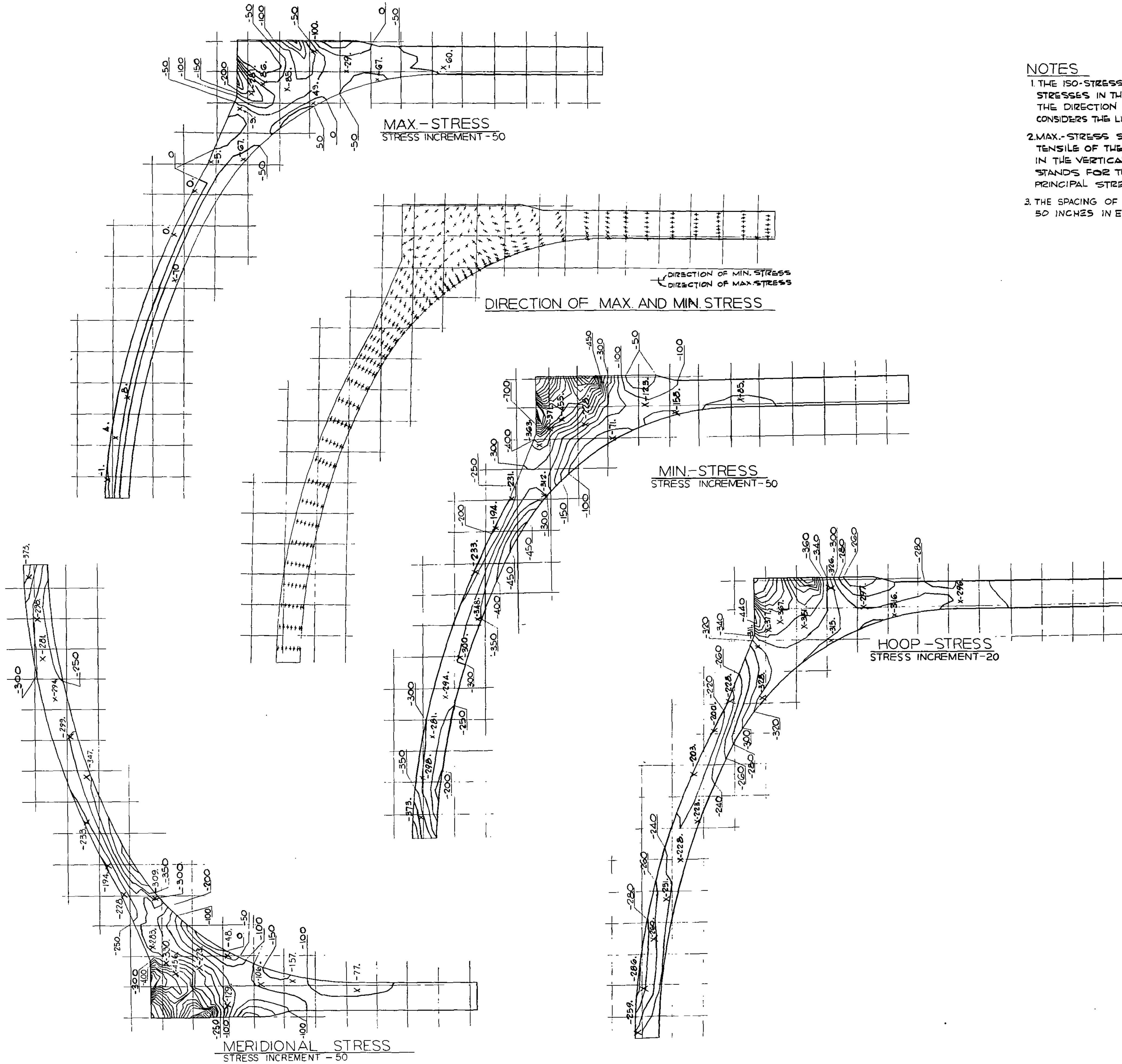
(New) Rev. 4 4/20/70



REACTOR BUILDING THERMAL GRADIENT

OCONEE NUCLEAR STATION

Figure 5 - 5



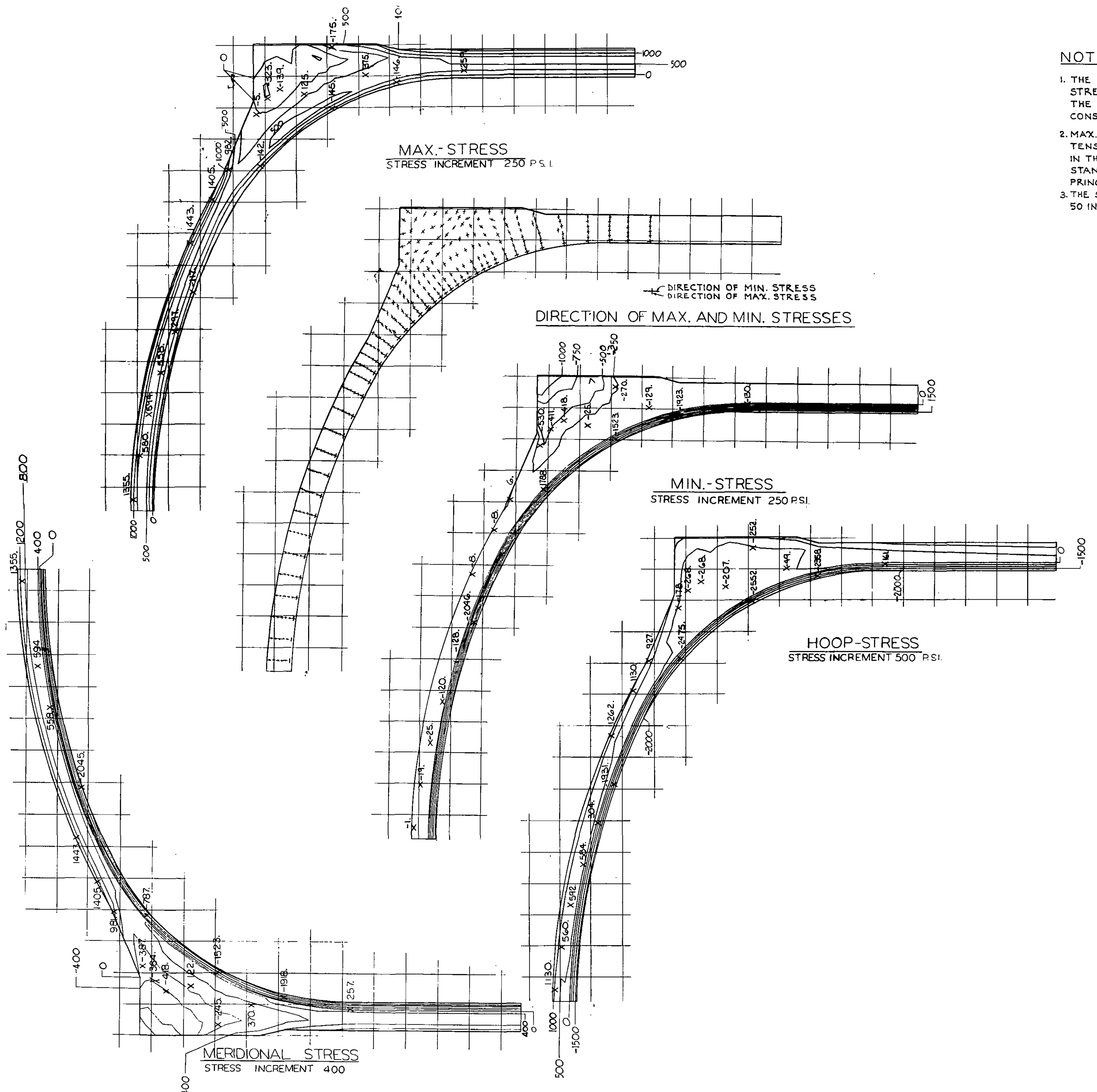
NOTES

1. THE ISO-STRESS CURVES SHOW THE STRESSES IN THE CONCRETE ONLY. THE DIRECTION OF PRINCIPAL STRESSES CONSIDERS THE LINER PLATE ALSO.
2. MAX.-STRESS STANDS FOR THE MORE TENSILE OF THE TWO PRINCIPAL STRESSES IN THE VERTICAL PLANE. MIN.-STRESS STANDS FOR THE MORE COMPRESSIVE PRINCIPAL STRESS.
3. THE SPACING OF THE COORDINATE GRID IS 50 INCHES IN EACH DIRECTION.

REACTOR BUILDING ISOSTRESS PLOT
WALL AND DOME

OCONEE NUCLEAR STATION

Figure 5 - 6
(Sheet 1 of 4)



NOTES

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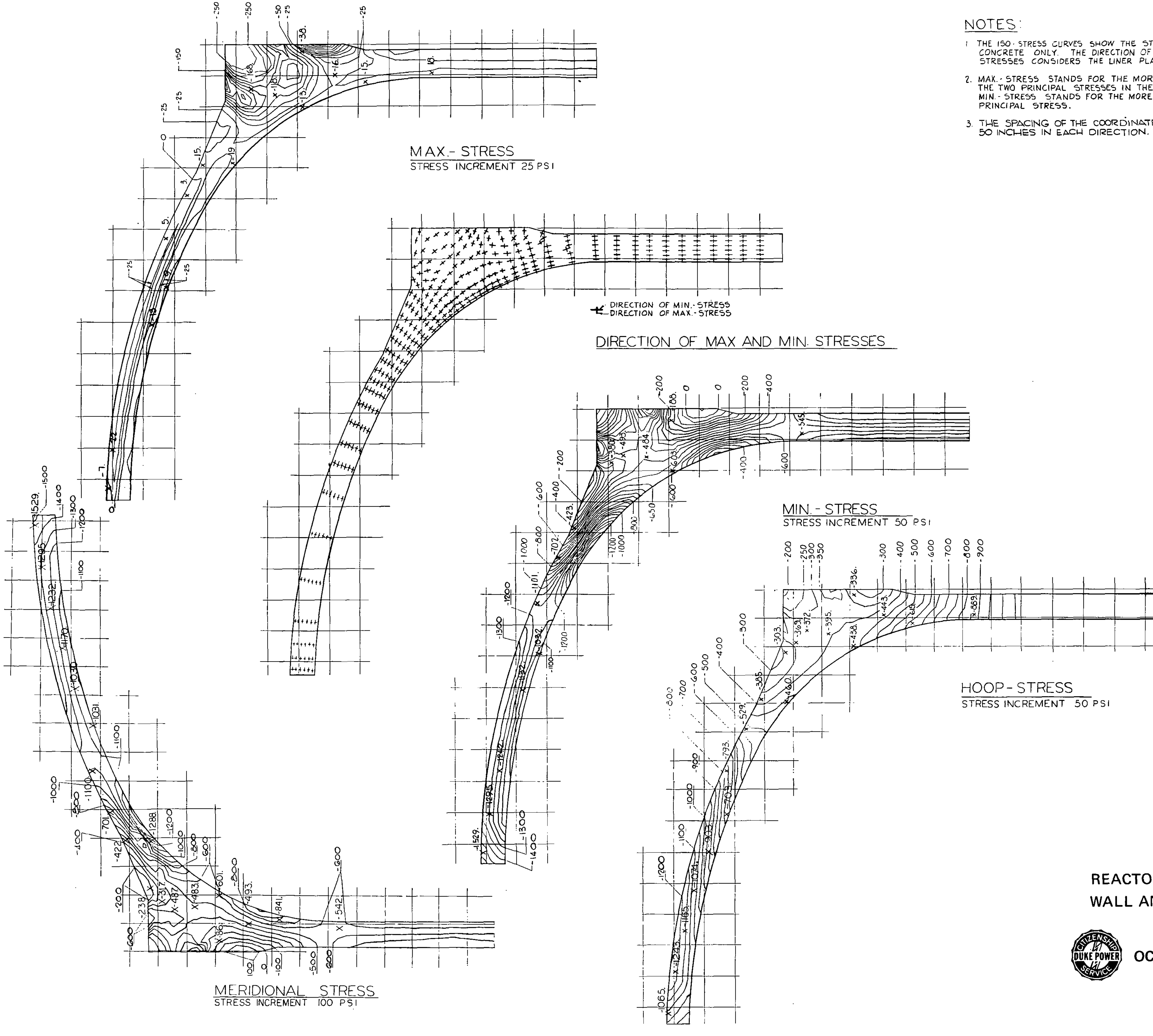
3. THE SPACING OF THE COORDINATE GRID IS 50 INCHES IN EACH DIRECTION.

REACTOR BUILDING ISOSTRESS PLOT
WALL AND DOME

OCONEE NUCLEAR STATION

Figure 5 - 6
(Sheet 2 of 4)



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3. THE SPACING OF THE COORDINATE GRID IS 50 INCHES IN EACH DIRECTION.

REACTOR BUILDING ISOSTRESS PLOT
WALL AND DOME

OCONEE NUCLEAR STATION

Figure 5 - 6
(Sheet 3 of 4)

NOTES

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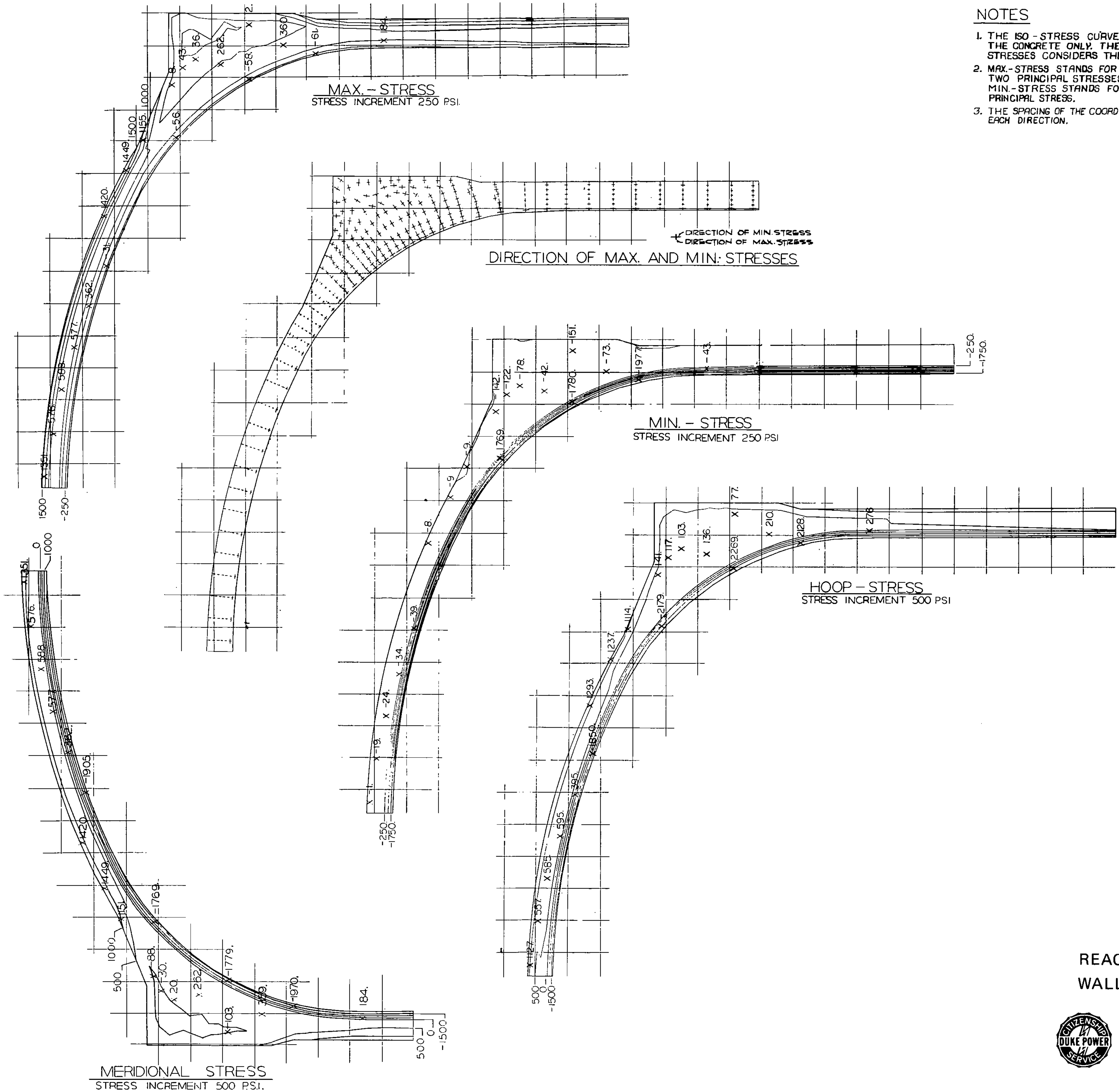
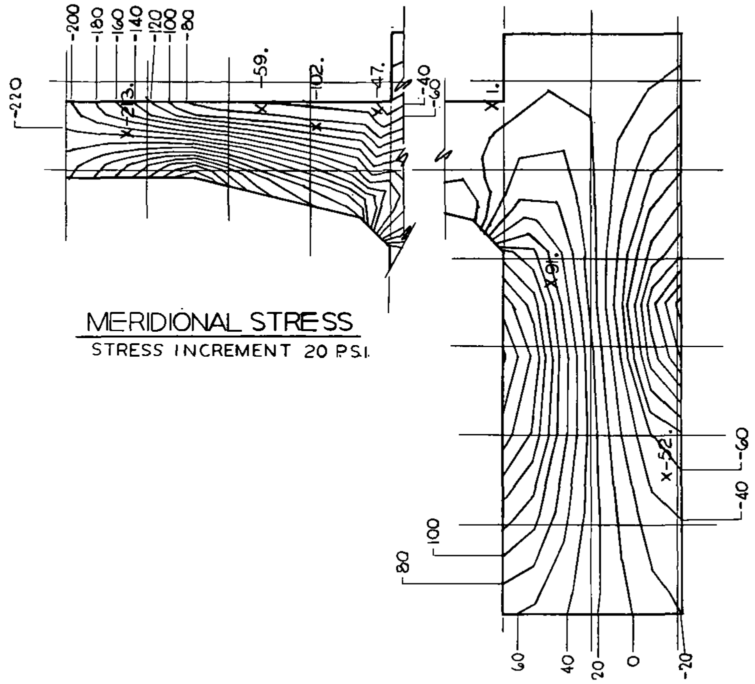


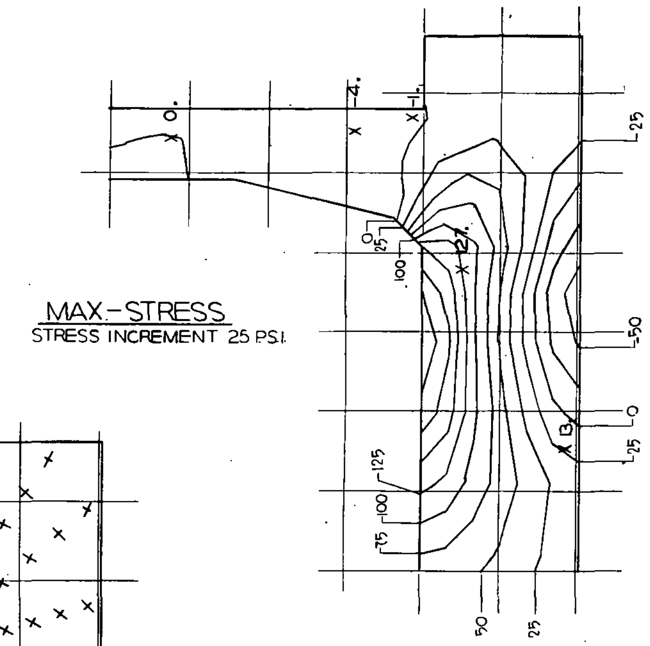
REACTOR BUILDING ISOSTRESS PLOT
WALL AND DOME

OCONEE NUCLEAR STATION

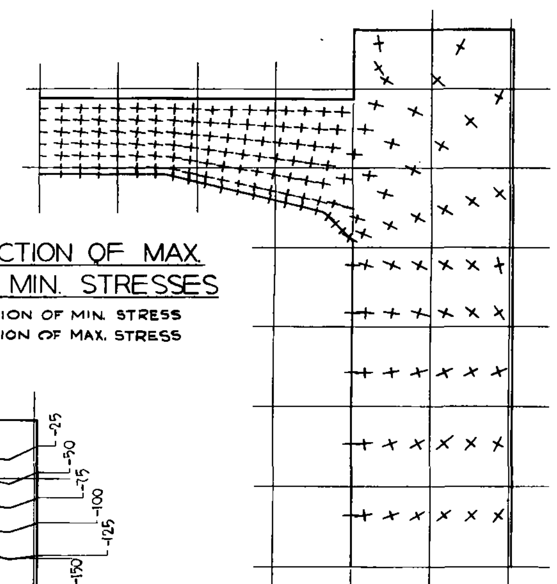
Figure 5 - 6
(Sheet 4 of 4)



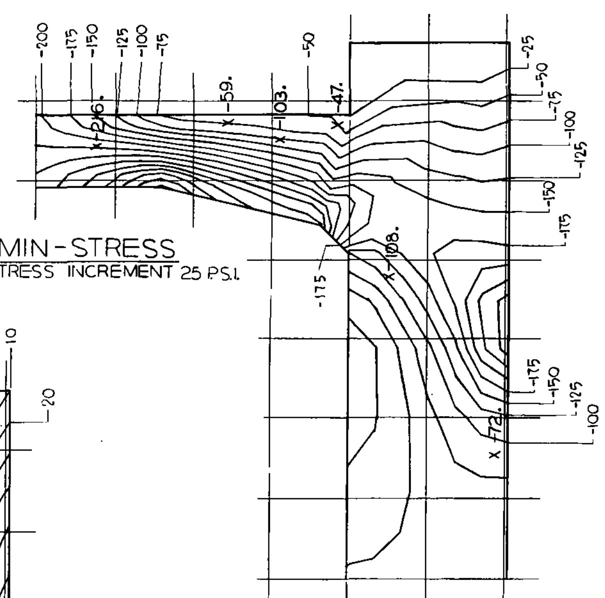
MERIDIONAL STRESS
STRESS INCREMENT 20 P.S.I.



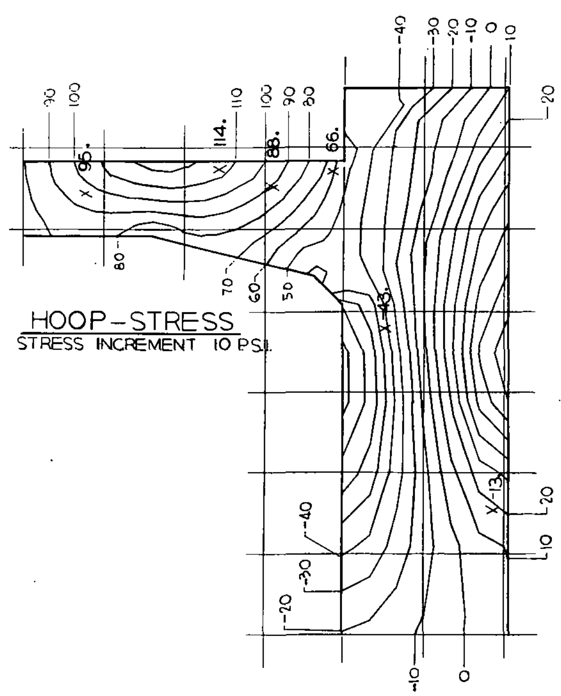
MAX-STRESS
STRESS INCREMENT 25 P.S.I.



DIRECTION OF MAX. AND MIN. STRESSES
DIRECTION OF MIN. STRESS
DIRECTION OF MAX. STRESS



MIN-STRESS
STRESS INCREMENT 25 P.S.I.



HOOP-STRESS
STRESS INCREMENT 10 P.S.I.

NOTES

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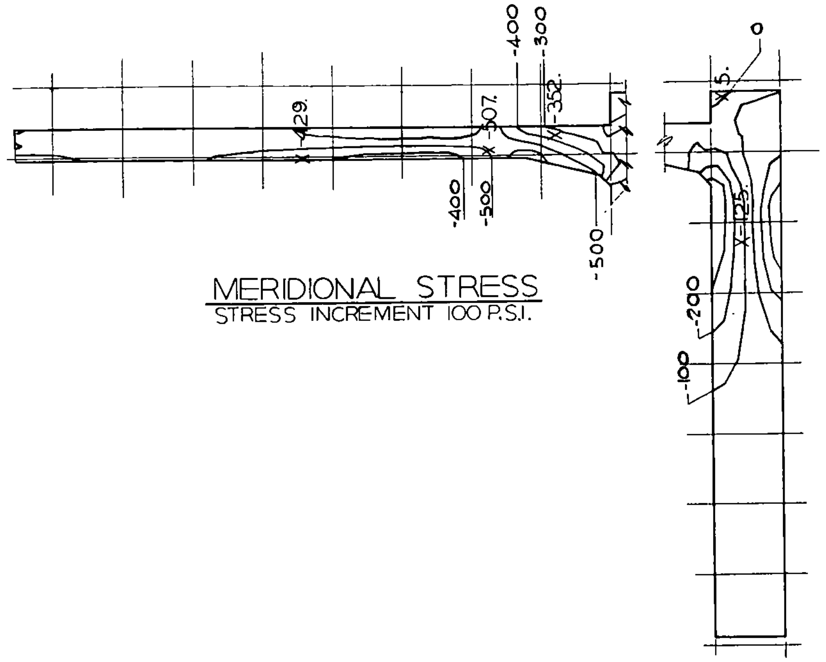
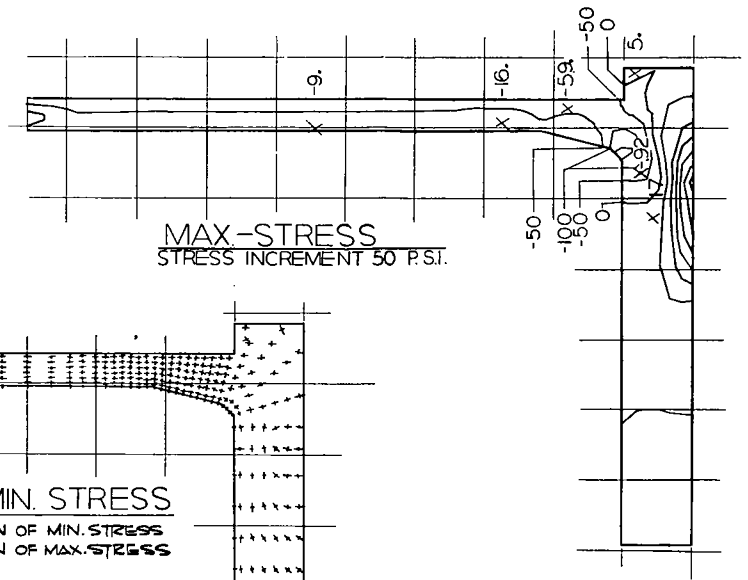
3. THE SPACING OF THE COORDINATE GRID IS 50 INCHES IN EACH DIRECTION.

REACTOR BUILDING ISOSTRESS PLOT
WALL AND BASE

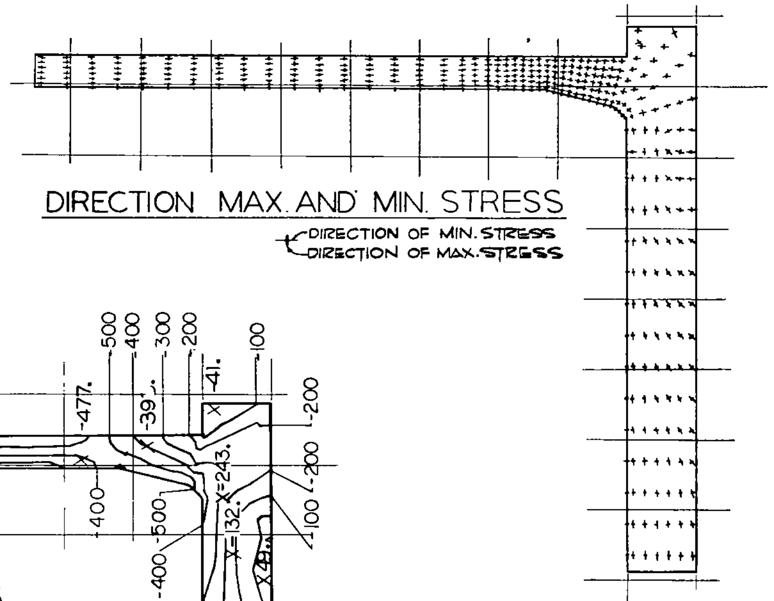
OCONEE NUCLEAR STATION
Figure 5 - 7
(Sheet 1 of 6)



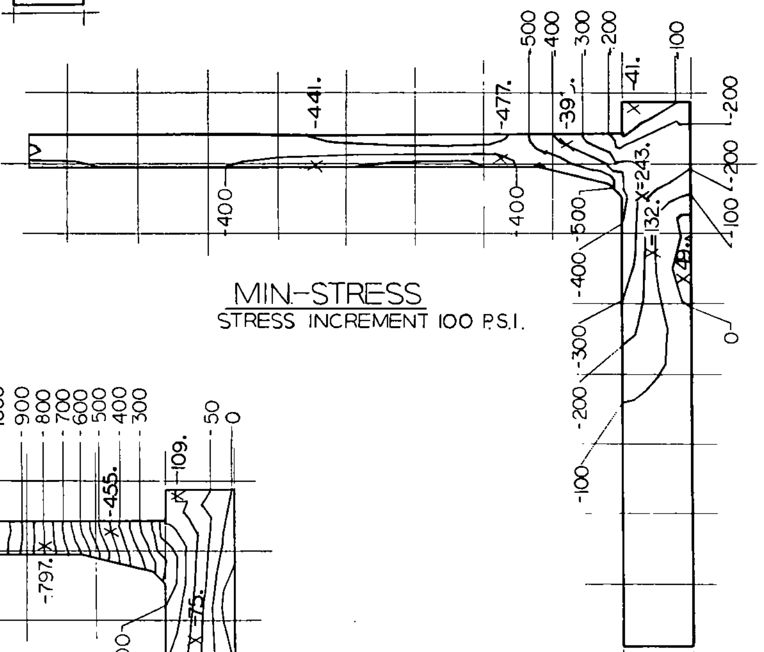
MERIDIONAL STRESS
STRESS INCREMENT 100 P.S.I.



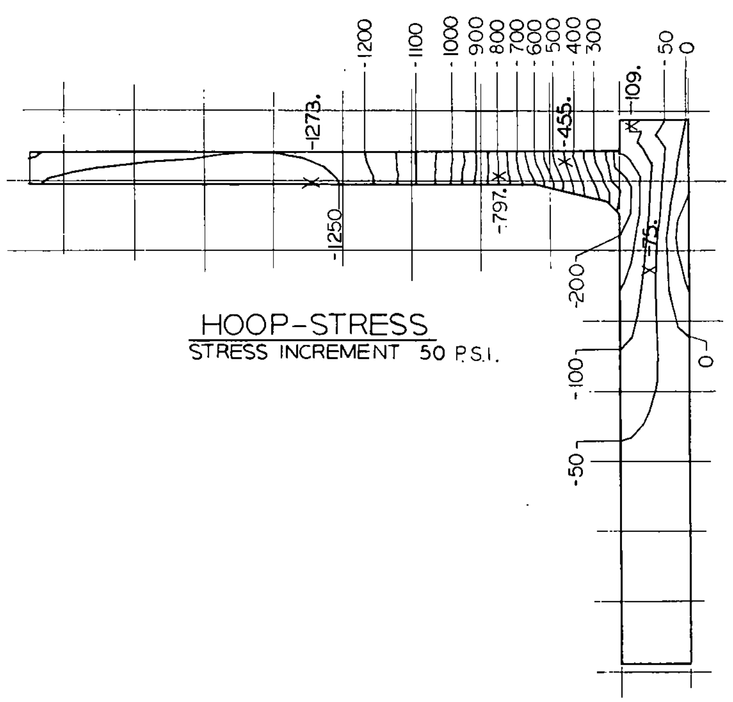
MAX-STRESS
STRESS INCREMENT 50 P.S.I.



DIRECTION MAX. AND MIN. STRESS
DIRECTION OF MIN. STRESS
DIRECTION OF MAX. STRESS



MIN-STRESS
STRESS INCREMENT 100 P.S.I.



HOOP-STRESS
STRESS INCREMENT 50 P.S.I.

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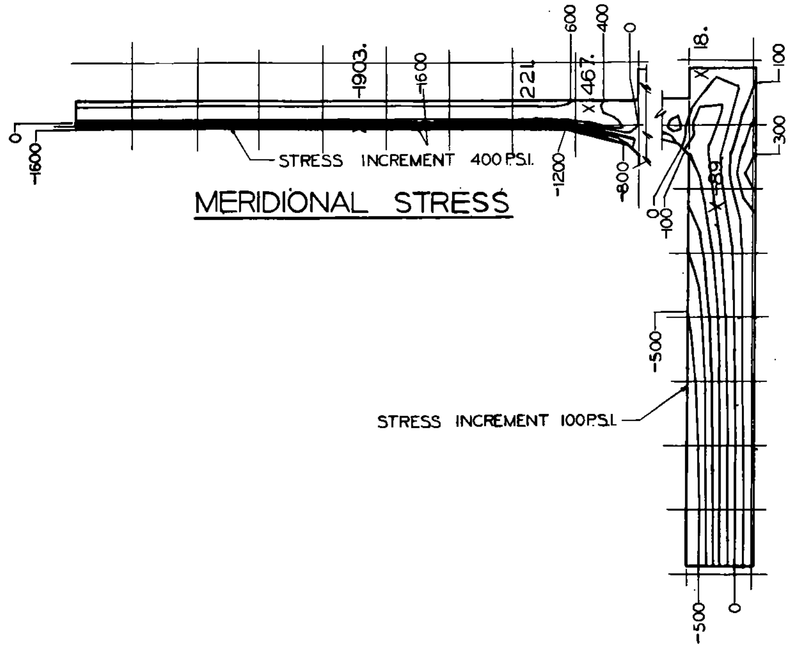
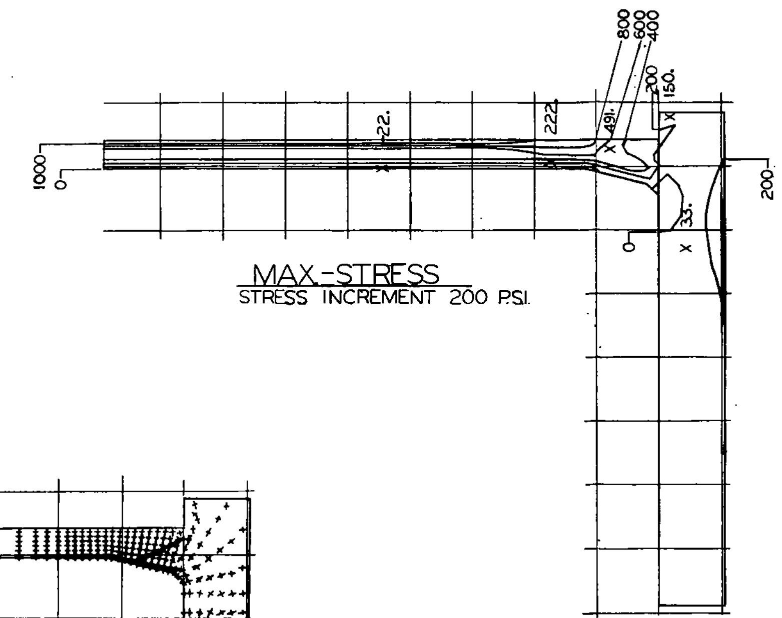
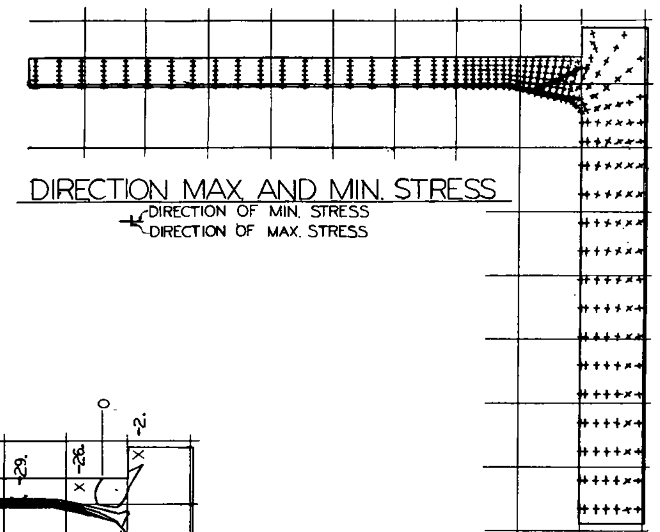
3. THE SPACING OF THE COORDINATE GRID IS 100 INCHES IN EACH DIRECTION.

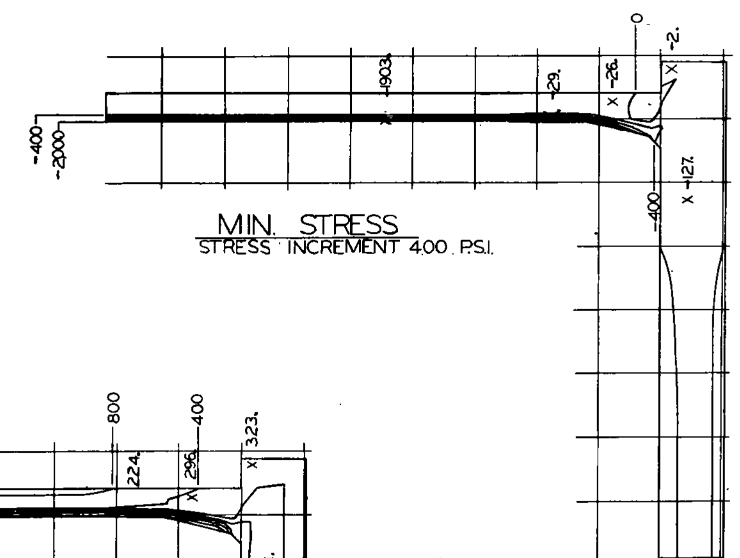
REACTOR BUILDING ISOSTRESS PLOT
WALL AND BASE

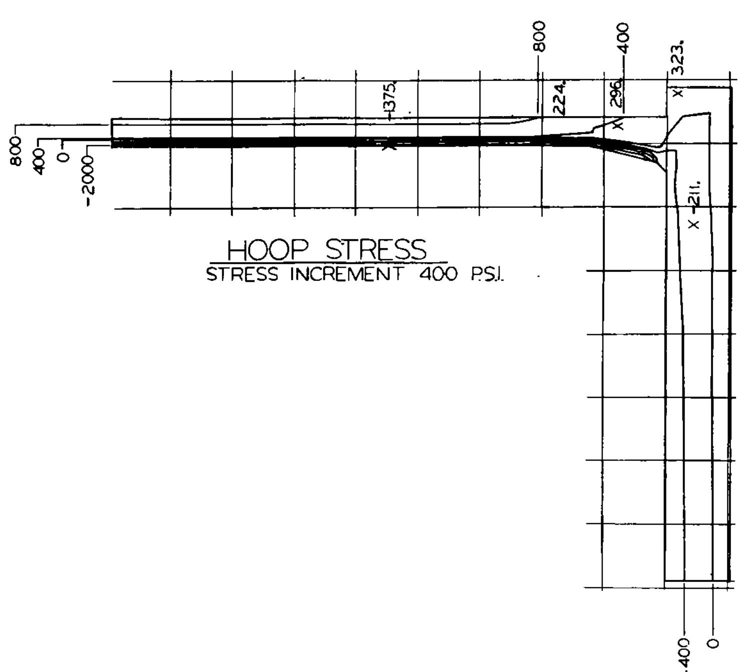
OCONEE NUCLEAR STATION
Figure 5 - 7
(Sheet 2 of 6)











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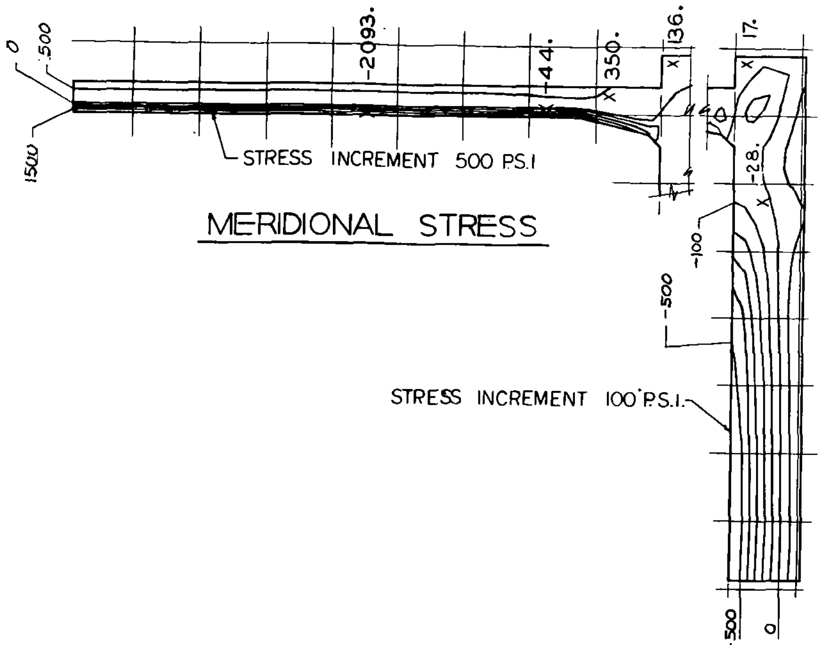
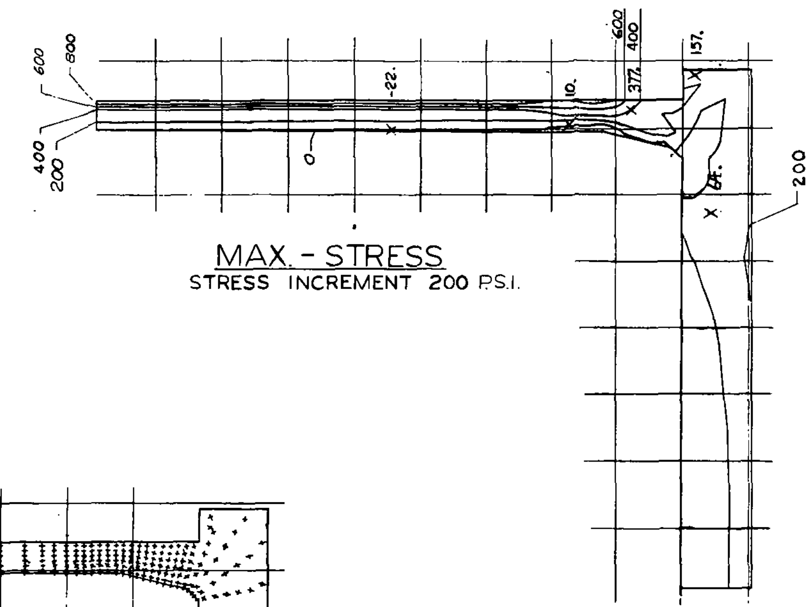
REACTOR BUILDING ISOSTRESS PLOT
WALL AND BASE

OCONEE NUCLEAR STATION

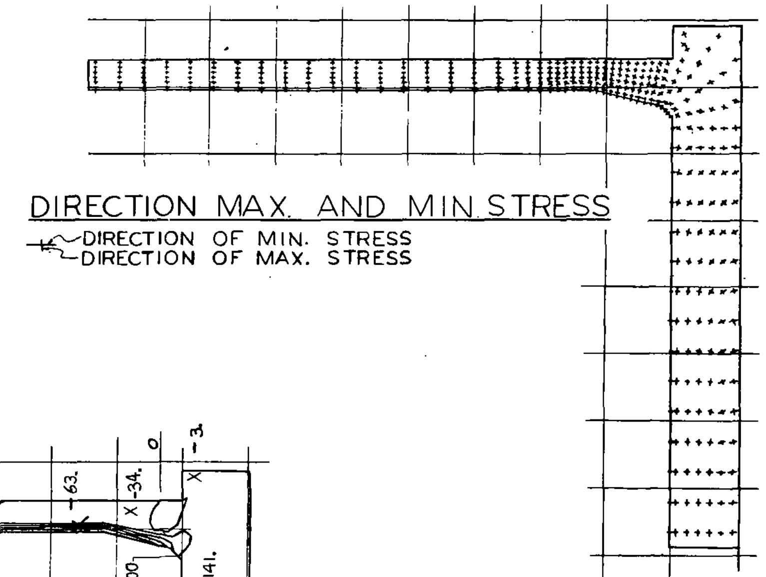
Figure 5 - 7
(Sheet 3 of 6)



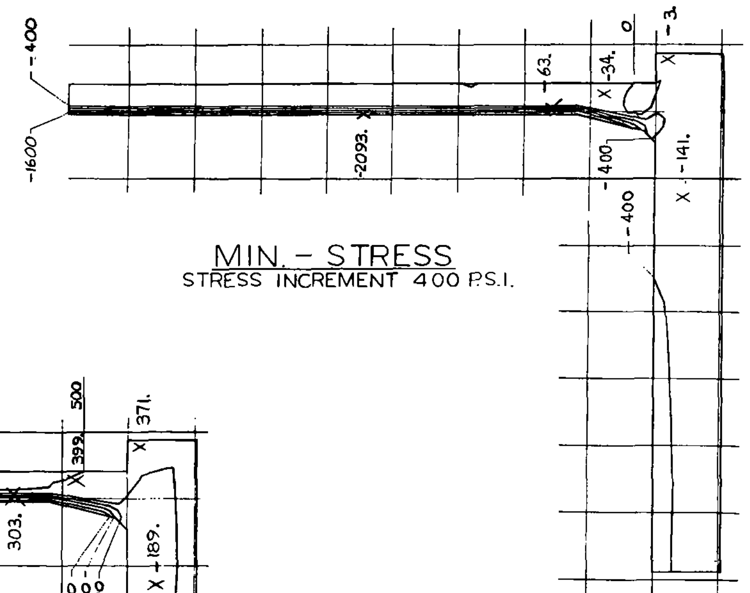
MERIDIONAL STRESS



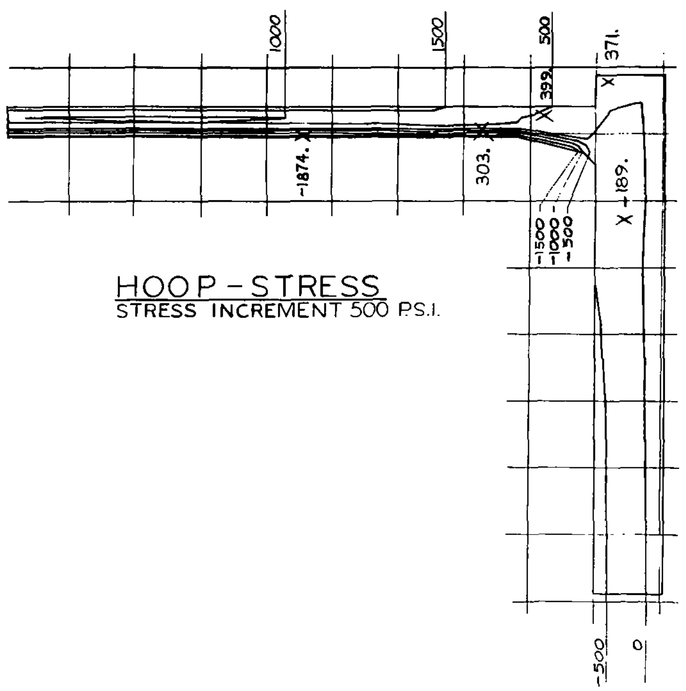
MAX. - STRESS
STRESS INCREMENT 200 P.S.I.



DIRECTION MAX. AND MIN. STRESS
DIRECTION OF MIN. STRESS
DIRECTION OF MAX. STRESS



MIN. - STRESS
STRESS INCREMENT 400 P.S.I.



HOOP - STRESS
STRESS INCREMENT 500 P.S.I.

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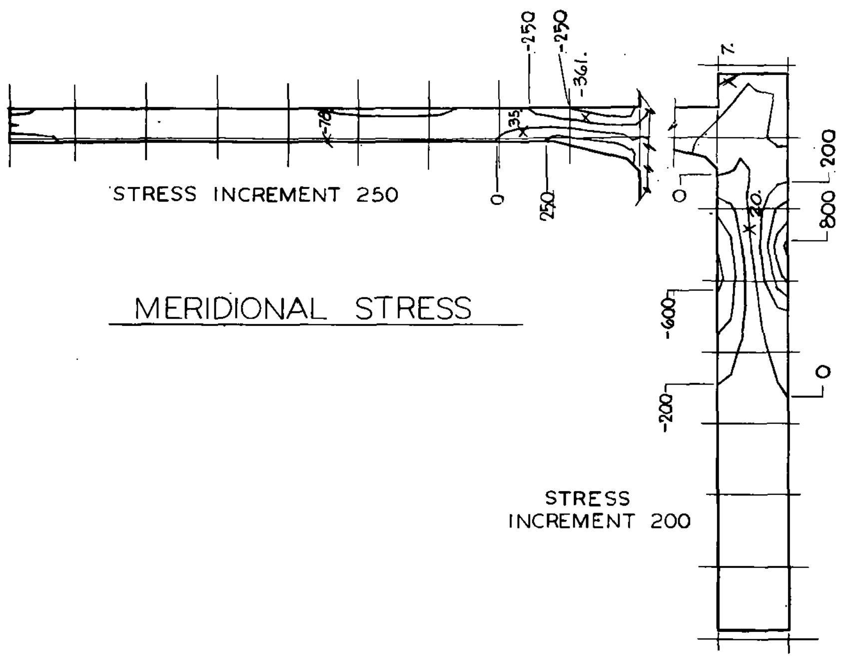
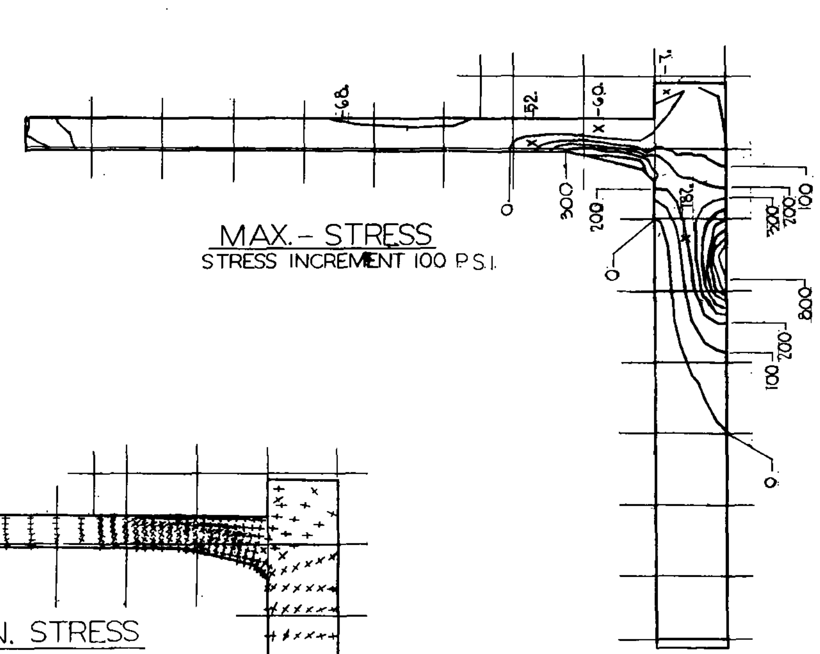
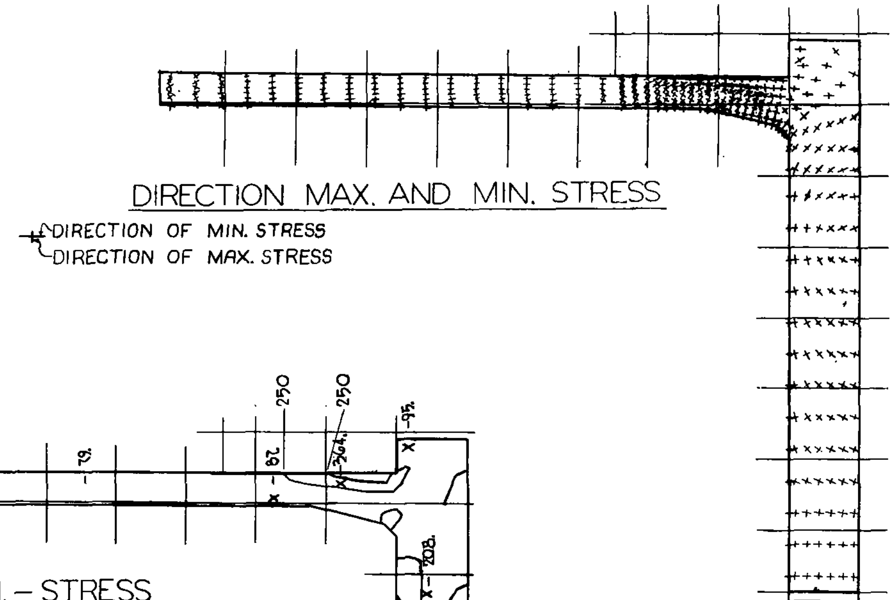
REACTOR BUILDING ISOSTRESS PLOT
WALL AND BASE

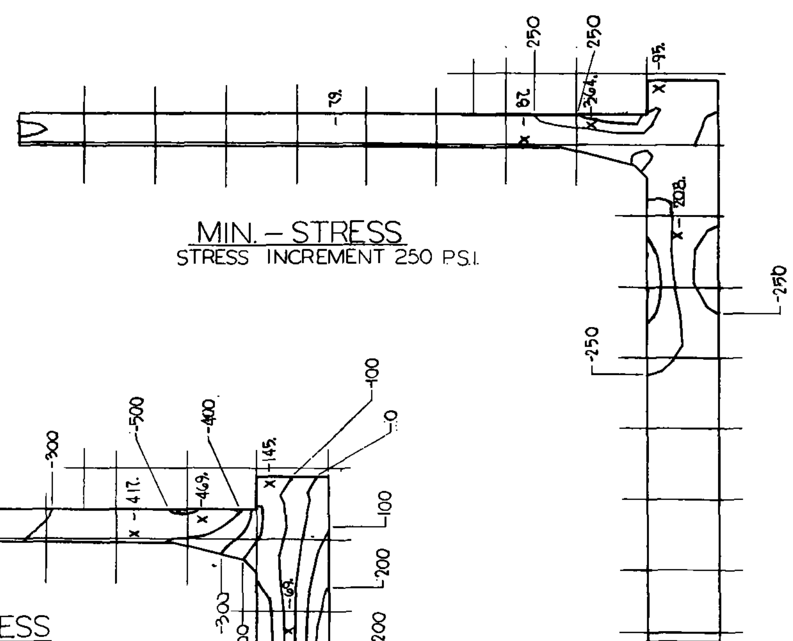
OCONEE NUCLEAR STATION

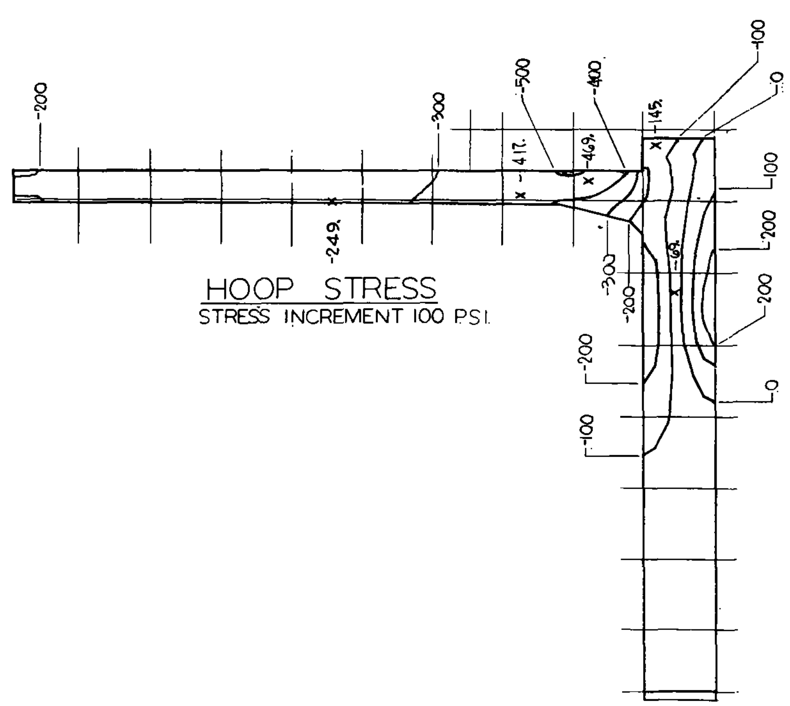
Figure 5 - 7
(Sheet 4 of 6)











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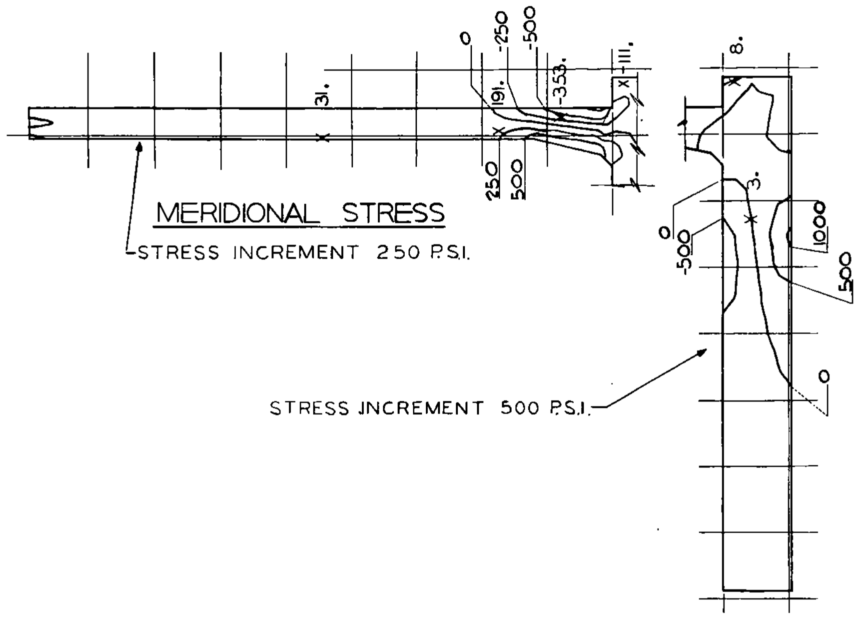
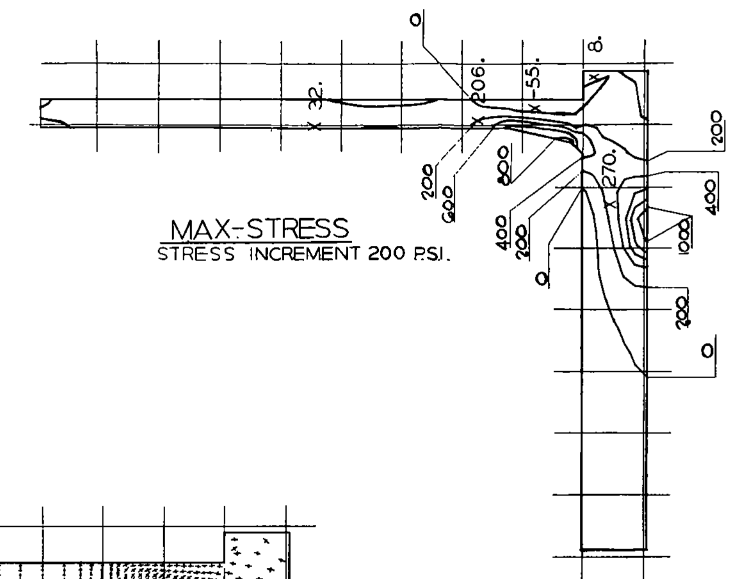
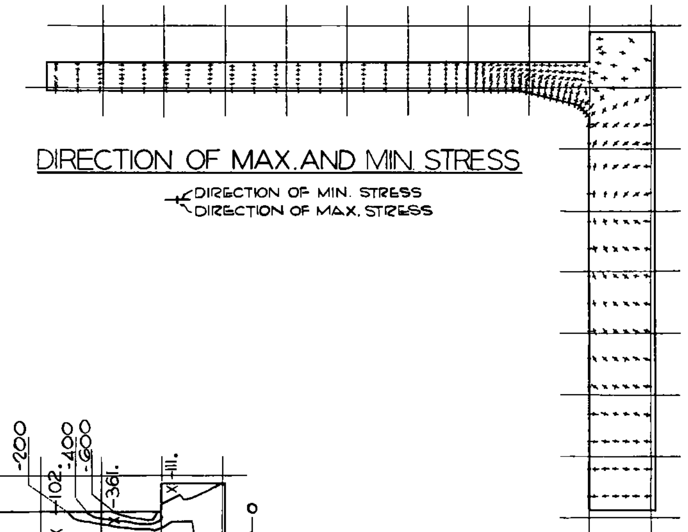
REACTOR BUILDING ISOSTRESS PLOT
WALL AND BASE

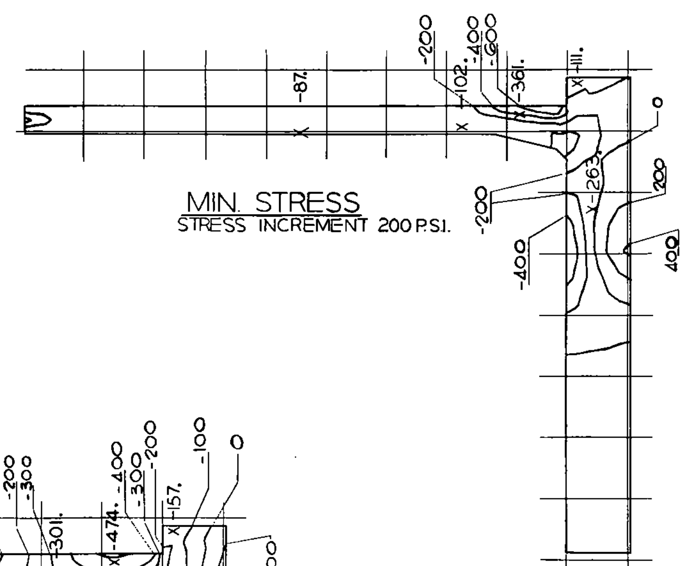
OCONEE NUCLEAR STATION

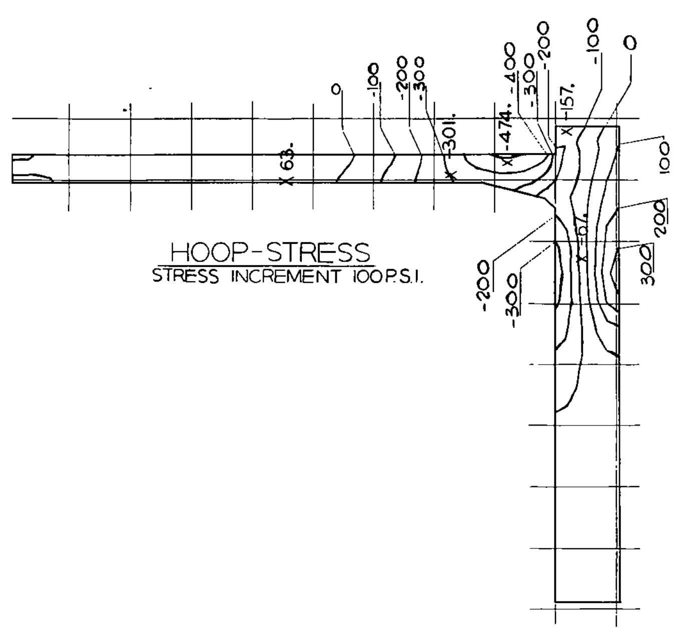
Figure 5 - 7
(Sheet 5 of 6)











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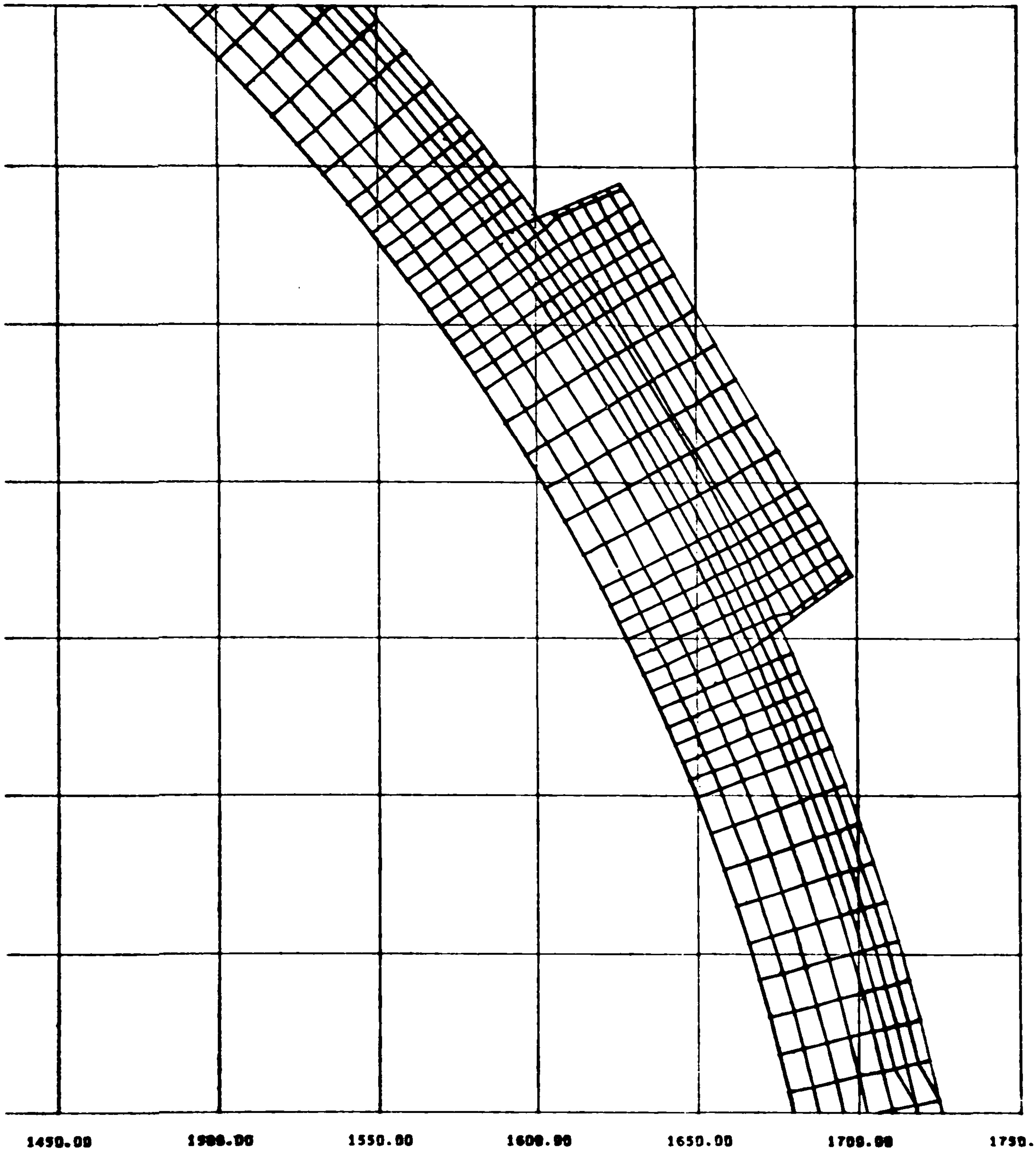
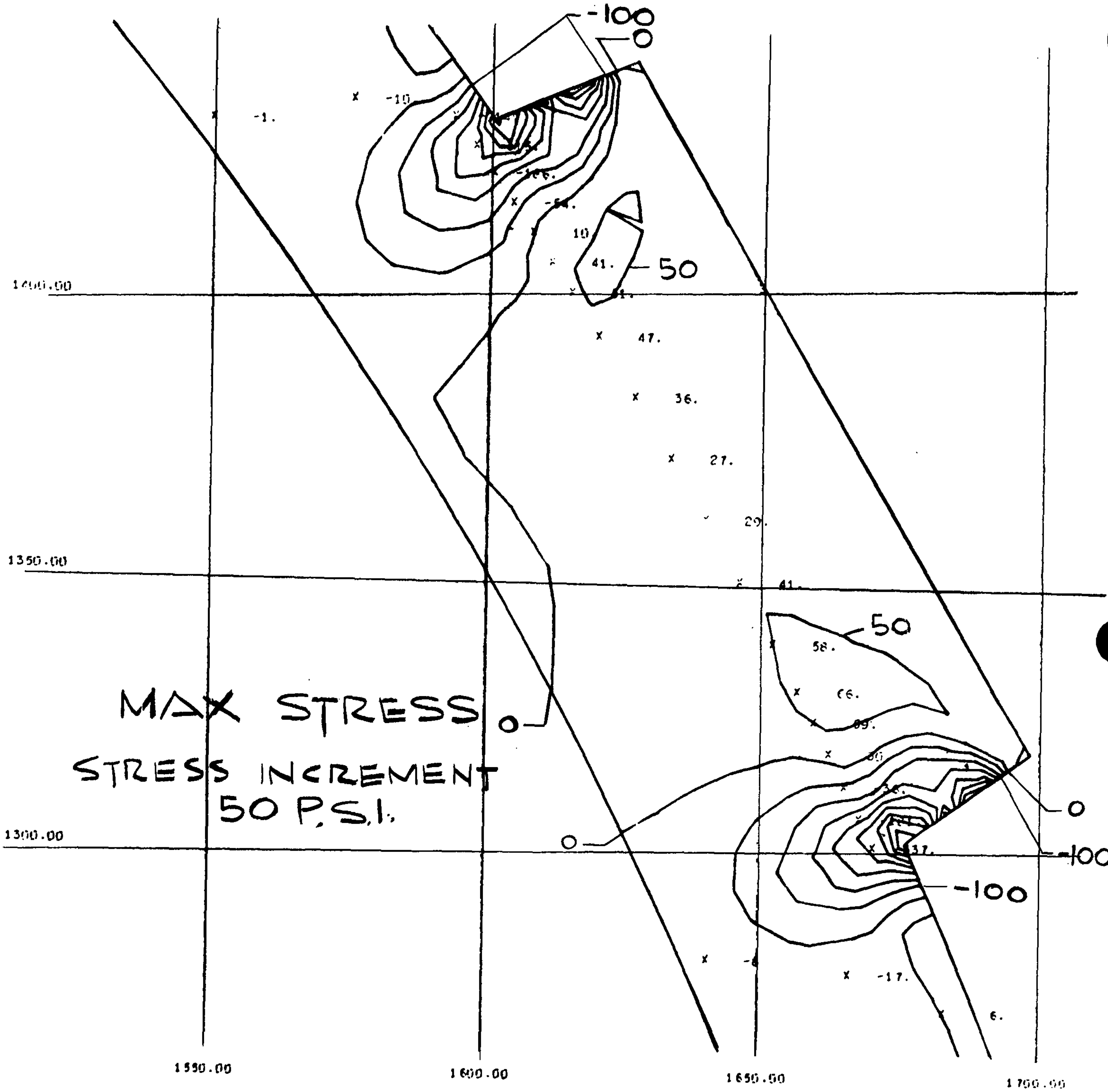
REACTOR BUILDING ISOSTRESS PLOT
WALL AND BASE

OCONEE NUCLEAR STATION
Figure 5 - 7
(Sheet 6 of 6)

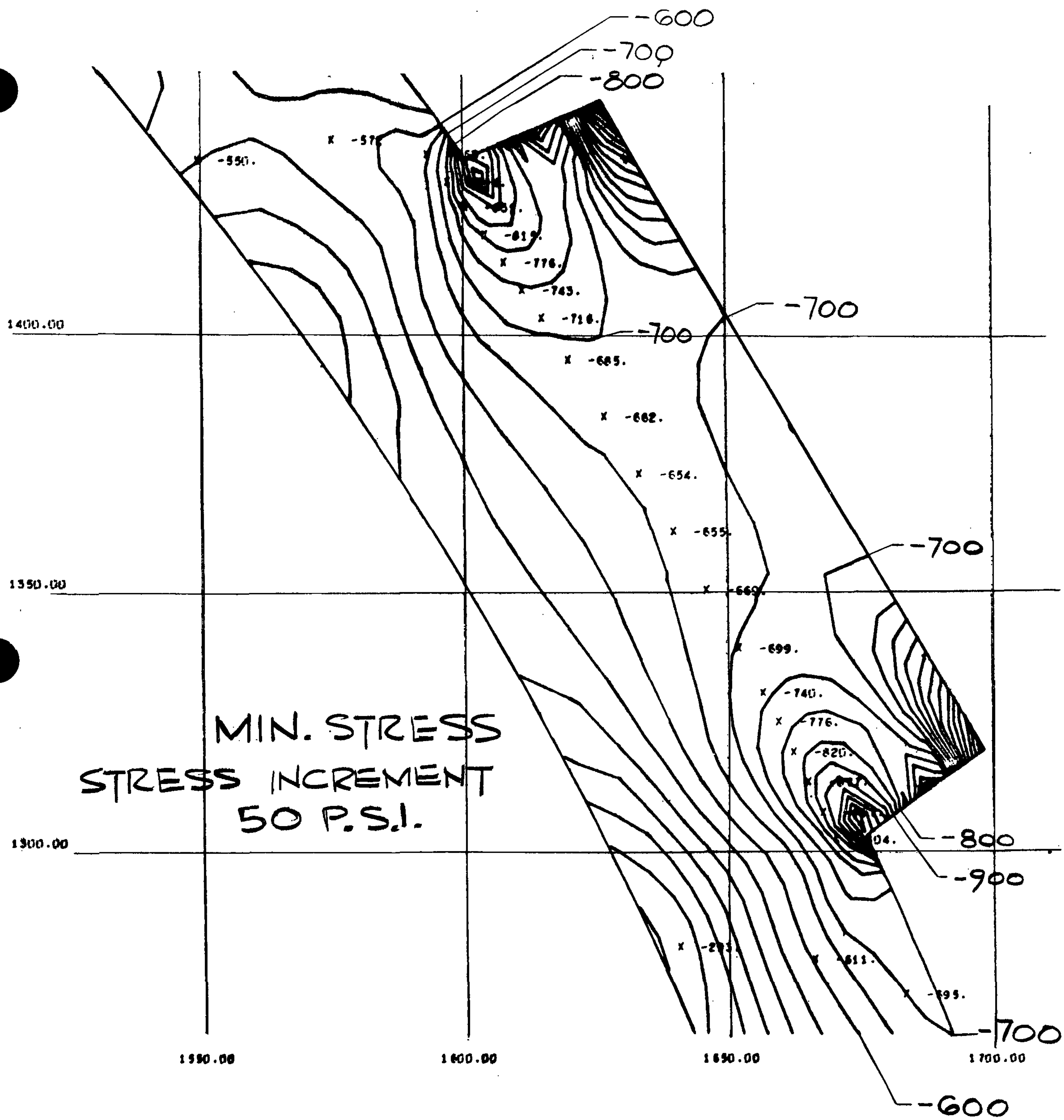


REACTOR BUILDING FINITE ELEMENT
MESH WALL BUTTRESSES



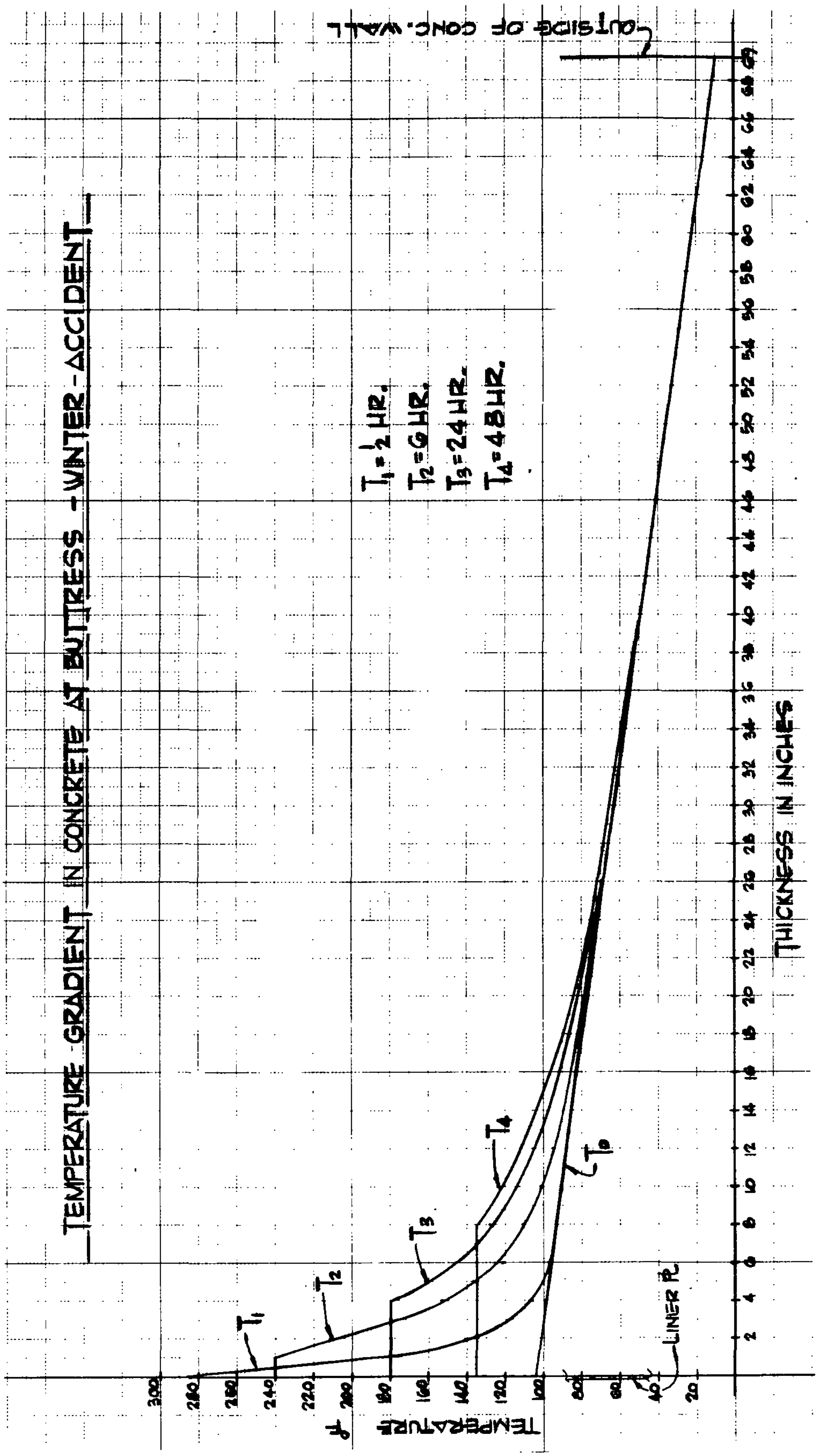
REACTOR BUILDING ISOSTRESS PLOT
FOR BUTTRESSES

OCONEE NUCLEAR STATION
Figure 5 - 9
(Sheet 1 of 2)



REACTOR BUILDING ISOSTRESS PLOT
FOR BUTTRESSES

OCONEE NUCLEAR STATION
Figure 5 - 9
(Sheet 2 of 2)

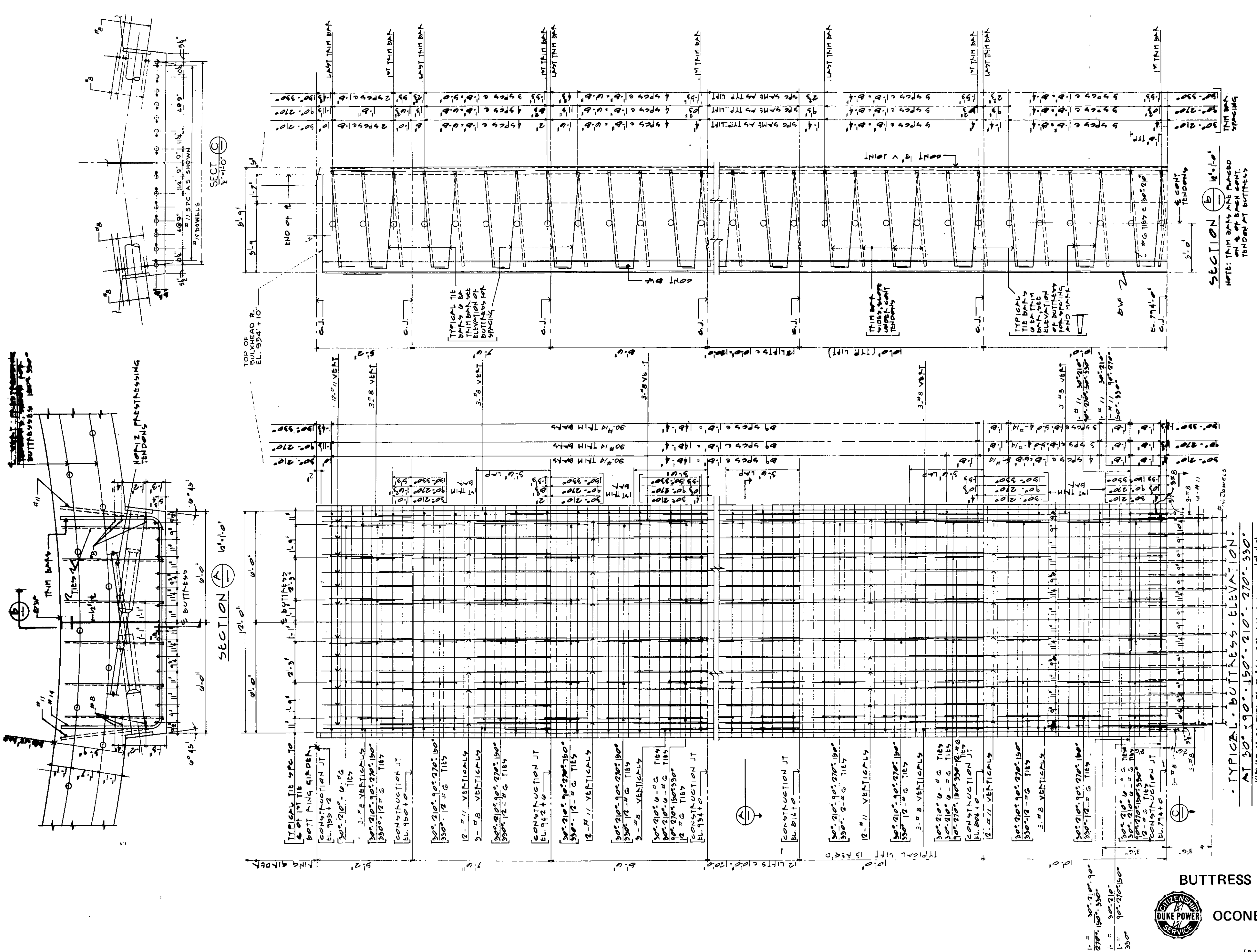


TEMPERATURE GRADIENT AT BUTTRESS

OCONEE NUCLEAR STATION

Figure 5 - 9A

(New) Rev. 6 6/22/70

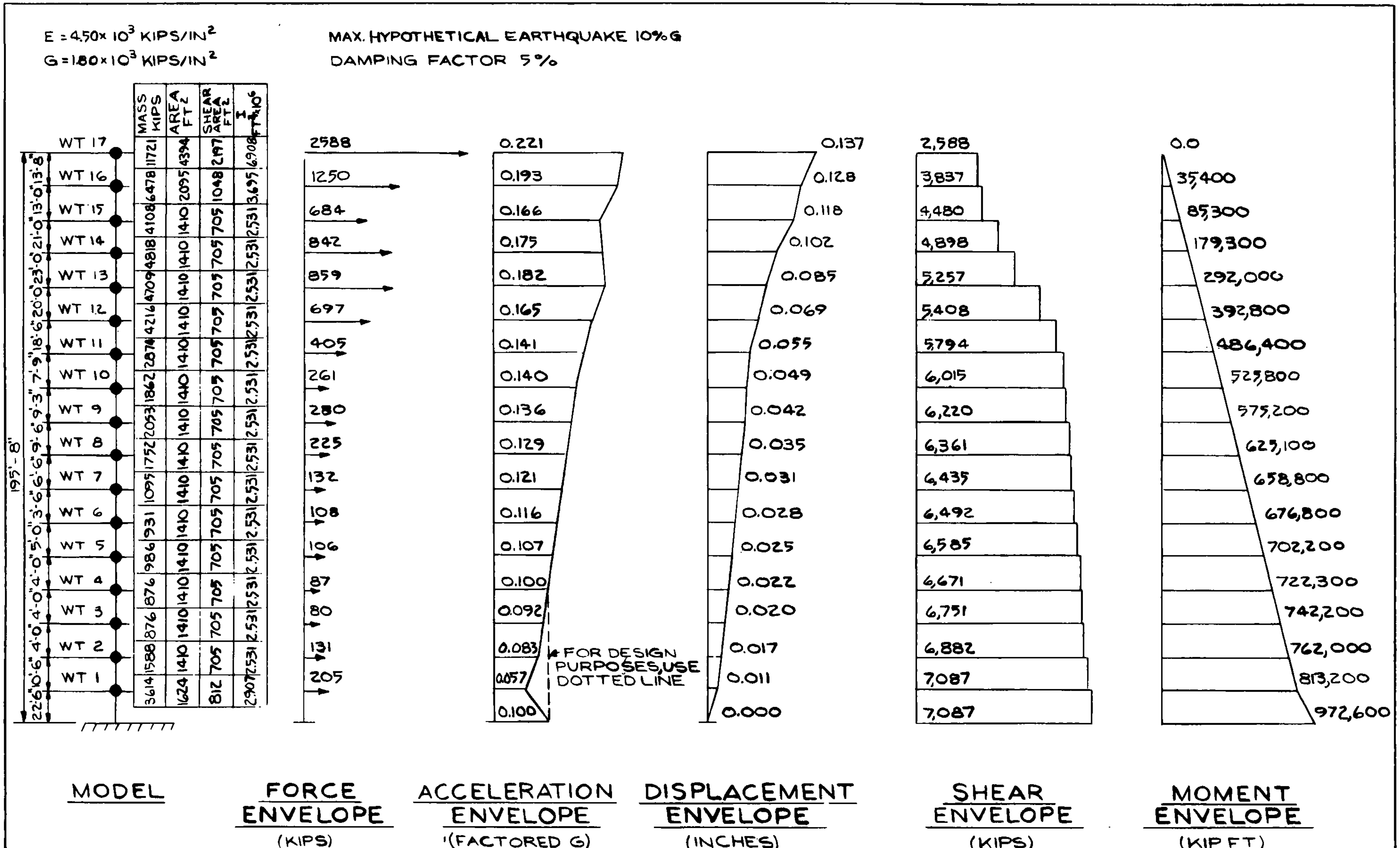


BUTTRESS REINFORCING DETAILS

OCONEE NUCLEAR STATION

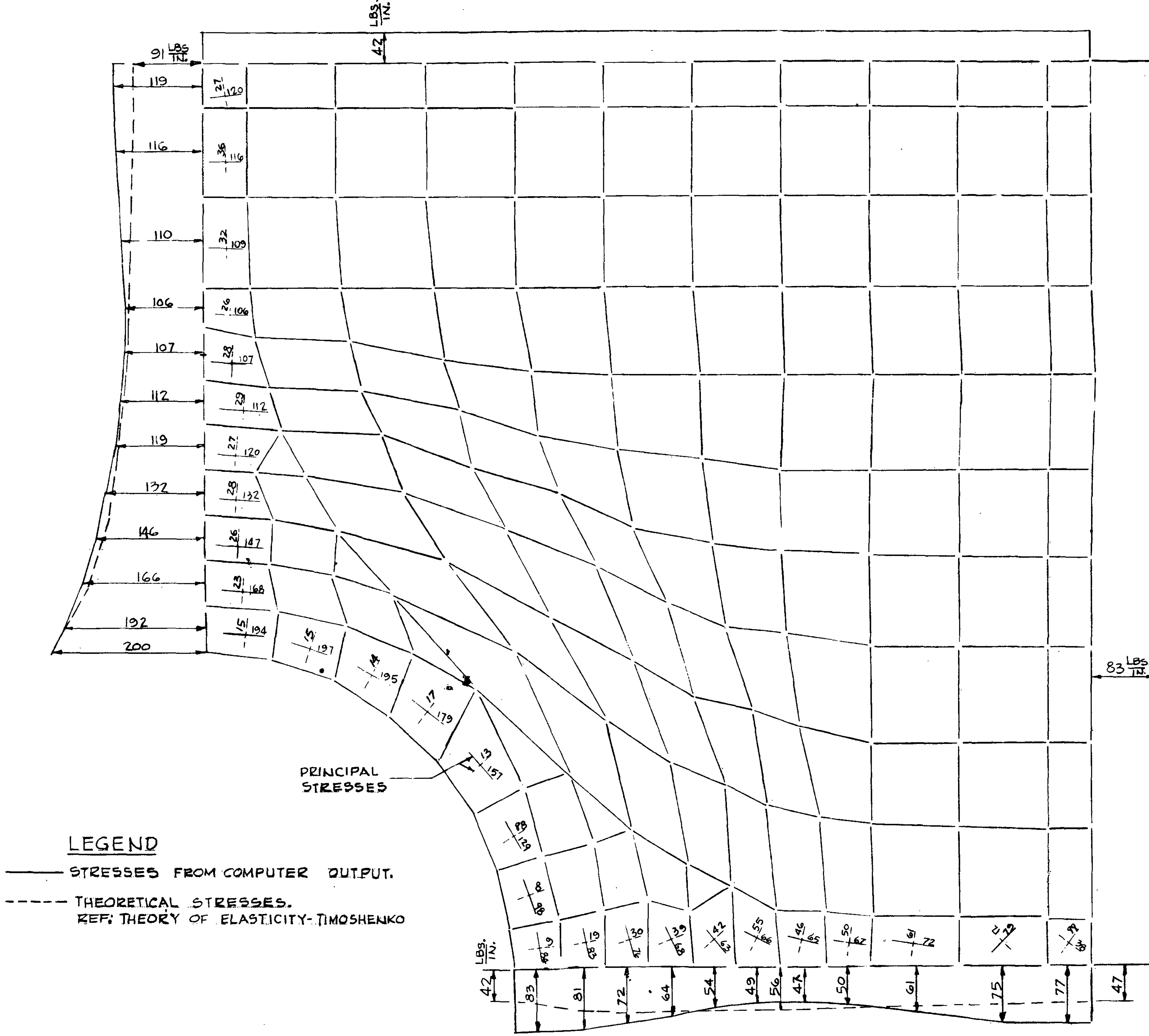
Figure 5 - 9B

(New) Rev. 6 6/22/70



REACTOR BLDG.- SEISMIC MODEL RESULTS

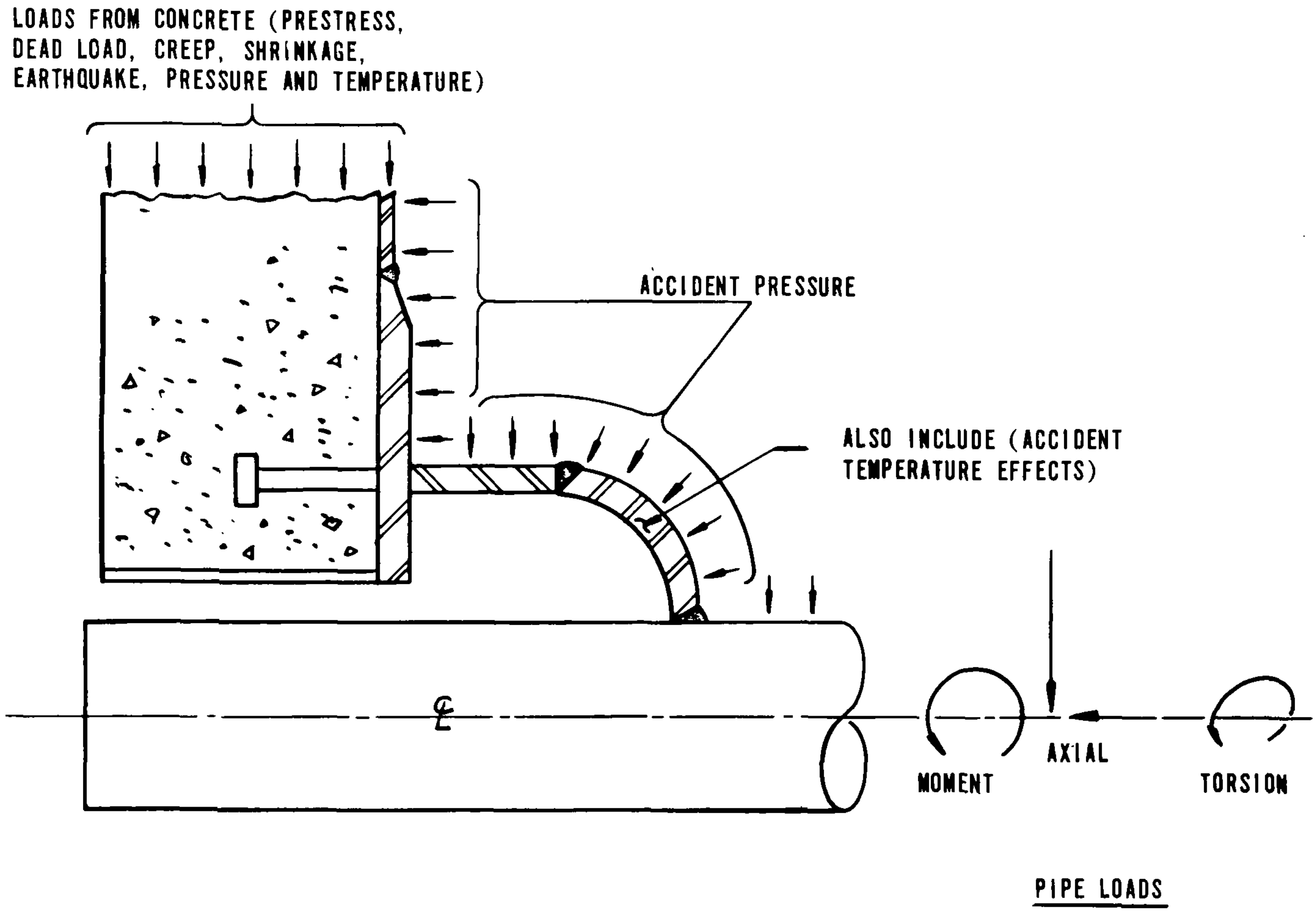
A- 2



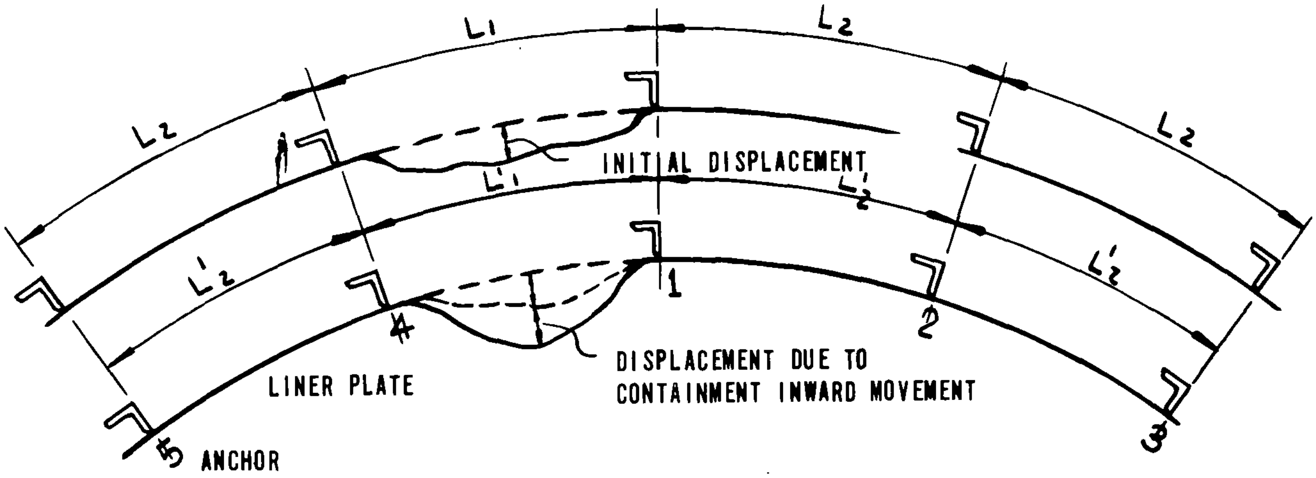
REACTOR BUILDING EQUIPMENT HATCH MESH

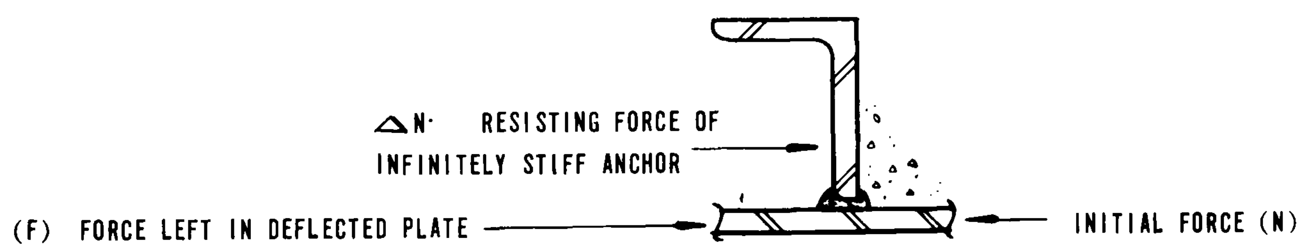
OCONEE NUCLEAR STATION

Figure 5 - 11



REACTOR BUILDING
PENETRATION LOADS



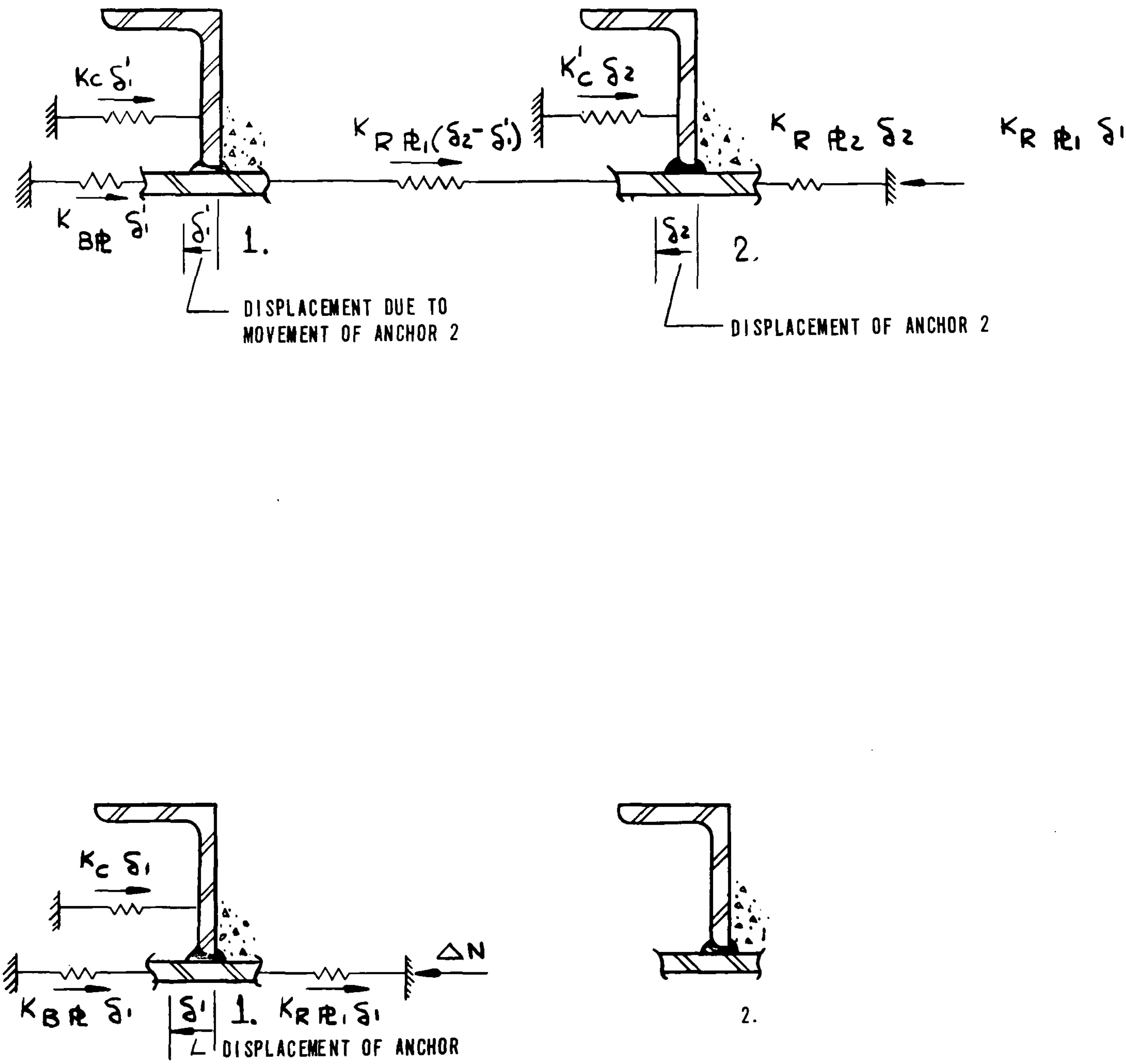


REACTOR BUILDING MODEL-LINER PLATE
ANALYSIS FOR RADIAL DISPLACEMENT

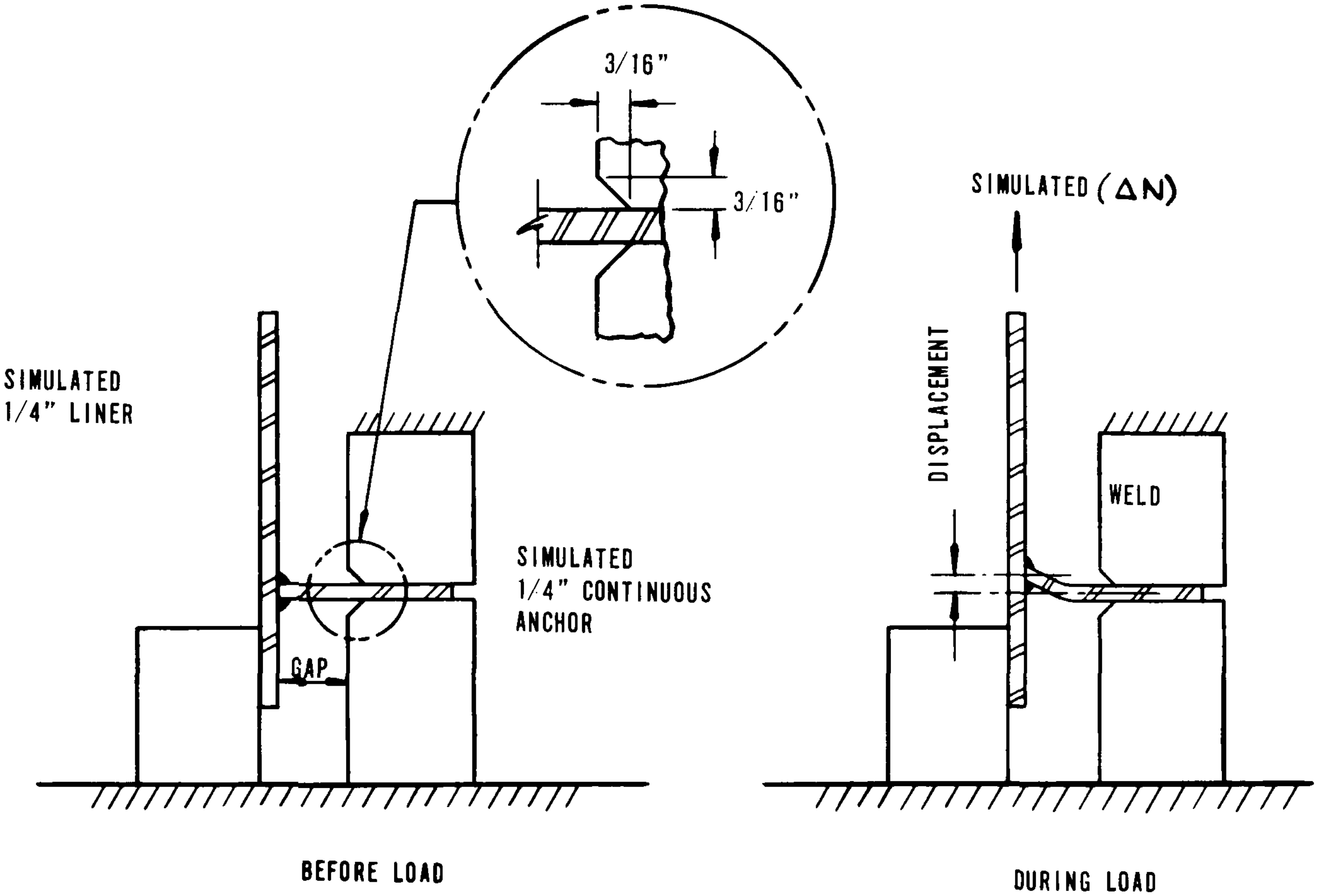
DUKE POWER CITIZENSHIP SERVICE

OCONEE NUCLEAR STATION

Figure 5 - 13



REACTOR BUILDING MODEL FOR LINER PLATE
ANALYSIS FOR ANCHOR DISPLACEMENT

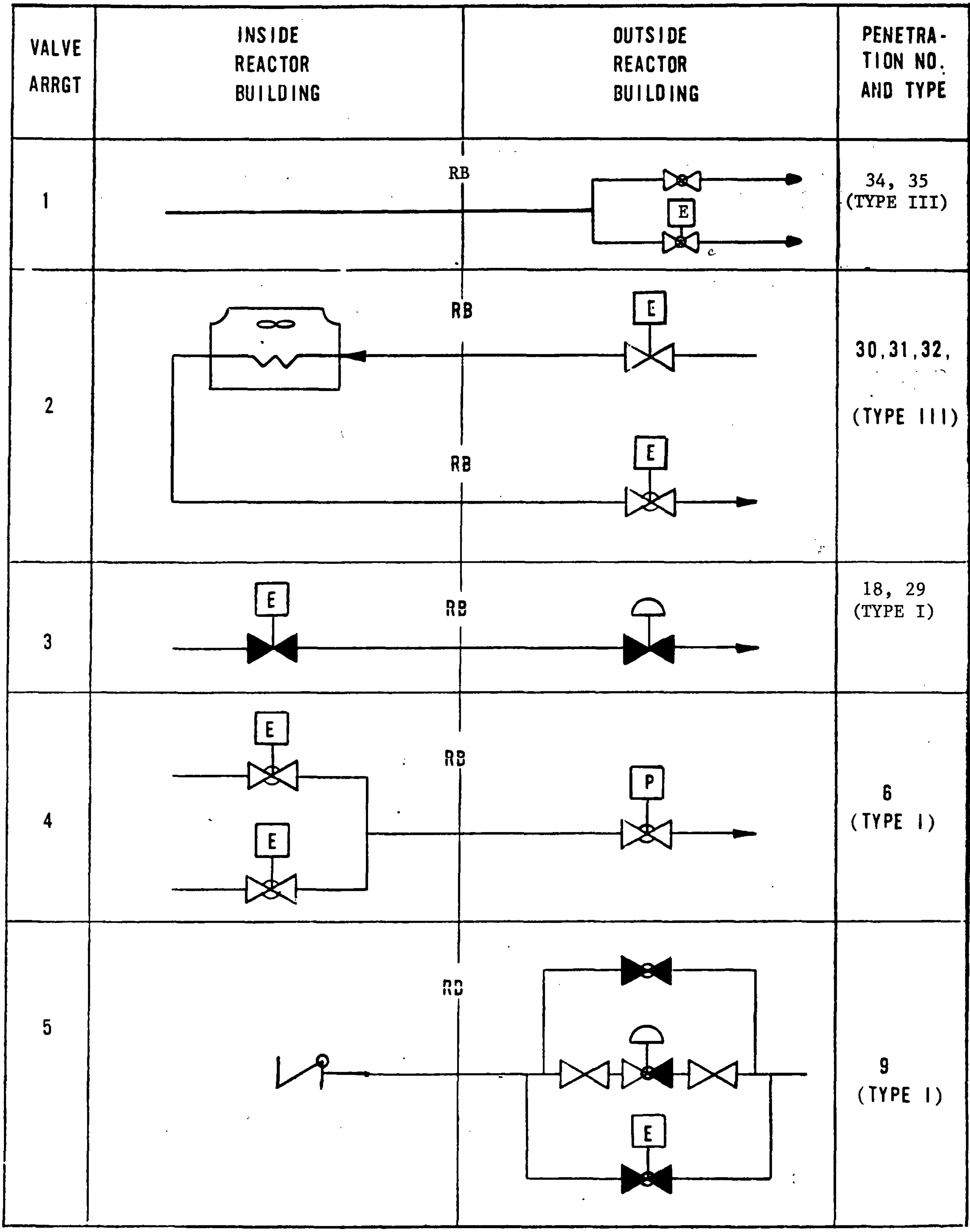


WELD CONFIGURATION	GAP (IN)	ULTIMATE LOAD (K/IN)	ULTIMATE DISPLACEMENT (IN)	LOCATION OF FAILURE
3/16	0	14.95	.14	LINER PLATE
3/16	5/8	5.56	.68	ANCHOR WELD
3/16 6-12	0	7.65	.18	ANCHOR WELD
3/16 6-12	5/8	2.93	.60	ANCHOR WELD
3/16 4-12	0	6.67	.18	ANCHOR WELD
3/16 4-12	5/8	2.46	.30	ANCHOR WELD

OCONEE NUCLEAR STATION

Figure 5 - 15

REACTOR BUILDING – RESULTS FROM TESTS ON LINER PLATE ANCHORS

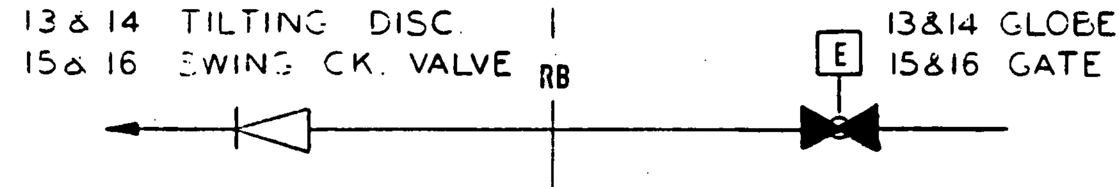
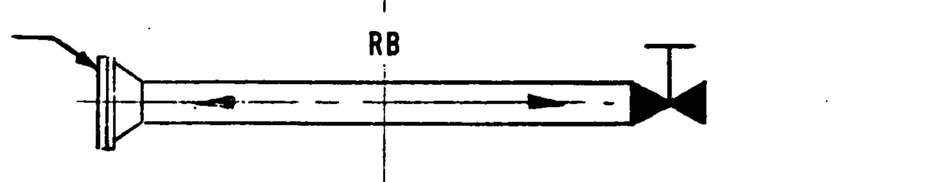
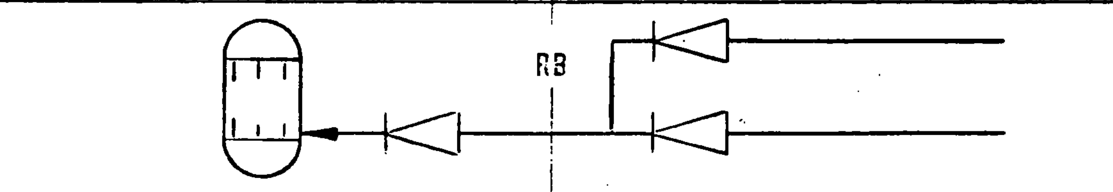
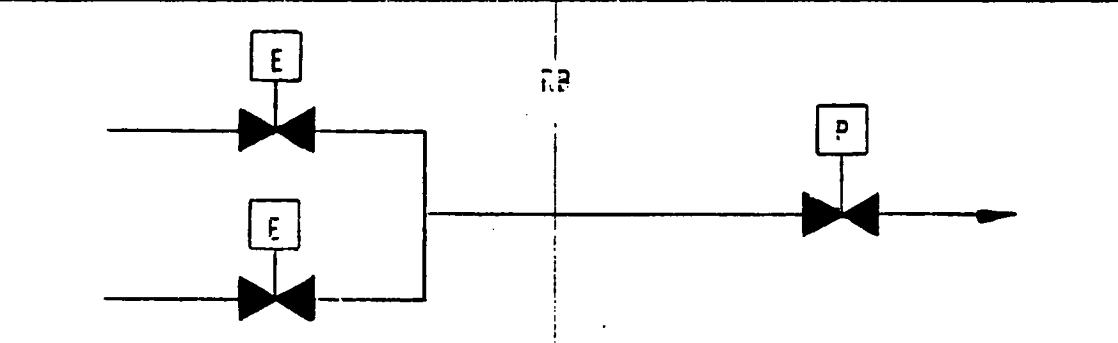
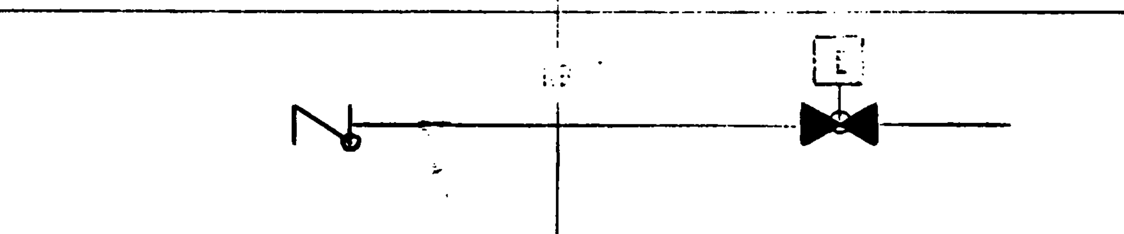
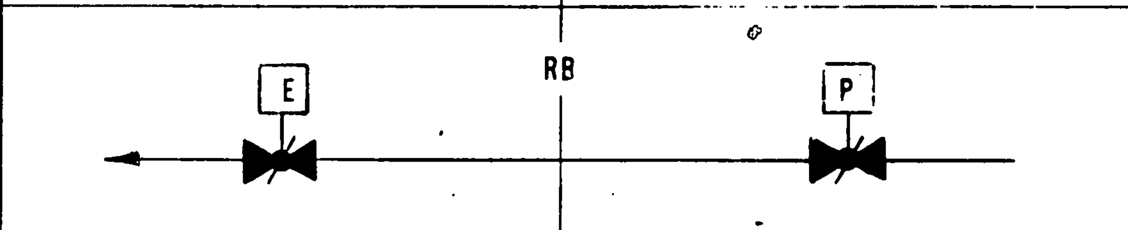
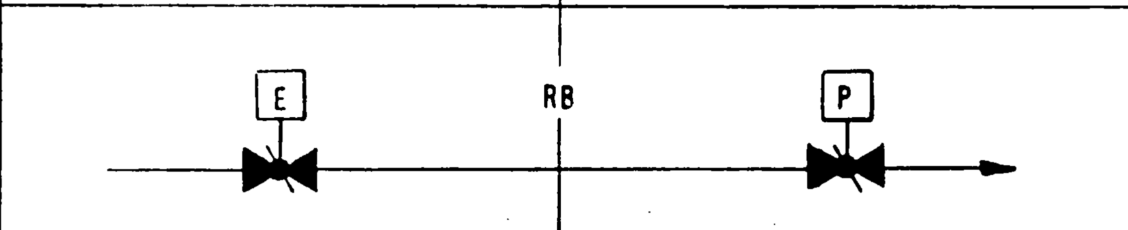


REACTOR BUILDING ISOLATION
VALVE ARRANGEMENT

OCONEE NUCLEAR STATION

Figure 5 - 16

Rev. 37 06/03/76

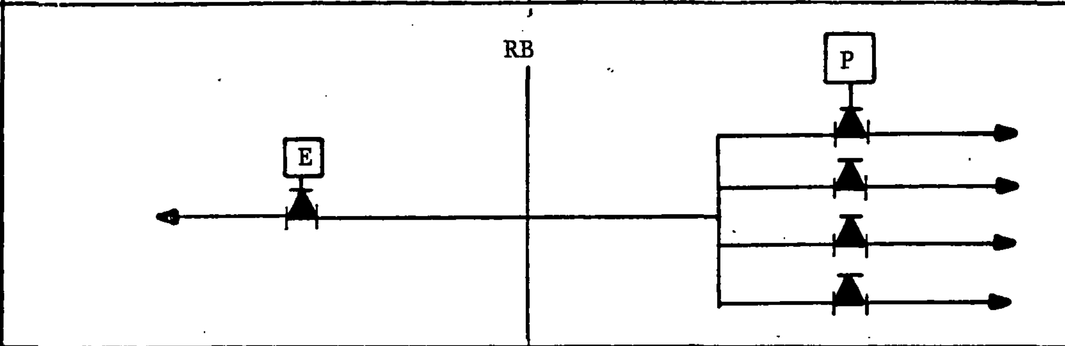
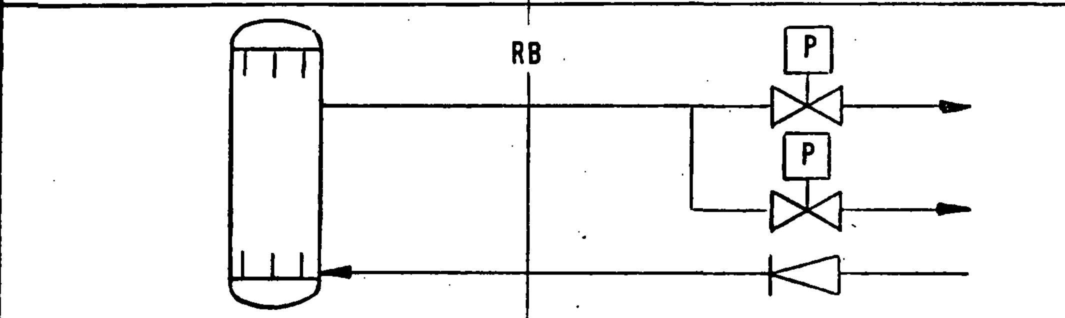
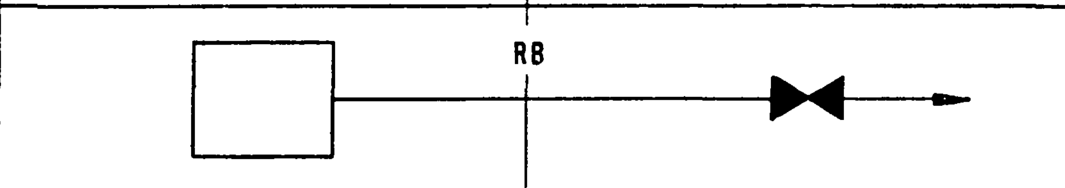
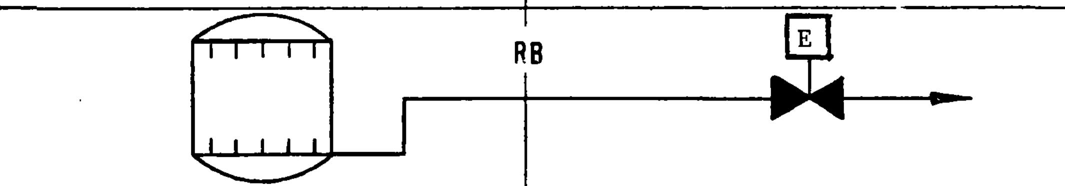
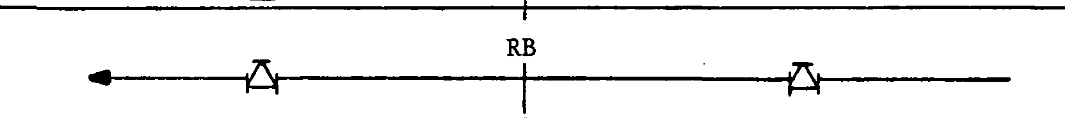
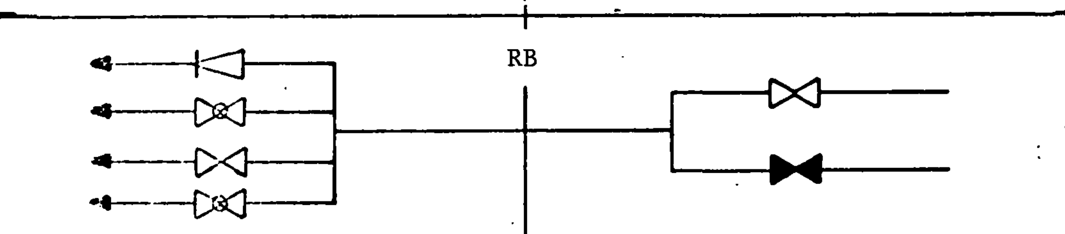
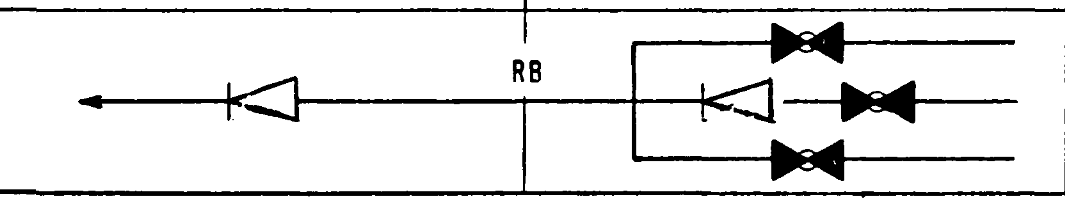
VALVE ARRGT	INSIDE REACTOR BUILDING	OUTSIDE REACTOR BUILDING	PENETRATION NO. AND TYPE
6	13 & 14 TILTING DISC 15 & 16 SWING CK. VALVE 	13&14 GLOBE 15&16 GATE	13, 14 (TYPE II) 15, 16 (TYPE I)
7	BLIND FLANGE 		11, 12 (TYPE IV)
8			17, 50 (TYPE III)
9			1 (TYPE I)
10			52 (TYPE I)
11			19 (TYPE II)
12			20 (TYPE II)

OCONEE NUCLEAR STATION

Figure 5 - 17

REACTOR BUILDING ISOLATION VALVE ARRANGEMENT

Rev. 37 06/03/76

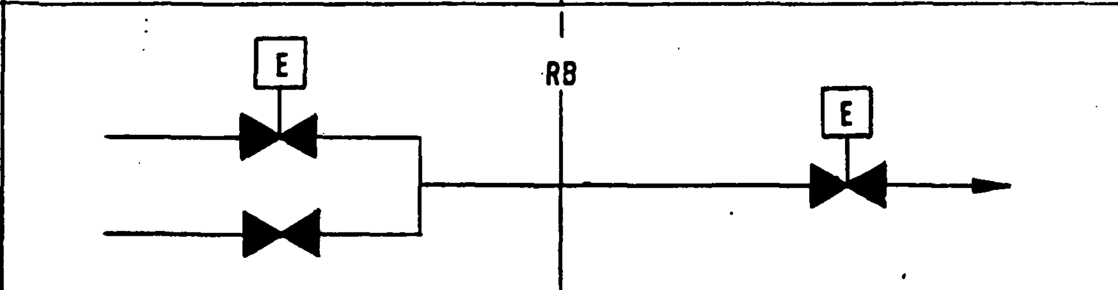
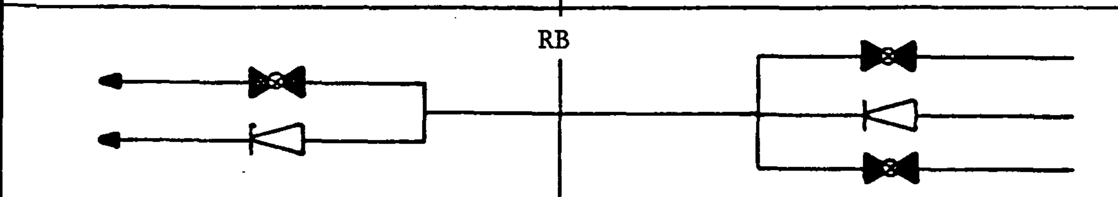
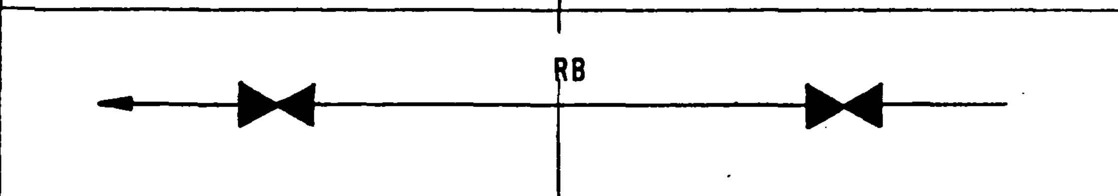
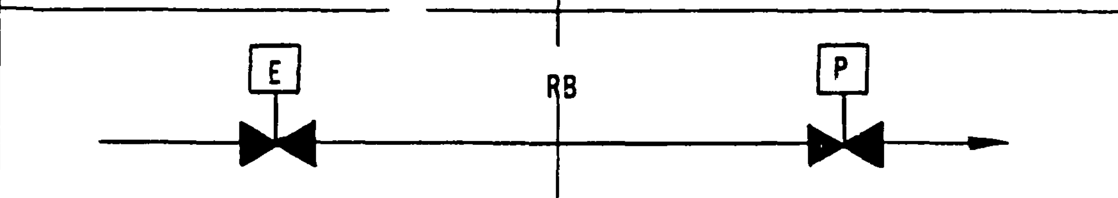
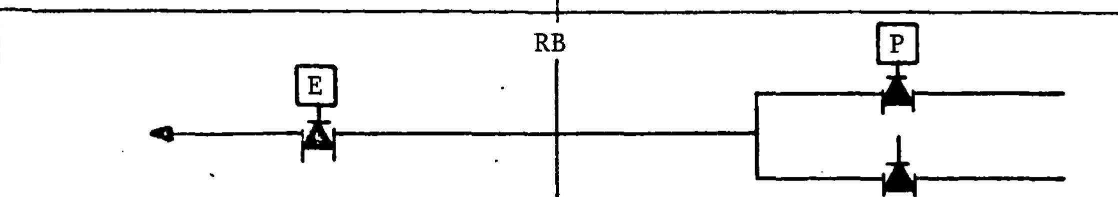
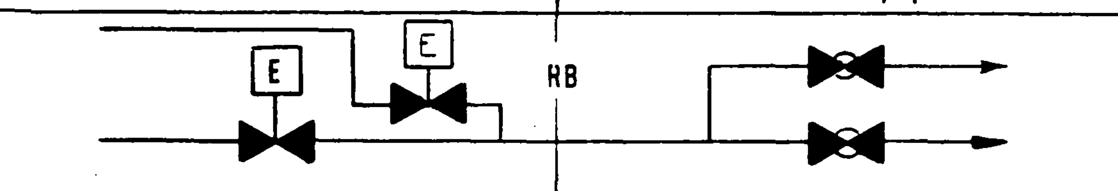
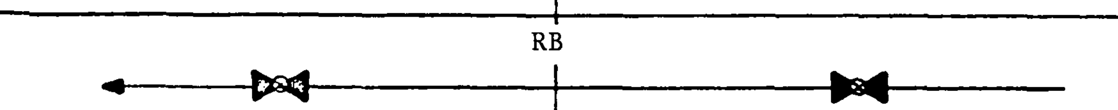
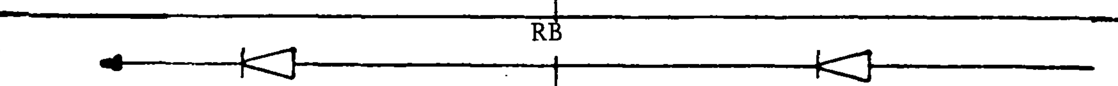
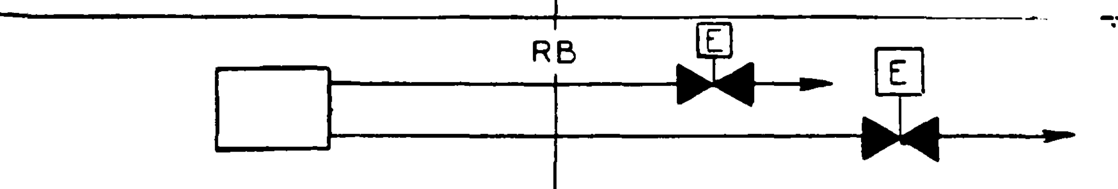
VALVE ARRGT	INSIDE REACTOR BUILDING	OUTSIDE REACTOR BUILDING	PENETRATION NO. AND TYPE
13			60 (TYPE II)
14			25 - 26 27 - 28 INCL. (TYPE III)
15			40 (TYPE IV)
16			4, 43 (TYPE III)
17			46 (TYPE IV)
18			49 (TYPE IV)
19			53 (TYPE IV)

REACTOR BUILDING ISOLATION VALVE ARRANGEMENT

OCONEE NUCLEAR STATION

Figure 5 - 18

Rev. 37 06/03/76

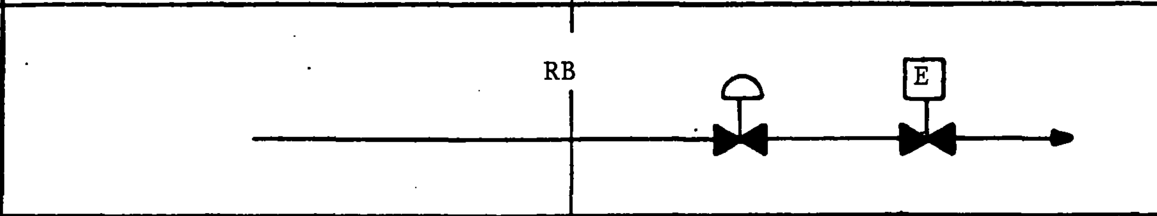
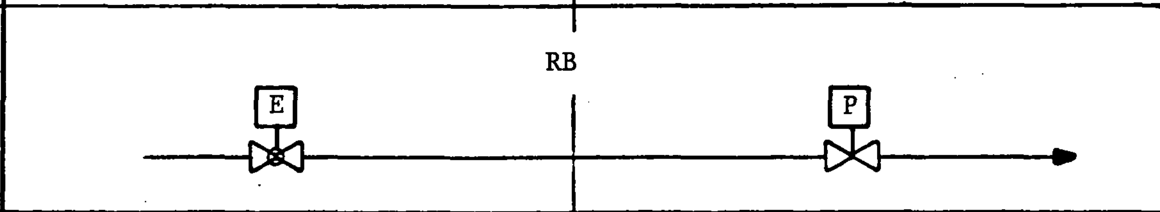
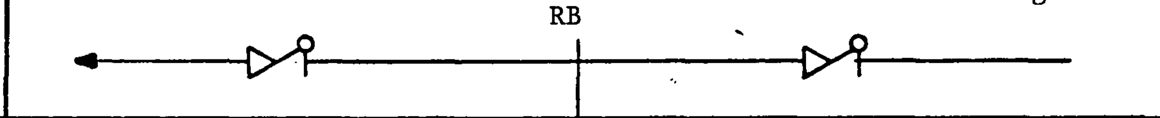
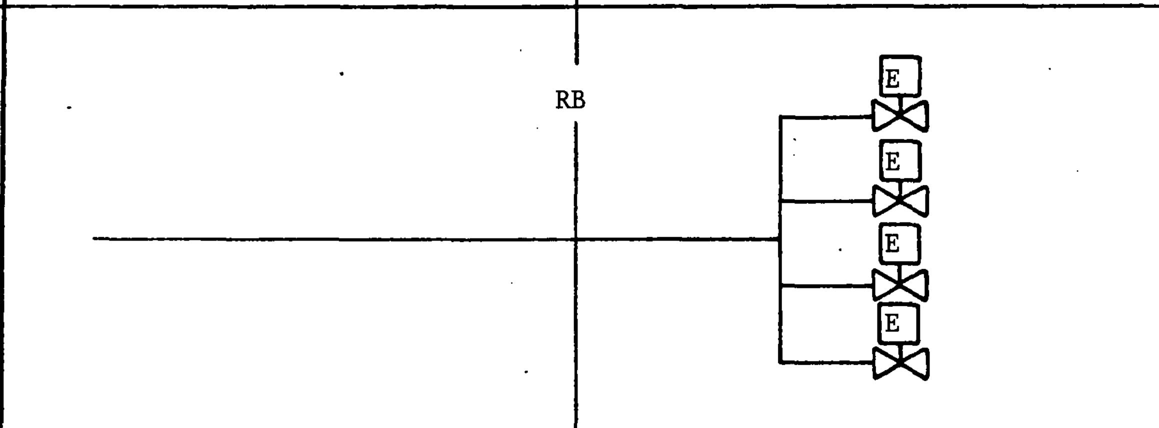
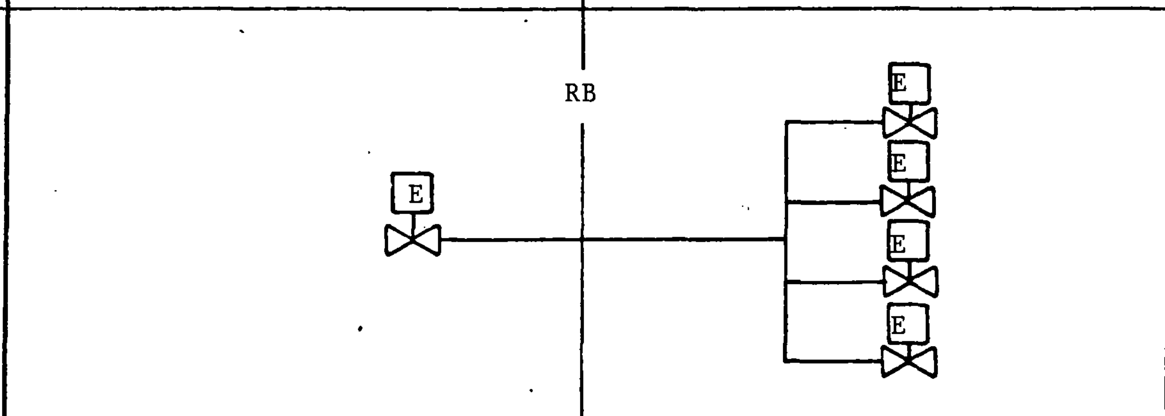
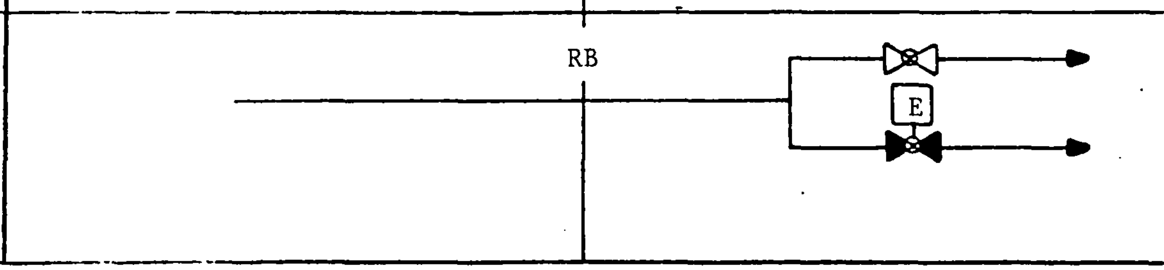
VALVE ARRGT	INSIDE REACTOR BUILDING	OUTSIDE REACTOR BUILDING	PENETRATION NO. AND TYPE
20			57 (TYPE IV)
21			39 (TYPE IV)
22			56 (TYPE IV)
23			54 (TYPE I) 2,58 (TYPE IV)
24			61 (TYPE II)
25			59 (TYPE IV)
26			41, 45 (TYPE II)
27			38 (TYPE II) 47
28			36 37 (TYPE IV)

OCONEE NUCLEAR STATION

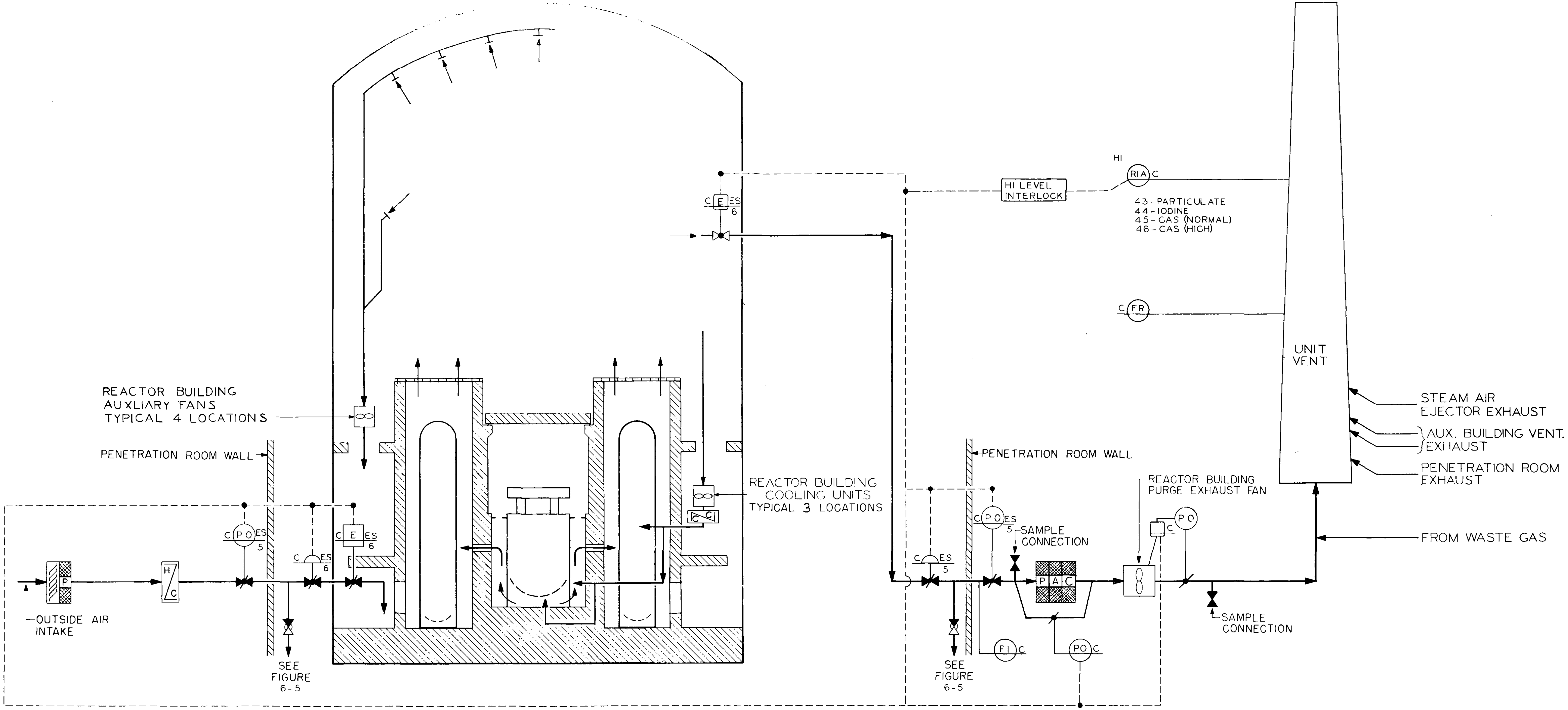
Figure 5 - 19

REACTOR BUILDING ISOLATION VALVE ARRANGEMENT

Rev. 37 06/03/76

VALVE ARRGT	INSIDE REACTOR BUILDING	OUTSIDE REACTOR BUILDING	PENETRATION NO. AND TYPE
29			5 (TYPE II)
30			7 (TYPE I)
31		For #3 Both Valves Swing Ck.	8,10,23,44 (TYPE I) 3 (TYPE III)
32			21 (TYPE III)
33			22 (TYPE III)
34			33 (TYPE III)

REACTOR BUILDING ISOLATION
VALVE ARRANGEMENT



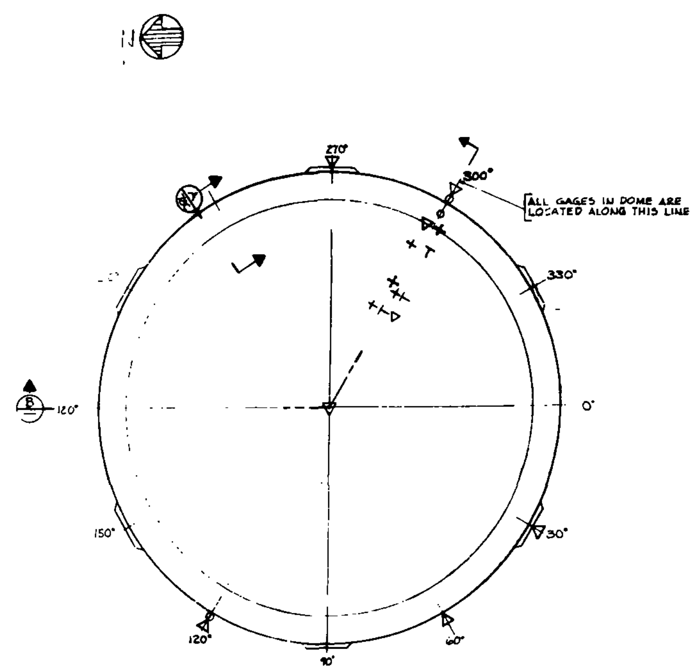
NOTE

FOR LEGEND NOMENCLATURE SEE FIGURE 9-1

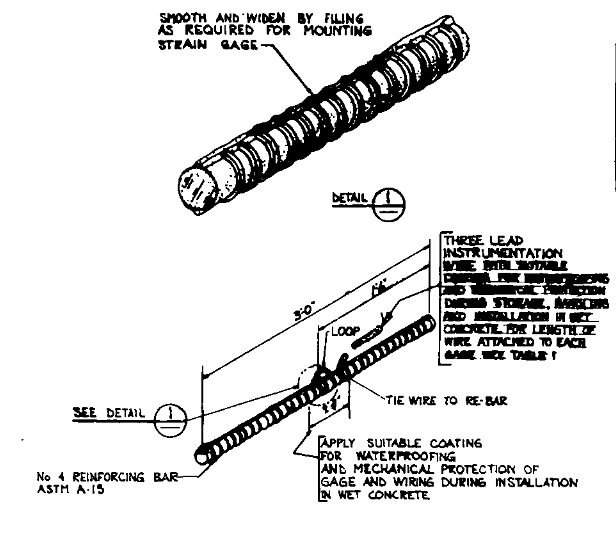
REACTOR BUILDING NORMAL VENTILATION SYSTEM

OCONEE NUCLEAR STATION

Figure 5 - 20



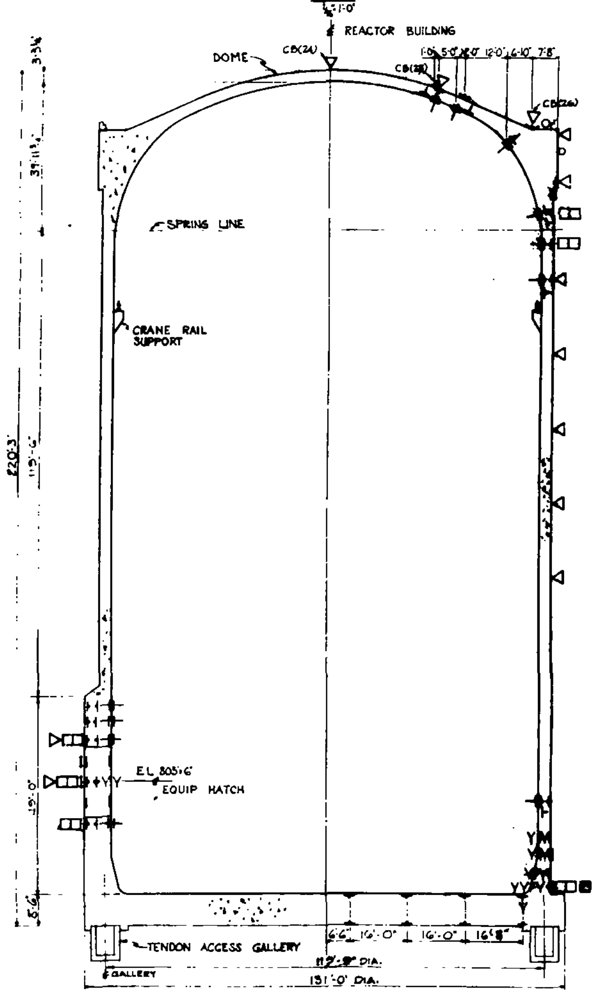
PLAN



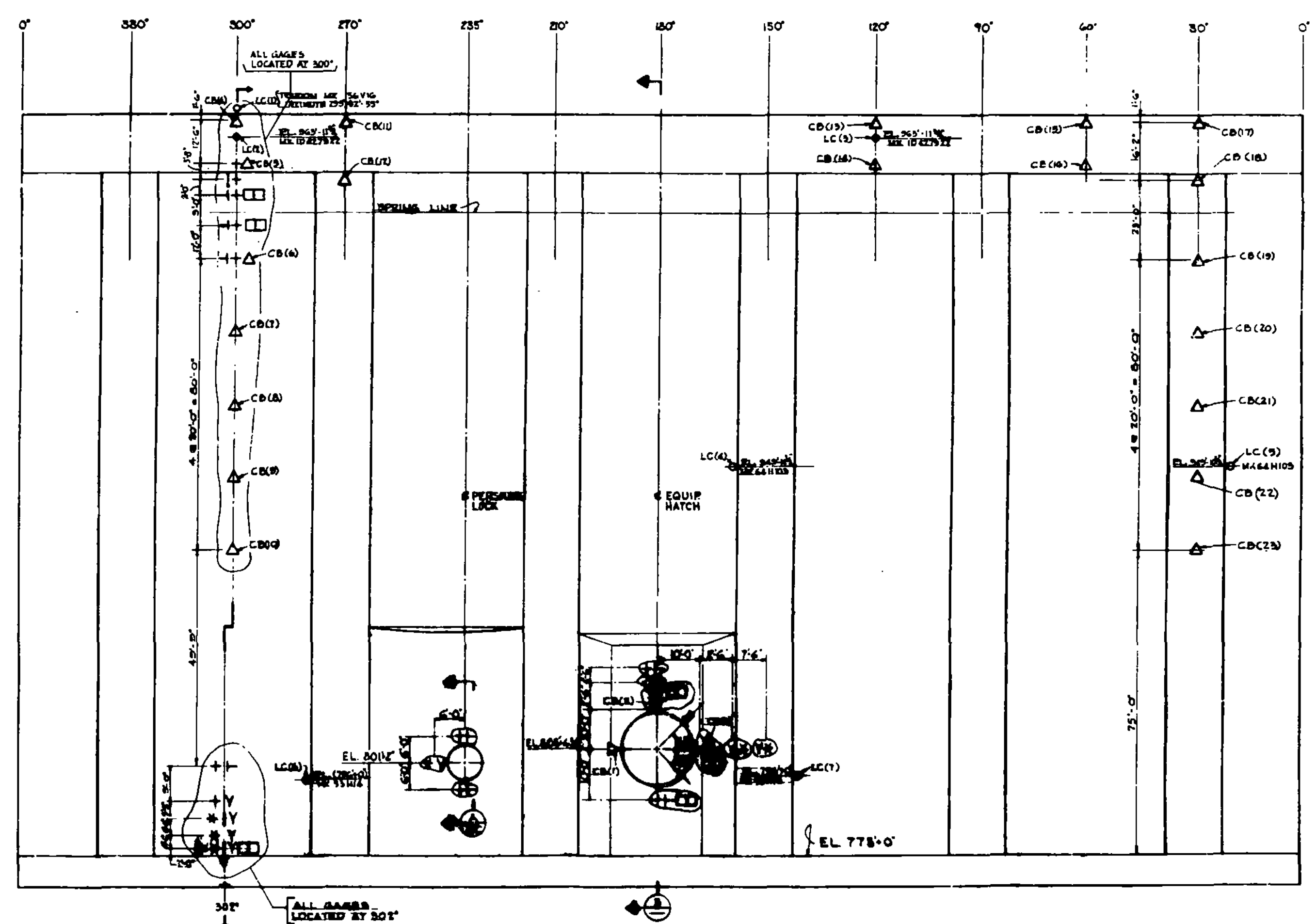
GAGE APPLICATION TO REINFORCING BARS

TABLE 1	
NUMBER OF GAGES	LENGTH OF WIRE FOR EACH GAGE
46	20'-0"
4	32'-0"
2	[illegible]
6	[illegible]
2	60'-0"
6	[illegible]
2	[illegible]

SYMBOL	GAGE DESIGNATION	TAG NO.
— + *	FAET-12C-1256	SGA-1
— +	AS 9-1 (VALORE TYPE)	SGE-2
▭▭	CP-1101EX (BUDD CO.)	SGC-3
Y	FAER-25D-1256	SGR-4
⊥	FAET-25A-1256	SFT-5
△	CALIBRATED BAR	CB
○	LOAD CELL	LC
▨	CEMENT PAINT	—



SECTION B



DEVELOPED ELEVATION OF REACTOR BUILDING

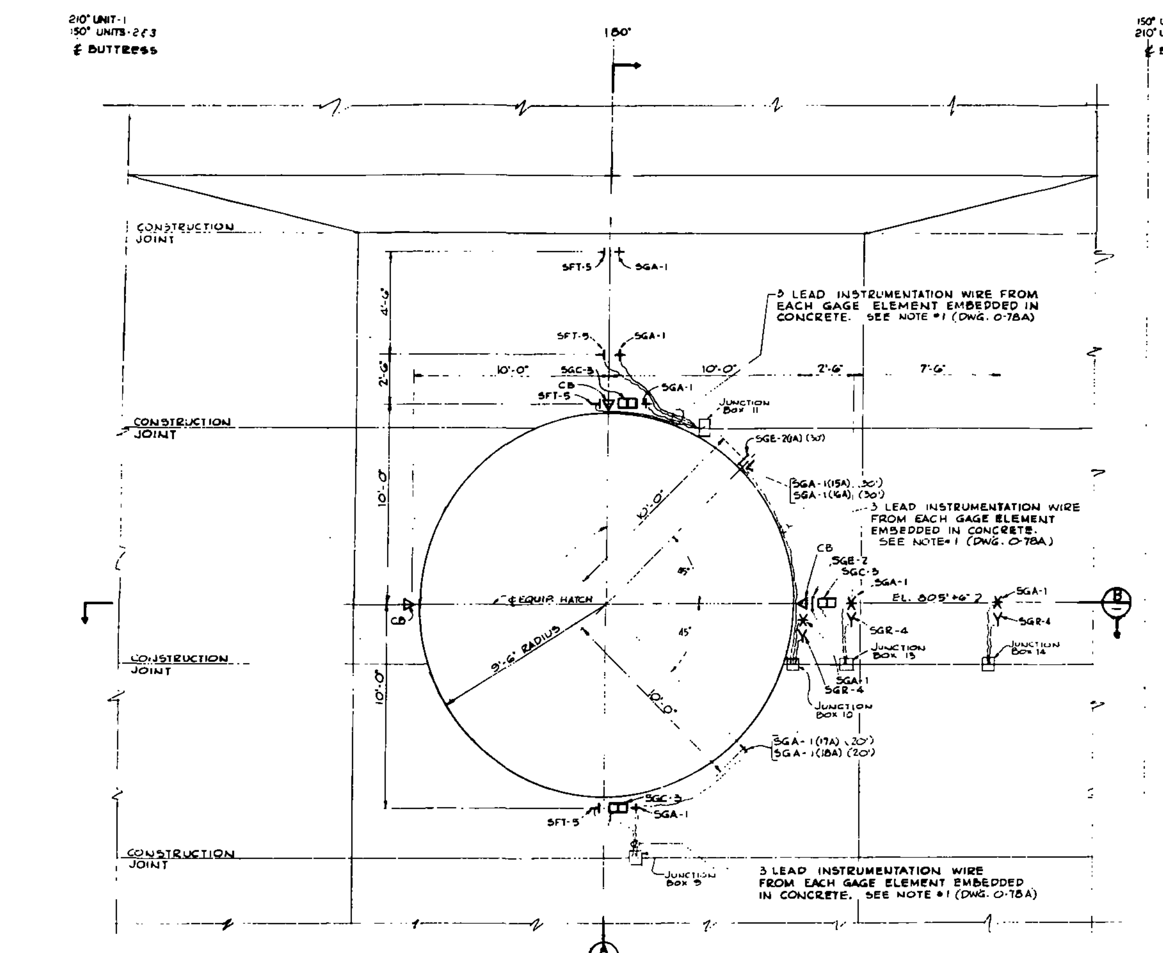
REACTOR BUILDING INSTRUMENTATION FOR UNIT 1

OCONEE NUCLEAR STATION

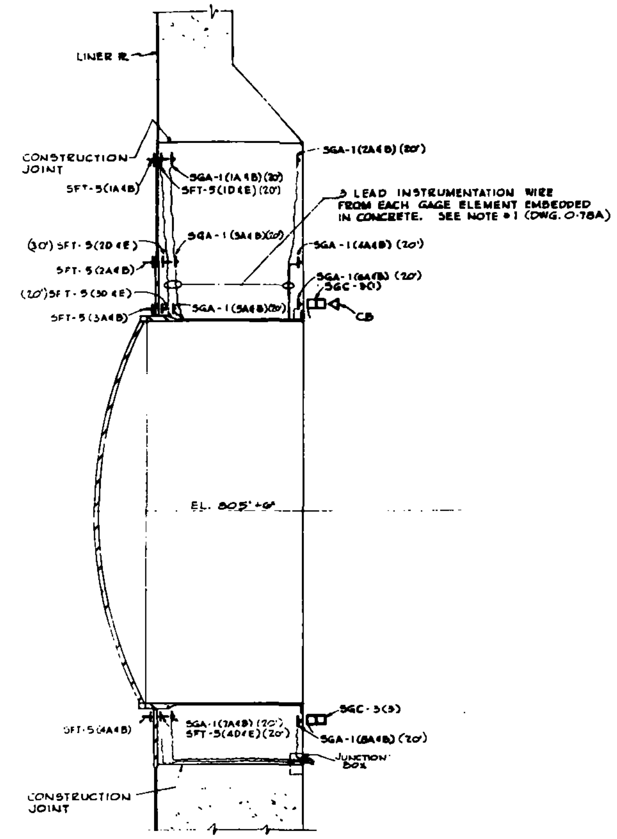
Figure 5 - 21

Sheet 1 of ~~3~~ 4

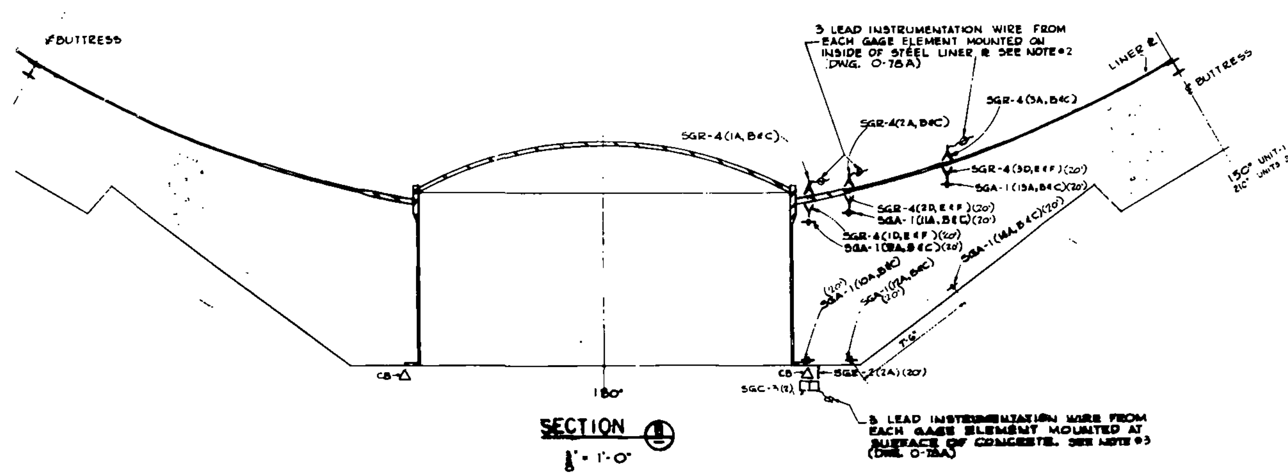
Rev. 5 5/25/70



ELEVATION
EQUIPMENT OPENING
3/8" = 1'-0"



SECTION A
3/8" = 1'-0"



SECTION B
1/8" = 1'-0"

REACTOR BUILDING INSTRUMENTATION FOR UNIT 1

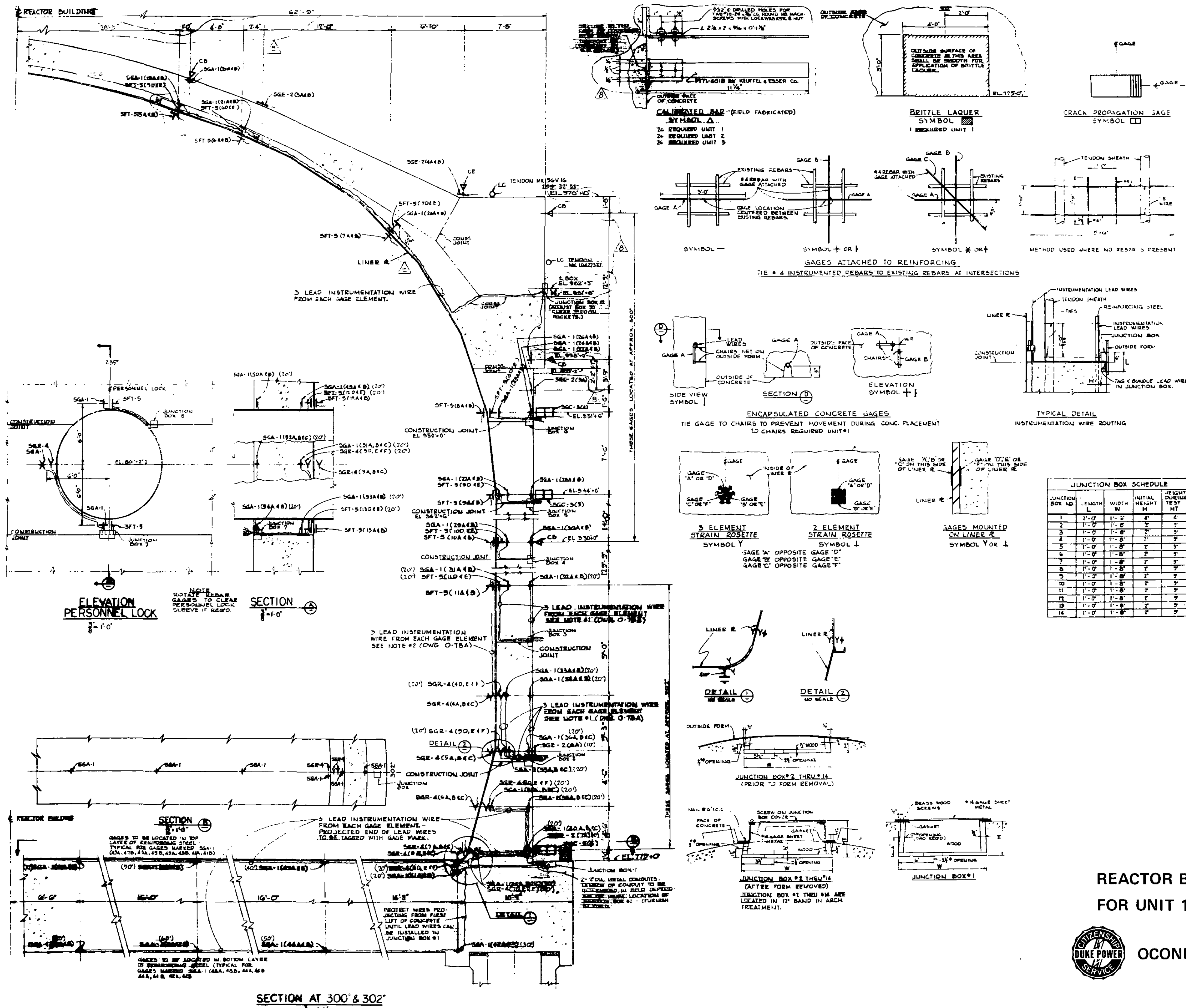
DUKE POWER — CITIZENSHIP SERVICE

OCONEE NUCLEAR STATION

Figure 5 - 21

Sheet 2 of ~~3~~ 4

Rev. 5 5/25/70

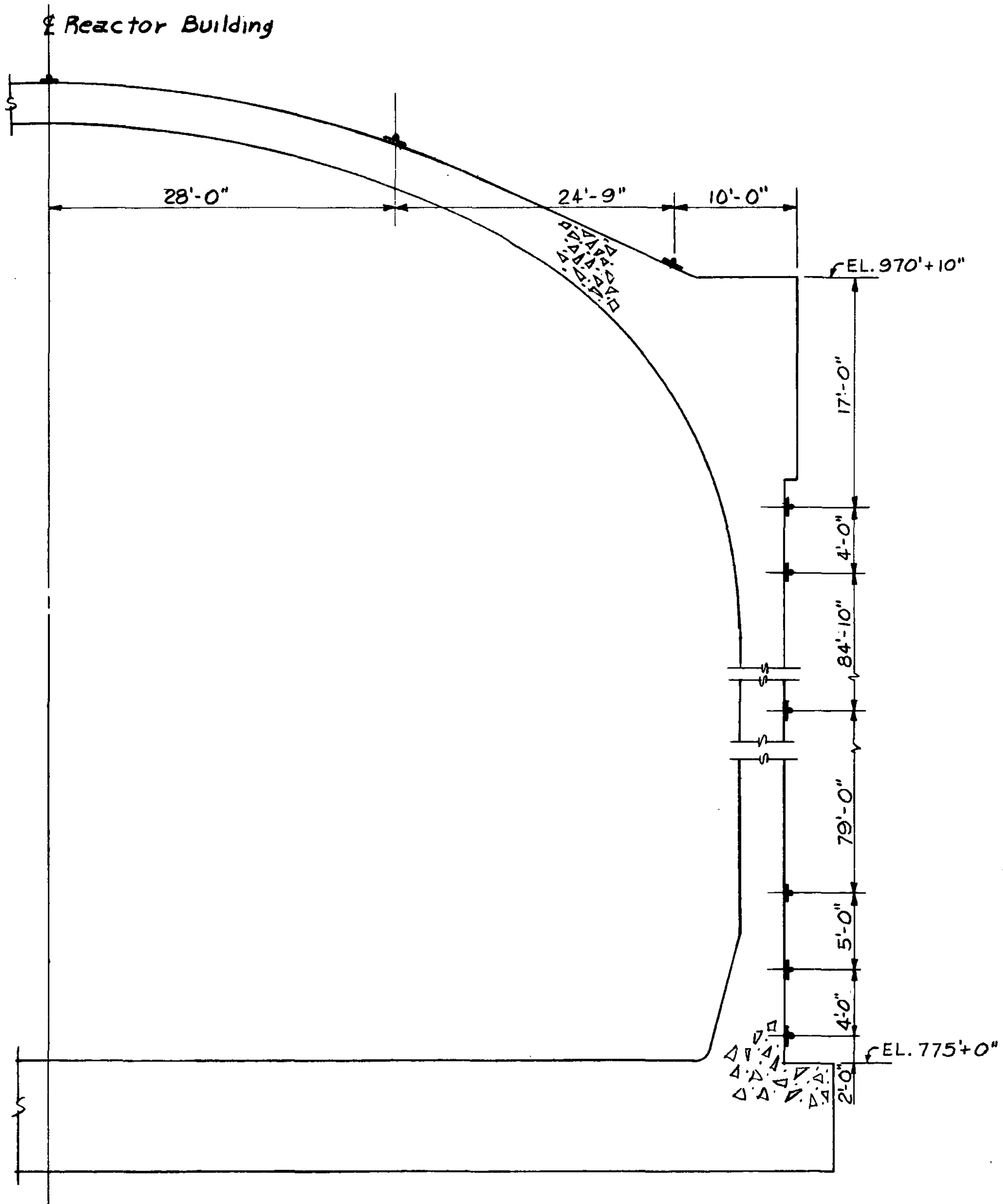


JUNCTION BOX SCHEDULE				
JUNCTION BOX NO.	LENGTH L	WIDTH W	INITIAL HEIGHT H	HEIGHT DURING TEST HT
1	2'-0"	1'-6"	4"	4"
2	1'-0"	1'-8"	2"	9"
3	1'-0"	1'-8"	2"	9"
4	1'-0"	1'-8"	2"	9"
5	1'-0"	1'-8"	2"	9"
6	1'-0"	1'-8"	2"	9"
7	1'-0"	1'-8"	2"	9"
8	1'-0"	1'-8"	2"	9"
9	1'-0"	1'-8"	2"	9"
10	1'-0"	1'-8"	2"	9"
11	1'-0"	1'-8"	2"	9"
12	1'-0"	1'-8"	2"	9"
13	1'-0"	1'-8"	2"	9"
14	1'-0"	1'-8"	2"	9"

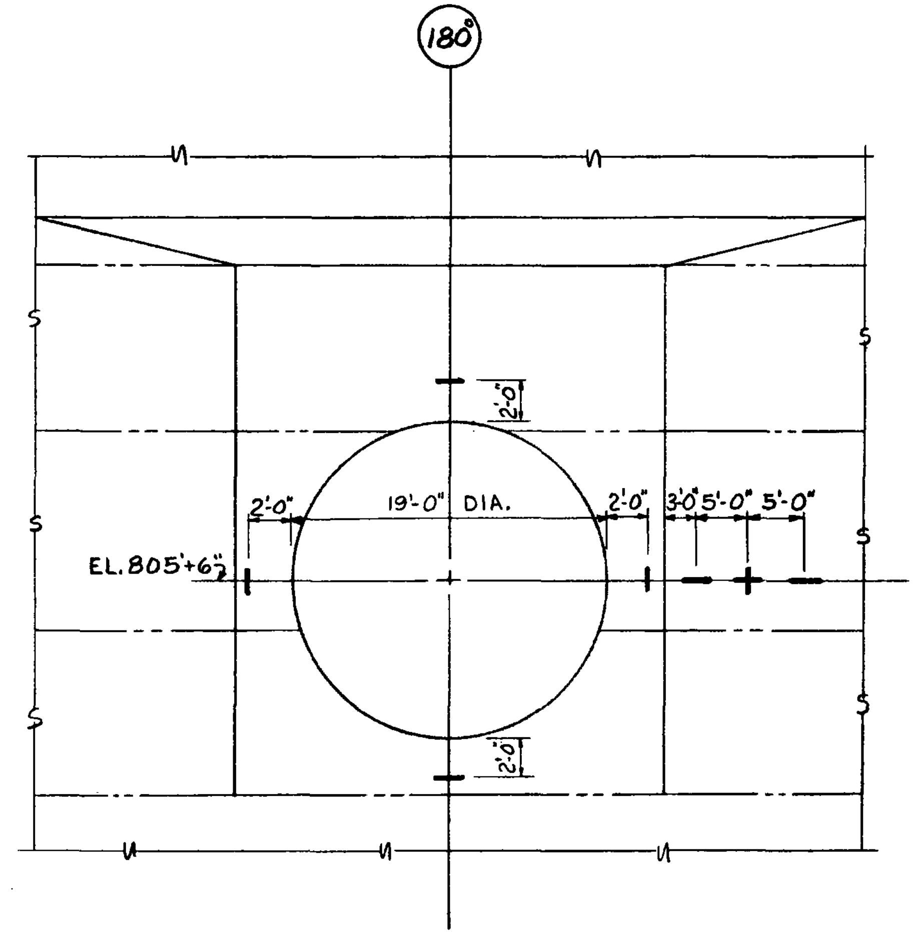
REACTOR BUILDING INSTRUMENTATION FOR UNIT 1

OCONEE NUCLEAR STATION

Figure 5-21
Sheet 3 of 4
Rev. 5 5/25/70



SECTION THROUGH CONTAINMENT
AT 300°



ELEVATION
EQUIPMENT OPENING
No Scale

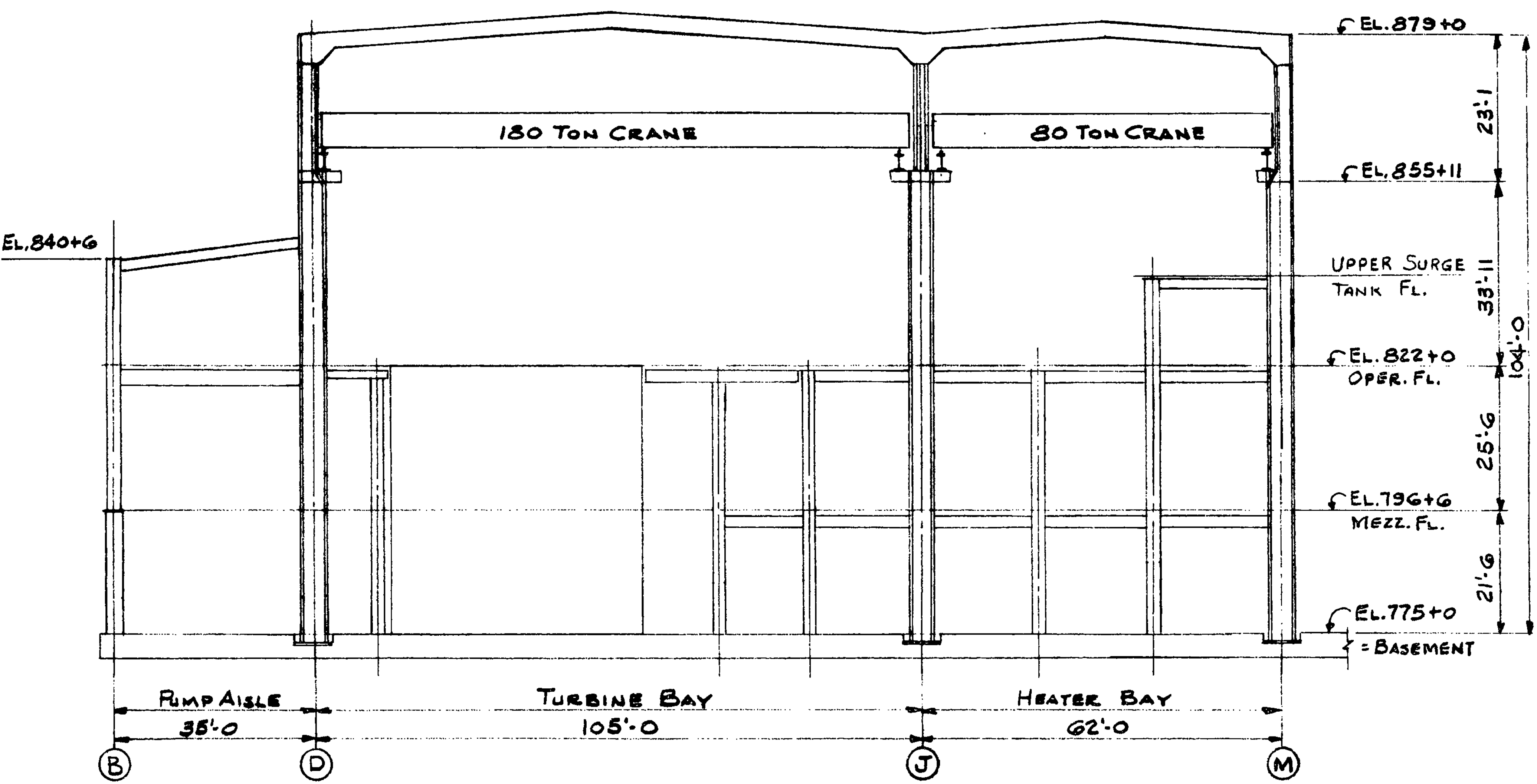
SYMBOL	GAGE DESIGNATION
— ⊥	Carlson SA10S Strain Gage

NOTE:
SEE SHEET 1 FOR ORIENTATION PLAN.

REACTOR BUILDING INSTRUMENTATION
UNIT NO. 1

OCONEE NUCLEAR STATION
Figure 5 - 21
Sheet 4 of 4

(New) Rev. 15 12/30/70



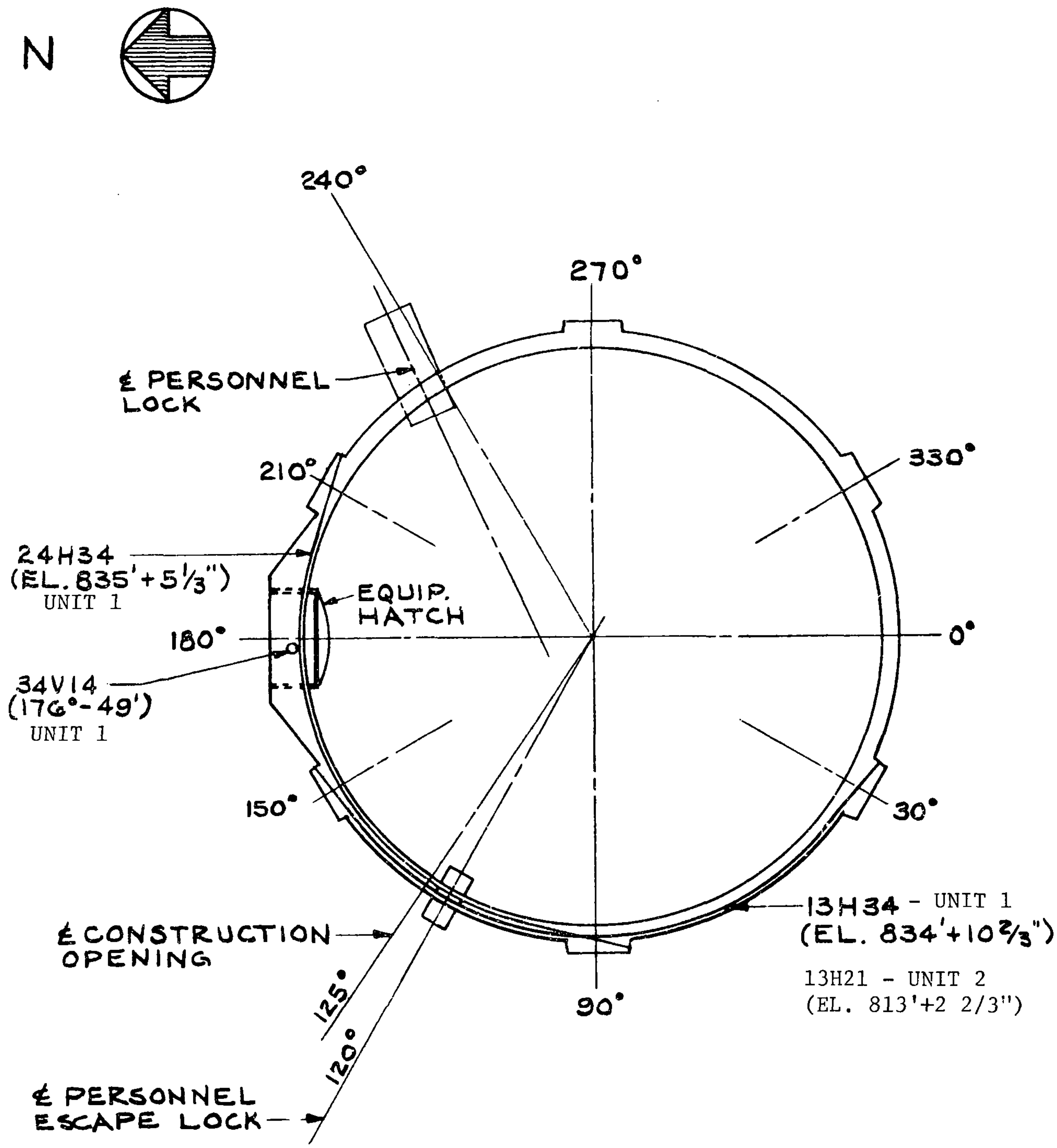
SCALE: 1"=25'

OCONEE NUCLEAR STATION

TURBINE BUILDING CROSS-SECTION AT LINE 21

Figure 5 - 22

(New) Rev. 4 4/20/70



UNIT 1 REACTOR BUILDING as shown
UNITS 2&3 OPPOSITE Hand about 90'-270' ℄

LOCATION OF PLUGGED SHEATHS

OCONEE NUCLEAR STATION

Figure 5 - 23

(New) Rev. 9 8/11/70
Rev. 21. 7/26/72

TABLE 5 - 4 REACTOR BUILDING ISOLATION VALVE INFORMATION

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to R. B.	Valve Type	Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Accident Position
1	Pressurizer Sample Lines	CA	Out	9	Inside	Gate	1/2"	EMO*	ES	Closed	As is	Yes	Closed
						Gate	1/2"	EMO	ES	Closed	As is	Yes	Closed
					Outside	Gate	1/2"	Air	ES	Closed	Closed	Yes	Closed
2	Steam Gen. Sec. Water Sample Line From 1A	CA	Out	22	Inside	Gate	1/2"	EMO	ES	Closed	As is	Yes	Closed
					Outside	Gate	1/2"	Air	ES	Closed	Closed	Yes	Closed
3	Component Cooling Water Inlet Line	CC	In	20	Inside	Tilting Disc Ck	6"	---	---	Open	---	No	Closed
					Outside	Tilting Disc Ck	6"	---	---	Open	---	No	Closed
4	Generator Drain Line	SS	Out	16	Outside	Gate	4"	Air	ES	Closed	Closed	Yes	Closed
5	Reactor Bldg. Normal Sump Drain Line	WD	Out	25	Outside	Gate	2"	EMO	ES Rad. Monit.	Closed	As is	Yes	Closed
					Outside	Gate	2"	Air	ES Rad. Monit.	Closed	Closed	Yes	Closed
6	Let Down Line to Purification Demineralizers	HP	Out	4	Inside	Globe	2-1/2"	EMO	ES	Open	As is	Yes	Closed
						Globe	2-1/2"	EMO	ES	Open	As is	Yes	Closed
					Outside	Globe	2-1/2"	Air	ES	Open	Open	Yes	Closed
7	Reactor Coolant Pump Seal Outlet Line	HP	Out	22	Inside	Gate	4"	EMO	ES	Open	As is	Yes	Closed
					Outside	Gate	4"	Air	ES	Open	Open	Yes	Closed
8	Reactor Coolant Pump Seal Inlet Line	HP	In	17	Inside	Stop CK (4)	4"	Manual	---	Open	---	No	Closed
					Outside	Globe	4"	Air	dP	Throttled	Closed	No	Closed
						Globe	2-1/2"	Air	dP	Throttled	Closed	No	Closed
						Globe	4"	Manual	---	losed	---	No	Closed
9	Normal Makeup to the Reactor Coolant System	HP	In	5	Inside	Stop Ck(2)	2-1/2"	Manual	---	---	---	No	---
					Outside	Globe	4"	EMO	ES	Closed	As is	Yes	Open
						Globe	2-1/2"	Air	RC Level	Throttled	Closed	No	Open
						Globe	2-1/2"	Manual	---	Closed	---	No	Closed

* All valves with electric motor operators are also equipped with handwheels.

TABLE 5 - 4 REACTOR BUILDING ISOLATION VALVE INFORMATION (Continued)

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to R. B.	Valve Type	Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Accident Position
11 12	Fuel Transfer Tubes	SF	In/Out	7	Inside	Special Closure	30"	---	---	Closed	---	No	---
					Outside	Gate	30"	Manual	---	Closed	---	Yes	Closed
13	Reactor Bldg Spray Inlet Line	BS	In	6	Inside	Tilting Disc Ck.	8"	---	---	---	---	No	---
					Outside	Globe	8"	EMO	ES	Closed	As is	Yes	Open
14	Reactor Bldg. Spray Inlet Line	BS	In	6	Inside	Tilting Disc.Ck.	8"	---	---	---	---	No	---
					Outside	Globe	8"	EMO	ES	Closed	As is	Yes	Open
15	Low Pressure Injection and Decay Heat Removal Line	LP	In	6	Inside	Swing Ck	10"	---	---	---	---	No	---
					Outside	Gate	10"	EMO	ES	Closed	As is	Yes	Open
16	Low Pressure Injection and Decay Heat Removal Line	LP	In	6	Inside	Swing Ck	10"	---	---	---	---	No	Open
					Outside	Gate	10"	EMO	ES	Closed	As is	Yes	Open
17	Emergency Feedwater Inlet Line	SS	In	8	Outside	Tilting Disc Ck	6"	---	---	---	---	No	---
					Inside	Tilting Disc Ck	6"	---	---	---	---	No	---
18	Quench Tank Vent Line	WD	Out	3	Inside	Gate	2"	EMO	ES	Closed	As is	Yes	Closed
					Outside	Gate	2"	Air	ES	Closed	Closed	Yes	Closed
19	Reactor Bldg. Inlet Purge Line	PR	In	11	Inside	Butterfly	48"	EMO	ES Rad. Monit.	Closed	As is	Yes	Closed
					Outside	Butterfly	48"	Air	ES Rad. Monit.	Closed	Closed	Yes	Closed
20	Reactor Bldg. Outlet Purge Line	PR	Out	12	Inside	Butterfly	48"	EMO	ES Rad. Monit.	Closed	As is	Yes	Closed
					Outside	Butterfly	48"	Air	ES Rad. Monit.	Closed	Closed	Yes	Closed
21 22	Reactor Coolant Pump Motors and Lube Oil	LPSW	In/OUT	13	Inside	Gate (4)	4"	EMO	Remote Manual	Open	As is	Yes	Open
					Outside	Gate	10"	EMO	ES	Open	As is	Yes	Closed
					Inside	Gate (4)	4"	EMO	Remote Manual	Open	As is	Yes	Open
					Outside	Gate	10"	EMO	ES	Open	As is	Yes	Closed

TABLE 5 - 4 REACTOR BUILDING ISOLATION VALVE INFORMATION

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to RB	Valve Type	Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Accident Position
1	Pressurizer Sample Lines	CA	Out	9	Inside	Gate	½"	EMO*	ES	Closed	As is	Yes	Closed
						Gate	½"	EMO	ES	Closed	As is	Yes	Closed
					Outside	Gate	½"	Air	ES	Closed	Closed	Yes	Closed
2	Steam Gen. Sec. Water Sample Line	CA	Out	23	Inside	Gate	½"	EMO	ES	Closed	As is	Yes	Closed
	From 1A				Outside	Gate	½"	Air	ES	Closed	Closed	Yes	Closed
3	Component Cooling Water	CC	In	31	Inside	Swing Ck	6 "	---	--	---	----	No	Closed
	Inlet Line				Outside	Swing Ck	6 "	---	---	---	----	No	Closed
4	Generator Drain Line	SS	Out	16	Outside	Gate	4 "	EMO	ES	Closed	Closed	Yes	Closed
5	RB Normal Sump Drain	WD	Out	29	Outside	Gate	2 "	EMO	ES Rad. Monit.	Closed	As is	Yes	Closed
	Line				Outside	Gate	2 "	Air	ES Rad. Monit.	Closed	Closed	Yes	Closed
6	Let Down Line to Purification	HP	Out	4	Inside	Globe	2½"	EMO	ES	Open	As is	Yes	Closed
						Globe	2½"	EMO	ES	Open	As is	Yes	Closed
	Demineralizers				Outside	Globe	2½"	Air	ES	Open	Open	Yes	Closed
7	RC Pump Seal	HP	Out	30	Inside	Globe	4 "	EMO	ES	Open	As is	Yes	Closed
	Outlet Line				Outside	Gate	4 "	Air	ES	Open	Open	Yes	Closed
8	Loop A_1, A_2 Nozzle Warming	HP	In	31	Inside	Stop Ck	1 "	---	---	Open	---	No	Closed
	Lines				Outside	Stop Ck							
9	Normal Makeup	HP	In	5	Inside	Swing Ck	4 "	---	---	---	---	No	----
	to the RC System				Outside	Globe	4 "	EMO	ES	Closed	As is	Yes	Open
						Globe	2½"	Air	RC Level	Throttled	Closed	No	Open
						Globe	2½"	Manual	---	Closed	---	No	Closed
10	RCP Seal Injection Lines	HP	In	31	Inside	Stop Ck	1½"	---	---	Open	---	No	Closed
					Outside	Stop Ck	1½"						

* All valves with electric motor operators are also equipped with handwheels.

TABLE 5 - 4 REACTOR BUILDING ISOLATION VALVE INFORMATION (Continued)

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to RB	Valve Type	Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Accident Position
11 12	Fuel Transfer Tubes	SF	In/Out	7	Inside	Special Closure	30 "	---	---	Closed	---	No	---
					Outside	Gate	30 "	Manual	---	Closed	---	Yes	Closed
13	RB Spray Inlet Line	BS	In	6	Inside	Tilting Disc Ck	8 "	---	---	---	---	No	---
					Outside	Globe	8 "	EMO	ES	Closed	As is	Yes	Open
14	RB Spray Inlet Line	BS	In	6	Inside	Tilting Disc Ck	8 "	---	---	---	---	No	---
					Outside	Globe	8 "	EMO	ES	Closed	As is	Yes	Open
15	LPI and Decay Heat Removal Line	LP	In	6	Inside	Swing Ck	10 "	---	---	---	---	No	---
					Outside	Gate	10 "	EMO	ES	Closed	As is	Yes	Open
16	LPI and Decay Heat Removal Line	LP	In	6	Inside	Swing Ck	10 "	---	---	---	---	No	Open
					Outside	Gate	10 "	EMO	ES	Closed	As is	Yes	Open
17	Emergency Feedwater Inlet Line	SS	In	8	Outside	Tilting Disc Ck	6 "	---	---	---	---	No	---
					Inside	Tilting Disc Ck	6 "	---	---	---	---	No	---
18	Quench Tank Vent Line	WD	Out	3	Inside	Gate	2 "	EMO	ES	Closed	As is	Yes	Closed
					Outside	Gate	2 "	Air	ES	Closed	Closed	Yes	Closed
19	RB Inlet Purge Line	PR	In	11	Inside	Butterfly	48 "	EMO	ES Rad. Monit.	Closed	As is	Yes	Closed
					Outside	Butterfly	48 "	Air	ES Rad. Monit.	Closed	Closed	Yes	Closed
20	RB Outlet Purge Line	PR	Out	12	Inside	Butterfly	48 "	EMO	ES Rad. Monit.	Closed	As is	Yes	Closed
					Outside	Butterfly	48 "	Air	ES Rad. Monit.	Closed	Closed	Yes	Closed
21 22	RCP Motors and Lube Oil	LPSW	In/Out	32	Outside	Gate (4)	4 "	EMO	Remote Manual	Open	As is	Yes	Open
				33	Inside	Gate	10 "	EMO	ES	Open	As is	Yes	Closed
					Outside	Gate (4)	4 "	EMO	Remote Manual	Open	As is	Yes	Open
					Outside	Gate	10 "	EMO	ES	Open	As is	Yes	Closed
23	RCP Seal Injection Line	HP	In	31	Inside	Stop Ck	1½"	---	---	Open	---	No	Closed
					Outside	Stop Ck	1½"						

TABLE 5 - 4 REACTOR BUILDING ISOLATION VALVE INFORMATION (Continued)

Penetra-tion No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to RB	Valve Type	Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Accident Position
25 26	Feedwater and Steam Lines	FDW MS	In/Out	14	Outside	Tilting Disc Ck	24 "	---	---	---	---	No	---
					Outside	Turb Stop	24 "	Hydr	Turb Prot System	Open	Closed	Yes	Closed
27 28	Feedwater and Steam Lines	FDW MS	In/Out	14	Outside	Tilting Disc Ck	24 "	---	---	---	---	No	---
					Outside	Turb Stop	24 "	Hydr	Turb Prot System	Open	Closed	Yes	Closed
29	Quench Tank Drain Line	WD	Out	3	Inside	Gate	4 "	EMO	ES	Closed	As is	Yes	Closed
					Outside	Gate	2 "	Air	ES	Closed	Closed	Yes	Closed
30 31 32	RB Emergency Cooler Inlet Line	LPSW	In	2	Outside	Gate	8 "	EMO	Remote Manual	Open	As is	Yes	Open
33	RB Emergency Cooler Outlet Line	LPSW	Out	34	Outside	Globe	8 "	Manual	---	Open	---	---	---
					Outside	Globe	8 "	EMO	ES	Closed	As is	Yes	Open
34		LPSW	Out	1	Outside	Globe	8 "	Manual	---	Open	---	---	---
35					Outside	Globe	8 "	EMO	ES	Open	As is	Yes	Open
36 37	RB Sump Recirc Line	LP	Out	28	Outside	Gate	14 "	EMO	Remote Manual	Closed	As is	Yes	Open
38	Quench Tank Cooler Inlet Line	WD	In	27	Inside	Tilting Disc Ck	2 "	---	---	---	---	No	---
					Outside	Tilting Disc Ck	2 "					No	
39	Nitrogen Supply Line	CF	In	21	Inside	Globe	1 "	Manual	---	Closed	---	---	---
						Tilting Disc Ck	1 "	---	---	---	---	---	---
					Outside	Globe	½"	Manual	---	Closed	---	---	---
						Globe	1 "	Manual	---	Closed	---	---	---
						Tilting Disc Ck	1 "	---	---	---	---	---	---
40	RB Emergency Sump Drain	WD	Out	15	Outside	Gate	2 "	Manual	---	Closed	---	Yes	Closed
41	IA System	IA	In	26	Inside	Globe	1 "	Manual	---	Closed	---	No	As is
					Outside	Globe	1 "	Manual	---	Closed	---	No	As is
43	Generator Drain Line	SS	Out	16	Outside	Gate	4 "	EMO	ES	Closed	Closed	Yes	Closed
44	Control Rod Drive	CC	In	31	Inside	Swing Ck	3 "	---	---	---	As is	No	Closed
					Outside	Swing Ck	3 "	---	---	---	As is	No	Closed
45	LRT System	LRT	In	26	Inside	Globe	6 "	Manual	---	Closed	---	No	As is
					Outside	Globe	6 "	Manual	---	Closed	---	No	As is
46	Reactor Head Wash System	FW	In	17	Inside	Saunder's Pat	6 "	Manual	---	Open	---	No	As is
					Outside	Saunder's Pat	6 "	Manual	---	Open	---	No	As is

TABLE 5 - 4 REACTOR BUILDING ISOLATION VALVE INFORMATION (Continued)

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to RB	Valve Type	Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Accident Position
47	RCP Seal Vents	DW	In	27	Inside	Tilting Disc Ck	½"	---	---	---	---	No	As is
					Outside	Tilting Disc Ck	½"	---	---	---	---	No	As is
48	BA System	BA	In	22	Inside	Gate	2 "	Manual	---	Closed	As is	No	As is
					Outside	Gate	2 "	Manual	---	Closed	As is	No	As is
49	RC System	N	In	18	Inside	Gate	2 "	Manual	---		As is	No	As is
						Globe	2 "						
						Globe	1½"						
						TD Ck	1 "						
					Outside	Gate (2)	2 "						
50	Emergency Feedwater Inlet Line	FDW	In	8	Outside	Tilting (2) Disc Ck	6 "	---	---	---	---	No	---
					Inside	Tilting Disc Ck	6 "	---	---	---	---	No	---
52	Emergency Reactor Injection Line	HP	In	10	Inside	Swing Ck	4 "	---	---	---	Open	No	Open
					Outside	Globe	4 "	EMO	ES	Closed	As is	Yes	Open
53	Nitrogen Supply Line	CF	In	19	Inside	Tilting Disc Ck	1 "	---	---	---	---	No	---
					Outside	Tilting Disc Ck	1 "	---	---	---	---	No	---
						Globe (3)	1 "	Manual	---	Closed	---	No	Closed
54	Component Cooling Water Outlet Line	CC	Out	23	Inside	Gate	8 "	EMO	ES	Open	As is	Yes	Closed
					Outside	Gate	8 "	Air	ES	Open	Open	Yes	Closed
56	Canal Fill and Drain Line	SF	In	22	Inside	Gate	8 "	Manual	---	Closed	---	Yes	Closed
					Outside	Gate	8 "	Manual	---	Closed	---	Yes	Closed
57	Decay Heat or Fuel Trans. Canal Outlet Line	LP	Out	20	Outside	Gate	12 "	EMO	Remote Manual	Closed	As is	Yes	Closed
					Inside	Gate	10 "	Manual	---	Closed	---	Yes	Closed
					Inside	Gate	12 "	EMO	---	Closed	As is	Yes	Closed
58	Steam Gen. Sec. Water Sample Line From 1B	CA	Out	23	Inside	Gate	½"	EMO	ES	Closed	As is	Yes	Closed
					Outside	Gate	½"	Air	ES	Closed	Closed	Yes	Closed
60	RB Sampling System Outlet (Rad. Monit.)	PR	Out	13	Inside	Saunder's Pat	2 "	EMO	ES	Closed	As is	Yes	Closed
					Outside	Saunder's Pat(3)	1 "	Manual	---	Closed	---	---	---
					Outside	Saunder's Pat	2 "	Air	ES	Closed	Open	Yes	Closed
61	RB Sampling System Inlet (Rad. Monit.)	PR	In	24	Inside	Saunder's Pat	2 "	EMO	ES	Closed	As is	Yes	Closed
					Outside	Saunder's Pat	1 "	Air	ES	Closed	Open	Yes	Closed
					Outside	Saunder's Pat	2 "	Manual	---	Closed	---	---	---
59	CF Tanks Sampling	CF	Out	25	Inside	Gate (2)	1 "	EMO	---	Closed	As is	Yes	Closed
					Outside	Globe (2)	1 "	Manual	---	Closed	---	---	Closed

LIST OF EFFECTIVE PAGES
FSAR APPENDIX 5A

Structural Design Bases

TABLE OF CONTENTS

APPENDIX 5A

5A STRUCTURAL DESIGN BASES

The design bases for normal operating conditions are governed by the applicable building design codes. The basic design criterion for the worst loss-of-coolant accident and seismic conditions is that there shall be no loss of function if that function is related to public safety.

AEC publication TID 7024, "Nuclear Reactors and Earthquake," as amplified herein will be used as the basic design guide for seismic analysis.

5A.1 CLASS OF STRUCTURES

The plant structures will be classified according to their function and the degree of integrity required to protect the public. These classes are:

5A.1.1 CLASS 1

Class 1 structures are those which prevent uncontrolled release of radioactivity and are designed to withstand all loadings without loss of function. Class 1 structures include the following:

> Portions of the Auxiliary Building that house engineered safeguards systems, control room, fuel storage facilities and radioactive materials.
>
> Reactor Building and its penetrations.
>
> Polar Crane (unloaded condition).
>
> Unit Vent.

5A.1.2 CLASS 2

Class 2 structures are those whose limited damage would not result in a release of radioactivity and would permit a controlled plant shutdown but could interrupt power generation. Examples of Class 2 structures are Intake Structure, Turbine and Auxiliary Buildings, except as included in Class 1.

5A.1.3 CLASS 3

Class 3 structures are those whose failure could inconvenience operation, but which are not essential to power generation, orderly shutdown or maintenance of the reactor in a safe condition. They include all structures not included in Classes 1 and 2.

5A.2 DESIGN BASES FOR CLASS 1 STRUCTURES

5A.2.1 NORMAL OPERATION

For loads experienced in normal plant operation, Class 1 structures are designed in accordance with design methods of accepted standards and applicable codes.

5A.2.2 ACCIDENT, WIND AND SEISMIC CONDITIONS

The Class 1 structures are proportioned to maintain elastic behavior when subjected to various combinations of dead loads, accident loads, thermal loads and wind or seismic loads. The upper limit of elastic behavior is considered to be the yield strength of the effective load-carrying structural materials. The yield strength for steel (including reinforcing steel) is considered to be the minimum given in the appropriate ASTM specification. Concrete structures are designed for ductile behavior wherever possible; that is, with steel stress controlling the design. The values for concrete, as given in the ultimate strength design portion of the ACI 318-63 Code, will be used in determining "Y", the required yield strength of the structure.

The design loads applied to the structures are increased by load factors based on the probability and conservatism of the predicted normal design loads.

The final design of Class 1 structures satisfies the following loading combinations and factors:

$Y = 1/\phi\ (1.0D + 1.0P + 1.0T + E')$
$Y = 1/\phi\ (1.05D + 1.25P + 1.0T + 1.25E \text{ or } W)$
$Y = 1/\phi\ (1.05D + 1.5P + 1.0T)$
$Y = 1/\phi\ (1.0D + 1.0W_t + 1.0P_i)$ for Tornado Forces.
(Use 0.95 where dead load subtracts from critical stress.)
(Wind, W, to replace earthquake, E, in the above formula where wind stresses control)

Where Y = required yield strength of the structure as defined above.
- D = dead loads of structure and equipment plus any other permanent loadings contributing stress, such as hydrostatic or soils. In addition, a portion of "live load" should be added when it includes piping, cable trays, etc, suspended from floors and an allowance should be made for future additional permanent loads.
- P = design accident pressure.
- T = thermal loads based on a temperature corresponding to the factored design accident pressure.
- E = seismic load based on design earthquake.
- E' = seismic load based on maximum hypothetical earthquake.
- W = wind load.
- W_t = stress induced by tornado wind velocity (drag, lift and torsion).
- P_i = stress due to differential pressure.
- ϕ = capacity reduction factor.
  - ϕ = 0.90 for concrete in flexure.
  - ϕ = 0.85 for tension, shear, bond and anchorage in concrete.
  - ϕ = 0.75 for spirally reinforced concrete compression members.
  - ϕ = 0.70 for tied compression members.
  - ϕ = 0.90 for fabricated structural steel.
  - ϕ = 0.90 for mild reinforcing steel (not prestressed) in direct tension excluding splices.
  - ϕ = 0.85 for mild reinforcing steel with mechanical splices (for lap splices, ϕ = 0.85 as above for bond and anchorage).
  - ϕ = 0.95 for prestressed tendons in direct tension.

The design earthquake ground acceleration at the site is 0.05g. The maximum hypothetical earthquake ground acceleration is 0.10g and 0.15g for Class 1 structures founded on bedrock and overburden respectively.

Seismic loads on structures and components are determined on the basis of dynamic analysis using the average velocity and acceleration spectrum curves shown as follows:

Plate Number	Earthquake	Location
II-4	0.05g	PSAR, Appendix 2B
2	0.10g	Supplement 1 to PSAR, Question 8.5
4	0.15g	Supplement 1 to PSAR, Question 8.5

Where realistic evaluation of dynamic properties is not possible, the maximum value of the acceleration response curve for the appropriate damping factor is used.

DAMPING FACTORS

Item	Percent of Critical Damping
Welded carbon and stainless steel assemblies (This includes reactor internals, supports and similar weldments.)	1
Steel frame structures (Both welded and high strength bolted)	2
Reinforced concrete equipment supports	2
Reinforced concrete frames and buildings	5
Prestressed concrete structures	
Under design earthquake forces	2
Under maximum hypothetical earthquake	5
Vital piping	0.5

Seismic forces are applied in the vertical and in any horizontal direction. The horizontal and vertical components of ground motion are applied simultaneously and the two components considered occurring in such a way that the stresses are directly additive.

The wind loads are determined from the fastest mile of wind for a 100-year occurrence as shown in Figure 1(b) of Reference 4. This is 95 mph at the site.

Simultaneous external loadings used in the tornado design are:

a. Differential pressure of 3 psi developed over 5 seconds.

b. External wind forces resulting from a tornado having a velocity of 300 mph.

c. Missile equivalent to an 8 inch diameter x 12 ft long piece of wood traveling end-on at 250 mph.

d. Missile equivalent to a 2000 pound automobile with a minimum impact area of 20 square feet traveling at a speed of 100 mph for 25 ft above grade.

The Reactor Building and engineered safeguards systems components are protected by barriers from all credible missiles which might be generated from the primary system. Local yielding or erosion of barriers is permissible due to jet or missile impact provided there is no general failure.

The final design of missile barrier and equipment support structures inside the Reactor Building is reviewed to assure that they can withstand applicable pressure loads, jet forces, pipe reactions and earthquake loads without loss of function. The deflections or deformations of structures and supports are checked to assure that the functions of the Reactor Building and engineered safeguards equipment are not impaired. Missile barriers are designed on the basis of absorbing energy by plastic yielding.

5A.3 DESIGN BASES FOR CLASS 2 STRUCTURES

5A.3.1 NORMAL OPERATION

For loads experienced in normal plant operation, Class 2 structures are designed in accordance with design methods of accepted standards and codes insofar as they are applicable.

5A.3.2 ACCIDENT AND SEISMIC CONDITIONS

For Class 2 structures, the working stress design method will be used and stress will be in accordance with ACI 318-63 and the AISC Codes.

5A.4 DESIGN BASES FOR CLASS 3 STRUCTURES

Class 3 structures are designed in accordance with design methods of accepted standards and codes insofar as they are applicable.

5A.5 WIND LOADING FOR CLASS 2 AND 3 STRUCTURES

The wind loads are determined from the fastest mile of wind for a 100-year occurrence as shown in Figure 1(b) of Reference 4. This is 95 mph at the site.

5A.6 LOADINGS COMMON TO ALL STRUCTURES

5A.6.1 ICE OR SNOW LOADING

A uniform distributed live load of 20 pounds per square foot is considered for roofs as stated in Section 1203.2 of the Southern Standard Building Code.

5A.6.2 TEMPERATURE

The station is designed for an ambient temperature range of 0 F to +100 F.

5A.7 MISSILE SHIELDING

Missile barriers inside the Reactor Building are designed to absorb the energy by plastic yielding.

5A.8 REFERENCES

1. Nuclear Reactors and Earthquakes, AEC Publication TID-7024.

2. Design of Nuclear Power Reactors Against Earthquakes, Housner, G. W., Proceedings of the Second World Conference on Earthquake Engineering, Volume 1, Japan 1960, Page 133.

3. Behavior of Structures During Earthquakes, Housner, G. W., Journal of the Engineering Mechanics Division, Proceedings of the American Society of Civil Engineers, October 1959, Page 109.

4. Wind Forces on Structures, Task Committee on Wind Forces, ASCE Paper No. 3269.

LIST OF EFFECTIVE PAGES
FSAR APPENDIX 5B

Quality Control

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

LIST OF TABLES

APPENDIX 5B

5B QUALITY CONTROLS

5B.1 FIELD WELDING

5B.1.1 SCOPE

This section outlines the general requirements for welding quality control to assure that all field welding is performed in full compliance with the applicable job specification. These requirements include the use of qualified welding inspectors and nondestructive testing technicians and the assurance that field welding is performed only by qualified welders using qualified procedures.

5B.1.2 QUALIFICATIONS FOR WELDING INSPECTORS

Duke Power welding inspectors are qualified by meeting the following minimum requirements:

(a) Inspectors will have a thorough knowledge of the various welding processes and the techniques employed in field construction and shall be able to demonstrate the proper methods for shielded metal-arc welding, gas tungsten-arc welding, gas metal-arc welding, and oxyacetylene welding.

(b) Inspectors will have a minimum of two years previous welding inspection experience or equivalent experience and training in welding fabrication.

(c) Inspectors will be required to demonstrate their knowledge of visual weld defects and of any test, such as vacuum boxing, that they may be required to perform.

5B.1.3 INSTRUCTIONS FOR FIELD WELDING INSPECTORS

The general instructions for field welding inspectors follow:

(a) Determine that the proper welding procedure specification is selected to match the base materials being welded and the welding processes being employed.

(b) Permit only welders properly qualified under the welding procedure specification to make welds under that procedure.

(c) Check to see that the welding electrodes, base filler rod, consumable insert rings, and backing rings all match that which has been specified.

(d) Inspect weld joints as necessary prior to welding to insure proper edge-preparation, cleaning, and fit-up.

(e) Check to see that the welding machine settings are correct and fall within the range of current and voltage specified.

(f) Check for proper preheat and interpass temperature.

(g) Inspect the in-process welding for proper technique, cleaning between passes, and appearance of individual weld beads.

5B.1.3.1 Postweld Heat Treatment

The field welding inspector inspects each postweld heat treatment operation to insure conformance with the applicable job specifications. Specific items to be checked include the following:

(a) A sufficient number of properly located thermocouples is selected to record temperatures accurately.

(b) Sufficient thermocouples are connected to temperature indicator-recorders to provide a permanent record of the heating rate, holding temperature and time, and the cooling rate.

(c) Temperature charts show proper heating rate, holding temperature, holding time, cooling rate, and the proper weld identification.

5B.1.3.2 Visual Inspection of Welds

The field welding inspectors are responsible for carrying out the necessary welding surveillance to insure that all welding meets the following requirements for visual quality and general workmanship. Visual inspection is performed before, during, and after welding.

All welds beads, passes, and completed welds do not contain more than acceptable limits of slag, cracks, porosity, incomplete penetration, and lack of fusion.

Cover passes are free of coarse ripples, irregular surface non-uniform bead pattern, high crown, deep ridges or valleys between beads, and blend smoothly and gradually into the surface of the base metal.

Butt welds are slightly convex, of uniform height, and have full penetration.

Fillet welds are of specified size, with full throat and, unless otherwise specified, the legs shall be of approximately equal length.

Repair, chipping, or grinding of welds is done in such a manner as not to gouge, groove, or reduce the base metal thicknesses.

Where different base metal thicknesses are joined by welding, the finished joint has a taper no steeper than 1:4 between the thick and thin sections.

5B.1.4 QUALIFICATIONS FOR NONDESTRUCTIVE TESTING TECHNICIANS

Duke Power NDT technicians are qualified by meeting the following minimum requirements:

(a) A technician will have a thorough knowledge of the type of testing he is to conduct. He will also be familiar with the welding procedure specification for the field welds he is inspecting.

(b) When required by the various codes, the technician is properly certified in accordance with the applicable section of the Society for Nondestructive Testing Recommended Practice No. SNT-TC-1A.

5B.1.5 INSTRUCTIONS FOR NONDESTRUCTIVE TESTING TECHNICIANS

5B.1.5.1 Radiographic Inspection

The NDT technicians are responsible for determining that all radiographic inspection, when required, is properly performed.

When the applicable job specifications require radiographic inspection of welds, the NDT technicians are responsible for determining that proper radiographic technique is followed and that the completed films are properly interpreted. The NDT technicians also review each completed radiograph.

Special attention shall be given to each of the following items for all radiographic inspection:

(a) Check the type of film intensifying screens, penetrameters, and source of radiation for conformance to the job specifications.

(b) Check the relative location of film, penetrameters, identifying numbers, and radiation source for each typical exposure.

(c) Review all completed film for quality and interpretation of defects. Check the exposed and developed film for proper density and visibility of penetrameters. If there is radiographic film of unacceptable quality or with questionable indications of defects, the weld is re-radiographed.

5B.1.5.2 Magnetic-Particle Inspection

The NDT technician is responsible for determining that magnetic-particle inspection, when required, is properly performed.

When the applicable job specifications require magnetic-particle inspection of welds, the NDT technician is responsible for determining that the proper technique is followed and that the results are properly interpreted.

Special attention is given to the following items for all magnetic-particle inspection:

(a) Determine that surfaces to be inspected are properly cleaned and are free of crevices which can produce false indications by trapping the iron powder.

(b) Determine that power source, current density, prod spacing, and application of iron powder all comply with the applicable specification requirements.

(c) Permit no arcing between the prods and weld surfaces.

(d) Interpret all linear or linearly disposed indications as defects.

(e) Probe questionable indications by thermal cutting, chipping, grinding, or filing to confirm the presence or absence of actual defects.

5B.1.5.3 Liquid Penetrant Inspection

The NDT technician is responsible for determining that all liquid penetrant inspection, when required, is properly performed.

When the applicable job specifications require liquid penetrant inspection for welds, the NDT technician is responsible for determining that the proper technique is followed and that the results are properly interpreted.

Special attention is given to the following items for all liquid penetrant inspection:

(a) Determine that surfaces to be inspected are properly cleaned and are free of crevices which can produce false indications by trapping the dye penetrant.

(b) Check to see that cleaner, dye penetrant, and developer are properly applied and the specified time intervals for dye penetration and developing are followed.

(c) Determine that indications are properly interpreted. Defects are identified as dye stains against the developer background. Lines or linearly disposed dots are indicative of cracks. Porosity and pinhole leaks appear as local patches or dots.

(d) Examine questionable indications by a 5x or stronger hand lens, and probe by grinding or filing to confirm the presence or absence of defects.

5B.1.5.4 Ultrasonic Inspection

The NDT technician is responsible for determining that all ultrasonic inspection, when required, is properly performed.

When the applicable job specifications require ultrasonic inspection for welds, the NDT technician is responsible for determining that the proper technique is followed and that the results are properly interpreted.

Special attention is given to the following items for all ultrasonic inspection:

(a) Determine that surfaces to be inspected have been properly prepared to give good coupling between the surface and the transducer.

(b) Determine that the inspection unit is functioning by use of a test block with known defects.

(c) Interpret all indications above a known reference level as defects.

5B.1.6 REPAIRS

It is the responsibility of the field welding inspector and the NDT technician to determine that all weld defects in excess of specified standards of acceptance are removed, repaired, and reinspected in accordance with the applicable job specifications.

5B.1.7 RECORDS

It is the responsibility of the welding inspector and the NDT technician to prepare records of inspections and testing to be kept on file at the job site.

5B.1.8 WELDING PROCEDURES

5B.1.8.1 Welding Procedure Specifications

All welding is in strict accordance with approved welding procedure specifications.

5B.1.8.2 Welder Qualification

All welders and welding operators who are to make welds under a code or standard which requires qualification of welders are tested and qualified accordingly before beginning production welding. Duke Power Company is responsible for testing and qualifying its own welders. The welding inspector is responsible in all cases for determining that the welders have passed the necessary qualification tests.

5B.2 PRESTRESSING

5B.2.1 GENERAL

These instructions and methods describe the quality control standards and measures applied in the control, manufacture, and field installation of the prestressing phase of construction of the reactor building.

5B.2.2 CONTROL

5B.2.2.1 Supervision

The subcontractor furnishes competent, experienced supervision of the tendon installation and tensioning operation until completion of post-tensioning. The above individual exercises a close check and rigid control of all post-tensioning operations, as necessary, for full compliance with specifications.

5B.2.2.2 Inspection of Duke's Work

The subcontractor is responsible for the inspection of Duke's handling and installation of tendon sheaths and bearing plates. To this end, he provides a competent technical representative to check the installation of these items by Duke. If any of Duke's work or actions jeopardize the subcontractor's work, he notifies Duke's Resident Engineer in writing. Failure to do this constitutes acceptance of Duke's work as it affects subcontractor's responsibilities.

5B.2.2.3 Arrangement of Prestressing Tendons

The configuration of the tendons in the dome is based on a three-way tendon system consisting of three groups of tendons oriented at 120 degrees with respect to each other. The vertical cylinder wall is provided with a system of vertical and horizontal (hoop) tendons. Hoop tendons are placed in a 120 degree system in which three tendons form a complete ring. Six buttresses are used as anchorages.

5B.2.3 DETAIL SHOP DRAWINGS

5B.2.3.1 Subcontractor

Upon award of the contract, Duke furnished engineering design drawings which were issued for construction of the prestressing work providing information required for the preparation of shop detail drawings by the subcontractor. The subcontractor furnished the following detail drawings and erection drawings to Duke:

(a) Outside dimensions of sheathing proposed for the tendon.

(b) Complete details of the post-tensioned wall and dome including dimensional locations of the tendons and necessary equipment and materials to place the tendons.

(c) Tendon characteristics indicating the A_s, f's, f_{sy}, and a typical stress-strain curve for the tendon used, as well as tendon force capability.

(d) Details of anchorages, bearing plates, and other accessories pertinent to the post-tensioning system.

(e) Erection drawings showing clearly the marking and positioning of tendons, anchorages and sheaths, and details showing alignment and setting tolerances required.

(f) Stressing sequence drawings.

5B.2.4 PRESTRESSING STEEL

5B.2.4.1 Materials and Fabrication

High strength steel wires are in accordance with ASTM A416 or A421 as a minimum requirement.

Wires are to be straightened if necessary to produce equal stress in all wires or wire groups that are to be stressed simultaneously or when necessary to insure proper positioning in sheaths. However, wires showing a permanent set are not to be straightened or installed if the bend exceeds 60 degrees and the radius is less
4. than 1.25 inches.

Tests were made on wire bent to 30, 60, and 90 degrees with a bend radius of 1.25
6. inches (5 times wire diameter) and wire bent to 30 and 60 degrees with a zero radius. The test specimens were from two different heats of 1/4 inch diameter

wire. All speciments within one test series were from the same heat and coil. In the sequence of cutting, every sixth specimen fell into the same group. The first group consisted of straight specimens for comparison.

Specimens were cut to a length of 15-1/2 inches, bent to the prescribed angle
6. and radius in a bend-tester, and straightened. The specimens were button
headed on each end and tensile tested to failure. The test results presented
in Table 5B-1 show that the strength of prestressing wire is not affected by
bending the wire 60 degrees around a 1.25 inch radius pin.

The button head is cold formed to a nominal diameter of 3/8 inch symmetrically
about the axis of the wires. If splitting is consistent and appears in all heads
3. or if there are more than two splits in which the opening exceeds 0.06 inch per
head, the wire is rejected. No forming process is used that causes indentation in
the wire. Wires showing indentations are rejected. Wires showing fabrication

TABLE 5B-1

BENT WIRE TEST RESULTS

	Sample	STRESS (psi)					
Group No.		1	2	3	4	5	6
Bend Angle (Degrees)		-	30	60	90	30	60
Bend Radius (inch)		-	1.25	1.25	1.25	0	0
SERIES I Heat #A67386	1	251,500	257,650	257,650	259,650	251,550	230,150
	2	254,600	259,650	257,650	257,650	251,550	237,250
	3	256,600	257,650	259,650	256,600	252,550	240,300
	4	258,650	258,650	258,650	256,600	247,450	235,250
	5	259,650	261,700	259,650	258,650	248,450	237,250
	6	258,650	259,650	260,700	258,650		
	7	260,700	254,600	261,700	258,650		
	8	259,650	258,650	260,700	258,650		
	9	260,700	258,650	260,700	257,650		
	10	260,700	258,650	255,600	260,700		
Average		258,850	258,550	259,250	258,350	250,300	236,050
SERIES II Heat #A72005	11	252,550	249,500	249,500		243,400	229,100
	12	252,550	249,500	251,550		243,400	227,100
	13	249,500	249,500	248,450		243,400	229,100
	14	248,450	249,500	250,500		242,350	227,100
	15	247,450	250,500	248,450		241,350	228,100
	16	250,500	249,500	248,450			
	17	254,600	253,550	252,550			
	18	251,550	251,550	251,550			
	19	252,550	251,550	249,500			
	20	249,500	254,600	249,500			
Average		250,900	250,900	250,000		242,750	228,100

defects, wires having welds or joints made during manufacture, or broken wires are removed and replaced.

The BBR Bureau Standard for button head splits is a maximum number of two splits with a width of 0.06 inch. The Prescon Corporationhas run tests on button
6. heads with splits and based on an evaluation of the test results, the BBR Bureau Standard is acceptable.

5B.2.4.2 Protection

Prestressing steel is protected from mechanical damage and corrosion during shipment, storage, installation, and tensioning. A thin film of No-Ox-Id (R) 500,
21. as manufactured by Dearborn Chemical Company or Visconorust 1601, manufactured by Viscosity Oil Company, is applied to the prestressing steel after fabrication in accordance with the manufacturer's instructions. The steel is then wrapped before shipment to the site. The steel is not handled, shipped, or stored in a manner that will cause a permanent set or notch, change it material properties, or expose it to inclement weather or injurious agents such as chloride containing solutions. Damaged or corroded tendons are rejected.

5B.2.4.3 Installation

Prestressing steel may be installed in the sheaths after the concrete has taken its initial set.

5B.2.5 ANCHORAGES AND BEARING PLATES

5B.2.5.1 Anchorages

Anchorages will develop the minimum guaranteed ultimate strength of the tendon and the minimum elongation of the tendon material as required by the applicable ASTM specification.

5B.2.5.2 Bearing Plates

Bearing plates are capable of developing the ultimate strength of the tendon and distributing the bearing load over the bearing surface of the concrete. Bearing plates conform to the following requirements:

(a) The transfer unit compressive stress on the concrete directly underneath the plate or assembly is in conformance with the ACI Code 318-63, latest edition.

(b) Bending stresses in the plates induced by the pull of the prestressing steel shall not exceed 22,000 psi for structural steel and 15,000 psi for cast steel, except as experimental data may indicate that higher stresses are satisfactory.

(c) Materials shall meet requirements of ASTM A36 for structural shapes or ASTM A148, Grade 80-40 for cast steel, or higher quality materials approved by Duke to meet strain requirements.

(d) Design, fabrication, and erection shall meet the requirements of the latest AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings."

5B.2.6 SHEATHS

5B.2.6.1 Materials

Sheaths for post-tensioning tendons are ungalvanized corrugated articulated tubing and meet the following requirements:

(a) The internal diameter is adequate to allow insertion of prestressing steel after concrete placement.

(b) The sheaths will withstand the placing of concrete at a pour rate of two feet per hour (with mechanical vibration) without ovalling or changing alignment.

(c) Sheaths are protected from corrosion during storage.

5B.2.6.2 Sheath Fabrication

The sheaths are cut to length and bent to shape. The bending is accomplished without wrinkling the metal. Dented or wrinkled sheaths are replaced. Finished bent or straight dimensions are in accordance with approved drawings.

5B.2.6.3 Installation (by Duke)

Sheaths are accurately installed in the forms at the location shown on the drawings to a tolerance of ± one-half (1/2) inch, except as otherwise indicated on the drawings. The sheaths are supported in such a manner as to prevent displacement during concrete placement. The sheath is supported at the ends and at such intervals as are in accordance with the drawings. Damaged or improperly bent sheaths are rejected.

5B.2.6.4 Cleaning and Venting

Just prior to insertion of the tendon, the sheath is cleaned by the use of compressed air or other suitable means.

5B.2.7 CORROSION PROTECTIVE GREASE

Corrosion protection is provided by grease injected into the sheaths under pressure. Grease will be Visconorust 2090P manufactured by Viscosity Oil Company.

The grease is sampled and laboratory tested for chemical analysis to establish conformance with specifications and for deleterious substances such as water soluble chlorides, nitrates, and sulfides.

Visconorust 2090P Casing Filler is a petroleum base corrosion preventive designed for bulk application and extended protection.

4.

It has:

A. 1. A three phase protective system starting with a polar agent preferentially wetting the wires and displacing any moisture, rust preventive additives molecularly attached to the wetting agent and a petroleum barrier completing the resistant coating.

2. The property to emulsify any moisture picked up in the system while being pumped through the casing and either carrying it out the other end or nullifying its rusting ability if the moisture is trapped in the casing.

3. Reserve Alkalinity - Base Number of 3. The basic formulation of Visconorust 2090P is very stable and resistant to exterior moisture encroachment as well as mild acids and alkali. However, because of the probability of picking up moisture as the rust preventive is pumped through the tendons, additional safety factor, besides the barrier action, is available to neutralize any acids that might form between the interface of the moisture and rust preventive.

   As a comparison for a more definitive value of Base Number 3 (equivalent neutralization value of 3 mg of Potassium Hydroxide per gram of product) Crankcase motor oils which undergo constant formation of acids require only a Base Number of 6. Hence the Base Number of 3 will provide enough additional protection, since the tendons are subject only to a static environmental condition.

   Tests have been run using volatile acids such as Hydro Bromic Acid, in an attempt to penetrate the Visconorust 2090P film and cause corrosion, without success thus far.

4. Only a trace amount of water soluble chlorides, sulfides or nitrates.

5. A plugging agent designed to supplement the natural tendency of the microwax crystals and amorphus solid components to form a filter cake bridging any hair line cracks in the concrete, with which the casing filler might come in contact.

6. Self healing qualities at the ambient temperature expected during operation, to take care of any voids created by wire movement.

7. Thixotropic properties that provide pumpability below 50°F.

8. Radiation Resistance:

   Visconorust 2090P has been subjected to 1×10^6 rads by the Gamma Process Company of New York. Results show that the Gamma rays did not have any material effect on either the physical or chemical structure (as noted by a negligible change in base number).

   Corroboration of the test results is readily noted in extensive literature on this subject, a few of which are listed below:

   Bibliography:

   a. The Lubrication of Nuclear Power Plants by R. S. Barnett - NLGI - October 1960.

b. How Radiation Affects Petroleum Lubricants - Power, Vol. 100 December 1956, Page 164.

c. Conventional Lubricants Are Sufficiently Radiation Resistant for Most Nuclear Power Reactor Applications by E. D. Reeves SAE Journal Vol. 66, May, 1956, Page 56-57.

d. Organic Lubricants and Polymers for Nuclear Power Plants by Bolt and Carroll.

B. The amount of nitrate found in the 90,000 gallons of Nuclear Grade material made for Palisades, Point Beach and Turkey Point plants, so far, was "0" and practically, in order to keep the trace amounts allowed, be it 2 or 4, the amounts must be kept at zero. However, the refinery requires the use of 4 parts per million figure as a maximum.

Infra-red spectographic analysis shows Visconorust 2090P and NO-OX-ID CM to be quite similar with approximately the same amounts of wetting agents and Rust Preventives in the petroleum carriers.

PERFORMANCE DATA

6.

Item	NO-OX-ID	Visconorust 2090P	ASTM Method
Weight Per Gal.	7.2 - 7.5 lbs.	7.3 - 7.6 lbs.	---
Pour Point	110° - 120°F		D-97
Flash Point (coc)	400°F	385°F	D-92
Viscosity @ 150°F	125 - 150 SSU	116 SSU	D-88
Viscosity @ 210°F	55-75 SSU	59 SSU	D-88
Spec. Gray @ 60°F	0.88 - 0.90	0.88 - 0.91	D-287
Pene. (cone) @ 77°F	325 - 370	370	D-937
Water Sol Chlorides	1 PPM	1 PPM	D-512
Water Sol Nitrates	2 PPM	4 PPM	D-1255
Water Sol Sulfides	1 PPM	1 PPM	D-992
Phenoloc Bodies (As Phenol)	1 PPM	1 PPM	---
Shrinkage Factor (150°F to 70°F)	3.5 - 4.5%	3.5 - 4.5%	---

5B.2.8 PRESTRESSING

5B.2.8.1 Tensioning Schedule

Prestressing begins after the concrete in the walls and the dome has reached the specified f'c. The dome and hoop tendons are tensioned from both ends, and the vertical tendons are tensioned from either the top end or from both ends. Six jacks are used throughout the post-tensioning operations.

Phase 1
Twelve hoop tendons above elevation 943 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 2
Thirty-six dome tendons in the periphery of the dome.

Phase 3
Twelve hoop tendons above elevation 943 feet + 6 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 4
Remaining 126 dome tendons.

Phase 5
One hundred and forty-one hoop tendons from elevation 865 feet + 0 inches to elevation 943 feet + 6 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 6
6. Close the construction opening if not closed prior to Phase 6.

Phase 7
One hundred and fifty-three hoop tendons from elevation 775 feet + 0 inches to elevation 865 feet + 0 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 8
Forty-two hoop tendons from elevation 776 feet + 0 inches to elevation 801 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 9
One hundred and seventy-six vertical tendons.

Phase 10
Two hundred and fifty-two hoop tendons from elevation 801 feet + 6 inches to elevation 943 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 11
Ten hoop tendons above elevation 949 feet + 10-2/3 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 12
Ten hoop tendons above elevation 949 feet + 10-2/3 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

5B.2.8.2 Force and Stress Measurements

Force and stress measurements are made by measurement of elongation of the prestressing steel after taking up initial slack and comparing it with the force in-
21. dicated by the jack-dynamometer or pressure gauge. Force jack pressure gauge

or dynamometer combinations are calibrated against known precise standards before application of prestressing force. All guages are calibrated on a dead weight calibration appartus. The presence of two gauges, one gauge on the pump and one gauge on the jack, provides a means to maintain a constant check of the calibra-
7. tion of the gauges. Based on the actual calibration tests of the stressing equipment, it was concluded that the pump efficiency does not influence the equipment accuracy and that the stressing accuracy depends only on the ram efficiency. Therefore, any combination of ram, gauge and pump may be used interchangeably. During stressing, records are made of elongations as well as pressures obtained. Jack dynamometer or gauge combinations are checked against elongation of the tendon and any discrepancy exceeding plus or minus 5 percent will be evaluated by Design Engineering. The measured elongation will differ from the calculated elongation because of the following:

1) The statistical modulus of elasticity of 29.3 million psi for straight, untwisted wire.

21. 2) The actual length and location of the tendon sheath will vary from the theoretical position due to approved placing tolerances.

3) All wires in a tendon are equal in length and the tendon is twisted to compensate for the difference in actual arc lengths. The twisting forms a wire cable configuration which does not follow the sheath centerline and which has a modified modulus of elasticity value.

4) The friction factor used in calculations is an average value based on experience. The true influence of friction on each tendon can be significantly different from the average value used in calculations.

5) The permissible tolerance in pressure gauge accuracy combined with the possible variables in stressing techniques such as reading the gauges and scales can constitute a significant difference.

Calibration of the pressure gauges are maintained accurate within the following limits:

0 to 2500 psi - Accuracy limit of the gauge, plus or minus 50 psi.
2500 to 7030 psi- Plus or minus 2 percent of gauge reading.

Pressure gauges are recalibrated after each stressing cycle on unit 3 and as requested by Duke Power, during and at the end of the tensioning operations on units 1 and 2.

5B.2.8.3 Strain Gauge Installation and Protection

Strain or force gauging devices are installed on certain tendon areas prior to and/or during installation. These strain devices are monitored during the tensioning operation and used during subsequent pressure testing. Approximately 4 tendon sets are instrumented with load cells.

5B.2.8.4 Tests, Samples, Inspections

Sampling and testing conforms to ASTM Standard A421 and as specified herein.

Each size of wire from each mill heat shipped to the site is assigned an individual lot number and tagged in such a manner that each such lot can be accurately identified at the job site. Anchorage assemblies are likewise identified. All unidentified prestressing steel or anchorage assemblies received at the job site are rejected.

Random samples as specified in the ASTM Standard stated above are takne from each lot of prestressing steel used in the work. With each sample of prestressing steel wire that is tested, there is submitted a certificate stating the manufacturer's minimum guaranteed ultimate tensile strength of the sample tested.

For the prefabricated tendons, one completely fabricated prestressing test specimen tendon 5 feet in length, including anchorage assemblies, is tested for each size of tendon contained in an individual shipping release.

No prefabricated tendon is shipped to the site without first having been released by Duke, and each tendon is tagged before shipment for identification purposes. The release of any material by Duke does not preclude subsequent rejection if the material is damaged in transit or later damaged or found to be defective.

Duke shop inspects the prefabricated tendons prior to being shipped to the job site.

The anchorages and tendons are inspected at the job site for corrosion and mechanical damage during shipment, storage, installation and tensioning. Damaged or corroded tendons and anchorages are rejected.

5B.2.8.5 Acceptance

Final acceptance for warranty purposes is the successful completion of the pressure testing of the reactor building.

5B.3 CONCRETE

5B.3.1 MIX DESIGN

5B.3.1.1 General

Concrete mixes are designed in accordance with "Recommended Practice for Selecting Proportions for Concrete" (ACI 613), using materials qualified and accepted for the work; and the strength, workability, and other characteristics of the mixes are ascertained before placement. Duke Power's concrete control laboratory is set up on the Oconee site. A batch-plant inspector is provided, and testing as shown below is performed. Field control is in accordance with the "Manual of Concrete Inspection" as reported by ACI Committee 611.

5B.3.1.2 Mix Design

Only those mixes meeting the design requirements specified for reactor building concrete are used. Trial mixes are tested in accordance with the applicable ASTM Codes as follows:

Test	ASTM
Air Content	C231
Slump	C143
Bleeding	C232
Making and Curing Cylinders in Laboratory	C192
Compressive Strength Tests	C39

Six cylinders are cast from each design mix for two tests on each of the following days: 7, 28, and 90.

Test cylinders are cast from the mix proportions selected for construction and the following concrete properties determined:

Uniaxial creep
Modulus of elasticity and Poisson's Ratio
Autogenous shrinkage
Thermal diffusivity
Thermal coefficient of expansion
Compressive strength

5B.3.2 TESTS

5B.3.2.1 Aggregates

Aggregate testing is performed as follows:

(a) Sand sample for gradation (ASTM C33 Fine Aggregate)
(b) Organic test on sand (ASTM C40)
(c) 3/4" sample for gradation (ASTM C33, Size No. 67)
(d) 1-1/2 inch sample for gradation (ASTM C33, Size No. 4)
(e) Check for proportion of flat and elongated particles.

Acceptability of aggregates is based on the following ASTM tests. These are performed by a qualified testing laboratory.

Test	ASTM
L. A. Rattler	C131
Clay Lumps Natural Aggregate	C142
Material Finer No. 200 Sieve	C117
Mortar making properties	C87
Organic impurities	C40
Potential Reactivity (chemical)	C289
Potential Reactivity (mortar bar)	C227
Sieve Analysis	C136
Soundness	C88
Specific Gravity and Absorption	C127
Specific Gravity and Absorption	C128

5B.3.2.2 Cement

Cement conforms to ASTM C150 and tested to ASTM C114.

The manufacturer submits certified copies of mill test reports showing the chemical composition and certifying that the cement complies with the specification on each shipment delivered to the site. In addition to the manufacturer's tests, cement is sampled periodically at the site and tested to ascertain conformance with ASTM Specification C150.

5B.3.2.3 Water

Water is potable and does not contain impurities in amounts that will cause a change of more than 25 percent in setting time for the Portland Cement, nor a reduction in the compressive strength of mortar of more than 5 percent as compared with results obtained using distilled water.

5B.3.2.4 Admixtures

Admixtures, as to be determined by detailed mix design, conform to applicable ASTM Specification covering such materials and their testing.

5B.3.2.5 Concrete Test Cylinders

Concrete cylinders for compression testing are made and stripped within 24 hours after casting, and marked and stored in the curing room. These cylinders are made in accordance with ASTM C21, "Tentative Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field."

Slump, air content, and temperature are taken when cylinders are cast and for each 35 yards of concrete placed. Slump tests are performed in accordance with ASTM C143, "Standard Method of Test for Slump of Portland Cement Concrete." Air tests are performed in accordance with ASTM C231, "Standard Method of Test of Air Content of Freshly Mixed Concrete by the Pressure Method." Compressive strength tests are made in accordance with ASTM C39, "Method of Test for Compressive Strength of Molded Concrete Cylinders."

Six standard test cylinders are obtained and molded for concrete placed in excess of 10 cubic yards in any one day, with 6 additional cylinders for each successive 100 cubic yards placed. Two cylinders are tested at the age of 7, 28, and 90 days.

5B.3.2.6 Construction Practice

The standards or specifications on quality control and tests of concrete during construction are equal to or better than requirements of ACI 301. Some of the areas where quality control exceeds the requirements of ACI 301 are as follows:

(a) Requirements for water quality.
(b) Placing temperature of concrete.
(c) Requirements for aggregate acceptability.
(d) Requirements for test cylinders.

Horizontal construction joints are prepared for receiving the next lift by blasting with compressed air. Surface set retardant compounds are not used.

Horizontal surfaces are wetted and covered with a coating of mortar of the same cement-sand ratio as used in the concrete immediately before the concrete is placed.

Vertical joints are also blasted with compressed air, cleaned, and wetted before placing concrete.

Vertical joints are placed at the center of each buttress to take advantage of the 50 percent additional horizontal prestress due to the overlapping of the anchored hoop tendons.

Horizontal joints between buttresses are at the same elevation. These joints are prepared as stated above to provide maximum possible bond. Principal tension in the membrane is limited to $3\sqrt{f'_c}$.

5B.4 REINFORCING STEEL

5B.4.1 GENERAL

All reinforcing steel conforms to the purchase order specification, and inspection and testing is performed at the mill to ASTM requirements. Mill test reports are submitted for engineering review and approval. Metallurgical inspection and testing of the reinforcing steel is done in accordance with the ACI Code 318-63, Chapter 8.

Reinforcing steel is inspected at delivery as well as at erection. The condition of the material must meet all of the requirements of ACI 318-63, as well as any additional requirements made by the inspector.

5B.4.2 SPLICES

Number 14S and 18S reinforcing steel for which the ACI Code requires welded or mechanical splices is spliced by the CADWELD process using full tensile strength "T" series connections. Quality control is maintained by qualification testing of the individual splicing crews, visual inspection of each completed connection, and random sampling and tensile testing of splices.

Prior to making any production splices, each individual splicing crew prepares sample splices for tensile testing covering each bar size and position used in production to qualify. The sample splices must be properly filled, free of porous metal and meet the minimum requirement for tensile strength as stated below.

All splices are subjected to visual inspection and must meet the following standards:

(a) Sound, nonporous filler metal must be visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed 1/4 inch from the end of the sleeve due to the packing material, and is not considered a poor fill.

(b) Splices which contain slag or porous metal in the riser, tap hole, or at the ends of the sleeves (general porosity) are rejected. A single shrinkage bubble present below the riser is not detrimental and should be distinguished from general porosity as described above.

In addition to the above, random splices are subjected to mechanical tests and must meet the following standards:

(a) The strength of 95 percent of the CADWELD splices tested will be greater than 125 percent of the specified minimum yield strength for the particular bar size and ASTM specification.

(b) The strength of the average of all the splices tested will be equal to or greater than the minimum ultimate strength for the particular bar size and ASTM specification.

(c) No failures of CADWELD splices below the required minimum yield strength are expected. In the unlikely event that one should occur, it would be sent to a testing laboratory for analysis of failure. Based on the testing laboratory's report, additional samples would be taken to insure that there are no other defective welds.

Tests are made in accordance with the following schedule for each position, bar size and grade of bar:

1 out of first 10 splices
3 out of next 100 splices
2 out of next 100 and each subsequent 100 splices

Test splices are made by having test bars of 3 feet length spliced in sequence with the production bars. In addition, two production splices are cut out and tested for each 100 test splices.

The inspections and tests are performed by individuals thoroughly trained by the CADWELD manufacturer.

For reinforcing steel of size 11 and under, lap splices are permitted in accordance with ACI 318-63, Chapter 8.

LIST OF EFFECTIVE PAGES
FSAR SECTION 6

Engineered Safeguards

Page	Revision
LOEP-- 1 of 2	Rev. 26
LOEP-- 2 of 2	Rev. 26
6-i	Rev. 21
6-ii	Original
6-iii	Rev. 21
6-iv	Rev. 19
6-v	Rev. 5
6-1	Original
6-2	Original
6-3	Original
6-4	Rev. 8
6-5	Rev. 19
6-6	Original
6-7	Original
6-8	Rev. 9
6-9	Original
6-10	Rev. 8
6-11	Original
6-12	Rev. 5
6-12a	Rev. 5
6-12b	Rev. 4
6-12c	Rev. 4
6-12d	Rev. 4
6-12e	Rev. 9
6-12e(a)	Rev. 9
6-12f	Rev. 9
6-13	Rev. 17
6-14	Rev. 21
6-15	Original
6-16	Original
6-17	Original
6-18	Original
6-19	Original
6-20	Rev. 5
6-20a	Rev. 5
6-21	Original
6-22	Original
6-23	Original
6-24	Rev. 16
6-25	Rev. 16
6-26	Rev. 16
6-27	Rev. 18
6-27a	Rev. 20
6-28	Rev. 8
6-29	Rev. 8
6-29a	Rev. 16
6-29b	Rev. 20

Engineered Safeguards

Page	Revision
6-29c	Rev. 10
6-30	Rev. 19
6-30a	Rev. 19
6-30b	Rev. 19
6-31	Rev. 16
6-32	Original
Fig. 6-1	Original
Fig. 6-2	Rev. 26
Fig. 6-3	Rev. 16
Fig. 6-4	Original
Fig. 6-5	Rev. 18
Fig. 6-5A	Rev. 5
Fig. 6-5B	Rev. 5
Fig. 6-5C	Rev. 16
Fig. 6-5D	Rev. 8
Fig. 6-5E	Rev. 5
Fig. 6-5F	Rev. 5
Fig. 6-6	Original
Fig. 6-7	Original
Fig. 6-8	Rev. 19
Fig. 6-9	Original
Fig. 6-10	Rev. 16
Fig. 6-11	Original
Fig. 6-11A	Rev. 5
Fig. 6-12	Original

CONTENTS (Cont'd)

CONTENTS (Cont'd)

Rev. 5. 5/25/70
Rev. 8. 7/23/70
Rev. 19. 5/5/72
Rev. 21. 7/26/72

LIST OF TABLES

LIST OF FIGURES
(At Rear of Section)

6 ENGINEERED SAFEGUARDS

Engineered safeguards are those systems and components designed to function under accident conditions to prevent or minimize the severity of an accident or to mitigate the consequences of an accident. During accident conditions when reactor coolant is lost, the engineered safeguards act to provide emergency cooling to assure structural integrity of the core, to maintain the integrity of the reactor building, and to collect and filter potential reactor building penetration leakage. Separate and independent engineered safeguards are provided for each of the three reactor units at Oconee. Special precautions are taken to assure high quality in the system design and components.

The engineered safeguards include provisions for:

a. High pressure injection.

b. Low pressure injection.

c. Core flooding.

d. Two types of reactor building cooling.

e. The collection and control of reactor building penetration leakage.

f. Reactor building isolation.

Figure 6-1 schematically depicts the portion of the engineered safeguards system related to core and building protection (see (a) through (d) above). A general description of the engineered safeguards provisions is presented below and a more detailed description is presented in the latter portion of this section. Since each reactor unit has the same arrangement of emergency safeguard systems, the performance of the systems is described on a unit basis.

The high and low pressure injection strings and the core flooding tanks are designed to form collectively an overall emergency core cooling system (ECCS), which is designed to prevent melting or physical disarrangement of the core over the entire spectrum of reactor coolant system break sizes. Figure 6-2 shows the emergency core cooling systems for one reactor unit. The high pressure injection system is arranged so that three pumps are available for emergency use. The low pressure injection system is arranged to assure that two pumps are normally available and a third pump is installed but normally valved off. The core flooding system for each unit is composed of two separate pressurized tanks containing borated water at reactor building ambient temperature. These tanks automatically discharge their contents into the reactor vessel at a preset reactor coolant system pressure without reliance on any actuating signal, any motive power or any external actuated component.

Reactor building integrity is assured by two full-capacity, independent, pressure reducing systems operating on different principles; the reactor building spray system and the reactor building emergency cooling system. (Refer to Figures 6-3 and 6-4.) These systems have the redundancy required to meet the single failure criterion. These systems operate to prevent reactor building pressure from exceeding design limits over the spectrum of reactor coolant system break sizes and to reduce the driving force for leakage of radioactive materials from the reactor building.

The reactor building penetration room ventilation system shown on Figure 6-5 collects and filters air leakage from reactor building penetrations following an accident. Two full-capacity filtering paths are provided. This system provides a secondary containment to permit the collection and filtering of radioactive material before release to the atmosphere.

Operability of engineered safeguards equipment is assured in several ways. Much of the equipment in these systems serves a function during normal reactor operation. In those cases where equipment is used for emergency functions only, such as the reactor building spray system, systems have been designed to permit meaningful periodic tests. Operational reliability is achieved by using proven component designs, and by conducting tests where either the component or its application was considered unique. Quality control procedures are imposed on the components of the engineered safeguards systems. These procedures include use of accepted codes and standards as well as supplementary test and inspection requirements to assure that all components will perform their intended function under the design conditions following a loss of coolant accident.

The purpose of this section is to describe the physical arrangement, design, and operation of the engineered safeguards systems as related to their safety function.

Reactor building isolation is described in Section 5. Other sections of the report contain information which is pertinent to the engineered safeguards systems. Section 7 describes the actuation instrumentation of these systems. Section 14 describes the analysis of the engineered safeguards systems' capability to provide adequate protection during accident conditions. Section 9 discusses functions performed by these systems during normal operation and gives further design details and descriptive information concerning those systems.

6.1 EMERGENCY CORE COOLING SYSTEMS (ECCS)

6.1.1 DESIGN BASES

The principal design basis for the emergency core cooling system as described in the proposed AEC General Design Criterion 44 has been met. Protection for the entire spectrum of break sizes is provided. Two separate and independent flow paths containing redundant active components are provided in the ECCS. Redundancy in active components assures performing the required functions should a single failure occur in any of the active components. Separate power sources are provided to the redundant active component. Separate instrument channels are used to actuate the systems. The adequacy of the installed ECCS to prevent fuel and clad damage is discussed in Section 14.

The ECCS is designed to operate in the following modes:

a. Injection of borated water from the borated water storage tank by the high pressure injection system.

b. Rapid injection of borated water by the core flooding system.

c. Injection of borated water from the borated water storage tank by the low pressure injection system.

d. Long term core cooling by recirculation of injection water from the reactor building sump to the core by the low pressure injection pumps.

Although the high and low pressure emergency injection strings operate to provide full protection across the entire spectrum of break sizes, each system may operate individually and each is initiated independently. High pressure injection prevents uncovering of the core for small coolant piping leaks where high system pressure is maintained, and to delay uncovering of the core for intermediate-sized leaks. The core flooding and low pressure injection provisions are designed to recover the core at intermediate-to-low pressures, and to assure adequate core cooling for break sizes ranging from intermediate breaks to the double-ended rupture of the largest pipe. The low pressure injection system is also designed to permit long-term core cooling in the recirculation mode after a loss-of-coolant accident. The injection and core flooding functions are subdivided so that there are two separate and independent strings, each including one high pressure pump, one low pressure pump, and one core flooding tank. The redundant protection afforded by the ECCS components, subsystems, and systems for the spectrum of reactor coolant pipe break sizes is illustrated in Figure 14-55.

6.1.2 SYSTEM DESIGN

6.1.2.1 ECCS Operation

The schematic diagram for the emergency core cooling systems is shown in Figure 6-2.

6.1.2.1.1 High Pressure Injection

During normal reactor operation, the high pressure injection system recirculates reactor coolant for purification and for supply of seal water to the reactor coolant circulating pumps. This normal operation mode and component data are described in Section 9.

The high pressure injection system is initiated at: (a) a low reactor coolant system pressure of 1,500 psig or (b) a reactor building pressure of 4 psig. Automatic actuation of the valves and pumps by the actuation signals switches the system from its normal operating mode to the emergency operating mode to deliver water from the borated water storage tank into the reactor vessel through the reactor coolant inlet lines. The following automatic actions accomplish this change:

a. The isolation valves in the purification letdown line and in the seal return lines close.

b. The high pressure injection pumps start.

c. The inlet valve in each high pressure injection line opens.

d. The valves in the lines connecting to the borated water storage tank outlet header open.

In addition to the automatic action described, the pumps and valves may be manually operated from the control room.

Operation of the high pressure injection system in the emergency mode will continue until the system action is manually terminated.

6.1.2.1.2 Low Pressure Injection (LPI)

The low pressure injection system is designed to maintain core cooling for larger break sizes. The low pressure system operates independently of and in addition to the high pressure system. A description of the normal reactor operation mode and component data for the system is given in Section 9.

Automatic actuation of the low pressure injection system is initiated at: (a)
8. 500 psig or (b) a reactor building pressure of 4 psig. Initiation of opera-
tion provides the following actions:

a. The valves in the lines connecting to the borated water storage tank outlet header open.

b. The low pressure injection pumps start on receipt of an engineered safeguards signal.

c. The inlet valves in the low pressure injection lines open.

d. Low pressure service water pumps start.

e. Service water valves from low pressure injection coolers open.

Low pressure injection is accomplished through two separate flow paths, each including one pump and one heat exchanger and terminating directly in the reactor vessel through core flooding nozzles located on opposite sides of the vessel.

The initial operation of the low pressure injection system involves pumping water from the borated water storage tank into the reactor vessel. With all pumps operating and assuming the maximum break size, this mode of operation lasts for a minimum of about 30 minutes. When the borated water storage tank is approximately 94% empty, a low water level alarm is annunciated in the control room. At this time the operator will take action to open the suction valve from the reactor building emergency sump, permitting recirculation of the spilled reactor coolant and injection water from the reactor building emergency sump.

6.1.2.1.3 Core Flooding System

The core flooding system provides core protection continuity for intermediate and large reactor coolant system pipe failures. It automatically floods the core when the reactor coolant system pressure drops below 600 psig. The core flooding system is self-contained, self-actuating, and passive in nature. The combined coolant volume in the two tanks is sufficient to recover the core hot spot assuming no liquid remains in the reactor vessel following the loss-of-coolant accident.

The discharge pipe from each core flooding tank (CFT) is attached directly to a reactor vessel core flooding nozzle. Each core flooding line at the outlet of the CFT's contains an electric motor operated stop valve adjacent to the tank and two in-line check valves in series. The stop valves at the core flooding tank outlet are fully open during reactor power operation. Valve position indication is shown in the control room. During power operation when the reactor coolant system pressure is higher than the core flooding system pressure, the two series check valves between the flooding nozzles and the CFT's prevent high pressure reactor coolant from entering the core flooding tanks.

The driving force to inject the stored borated water into the reactor vessel is supplied by pressurized nitrogen which occupies approximately one-third of the core flooding tank volume. Connections are provided for adding both borated water and nitrogen during power operation so that the proper level and pressure may be maintained. Each core flooding tank is protected from overpressurization by a relief valve installed directly on the tank. The size of these re-
16. lief valves is based upon maximum water makeup rate to the tank. Redundant level and pressure indicators and alarms are provided in the control room for each tank.

Design data for major system components are shown in Table 6-1.

Table 6-1
Core Flooding System Components Data

	Core Flooding Tanks	
	Number	2
	Design Pressure, psig	700
	Operating Pressure, psig	600
19.	Minimum Pressure, psig	575
	Design Temperature, F	300
	Operating Temperature, F	110
	Total Volume, ft^3	1,410
19.	Normal Water Volume, ft^3	1,040
	Minimum Water Volume, ft^3	1,010
	Material of Construction	Carbon steel lined with SS
	Check Valves	
	Number per Flood Line	2
	Size, in.	14
2.	Material	316 SS
	Design Pressure, psig	2,500
	Design Temperature	
	Valve nearest reactor, F	650
	Valve nearest tank, F	300

Rev. 2. 2/9/70
Rev. 16. 7/30/71
Rev. 19. 5/5/72

Table 6-1 (Cont'd)
Core Flooding System Components Data

Isolation Valves

Number per Flood Line	1
Size, in.	14
Material	304 SS
Design Pressure, psig	2,500
Design Temperature, F	300

Piping	Reactor to First Check Valve	First Check Valve to Isolation Valve	Isolation Valve to Tank
Size, in.	14	14	14
Material	316 SS	304 SS	304 SS
Design Pressure, psig	2,500	2,500	700
Design Temperature, F	650	300	300

6.1.2.2 Codes and Standards

The high pressure injection,low pressure injection, and core flooding systems are designed and manufactured to the Codes and Standards in Table 6-5 and in Section 9.

6.1.2.3 Material Compatibility

All components with surfaces in contact with water containing boric acid are protected from corrosion and deterioration. The high pressure injection system, which operates continuously with borated reactor coolant, is constructed entirely of stainless steel. With the exception of the borated water storage tank, the major components in low pressure injection are constructed of stainless steel. The borated water storage tank is carbon steel with an interior phenolic coating. The core flooding piping and valves are stainless steel and the tanks are constructed of stainless clad carbon steel.

6.1.2.4 Component Design

Piping

The high pressure injection and low pressure injection lines are designed for the normal operating conditions. The system temperature and pressure requirements are greater than those encountered during emergency operation. The low pressure injection system piping and valves are subjected to more severe conditions during decay heat removal operation than during emergency operation and, therefore, operate well within the design conditions. Table 6-6 gives the design pressure and temperatures of these systems. To assure system integrity, major piping has welded connections except where flanges are dictated for maintenance reasons.

Pumps

The pumps used in the emergency core cooling systems are of proven design and have been used in many other applications. Pumps similar to the high pressure injection pumps have been used in boiler feed pump service and in high pressure makeup pump nuclear reactor service. Pumps similar to the low pressure injection pumps are used extensively in refinery service. The low pressure injection pump seals have been tested satisfactorily under the conditions which would be encountered during the loss of coolant accident.(1) Both the high pressure and low pressure injection pump casings are liquid penetrant tested by methods described in the ASME Boiler and Pressure Vessel Code, Section VIII, and have been hydrotested and qualified to be able to withstand pressures as great or greater than 1.5 times the system design pressure. The pumps are designed so that periodic testing may be performed to assure operability and ready availability. The operating characteristics of each engineered safeguard pump are verified by shop testing before installation of the pumps.

Heat Exchangers

The low pressure injection heat exchangers are designed and manufactured to the requirements of the ASME VIII and the TEMA-R (Rigorous) Standards. In addition to these requirements, uniformity of the tubes is assured by eddy current testing, and the tubes are seal welded to the tube sheet to decrease the possibility of leakage. All tube welded ends are liquid penetrant tested to assure the absence of welding flaws. The heat exchangers have been fabricated with surface areas greater than those dictated by the most severe heat transfer conditions.

Valves

All remotely operated valves in the emergency core cooling systems are manufactured and inspected in accordance with the intent of the ASME Nuclear Power Piping Code B31.7. Liquid penetrant, radiography, ultrasonic, and hydrotesting is performed as the Code classification requires.

The seats and discs of these valves are manufactured from materials which will be free from galling and seizing. All valve material is certified to be in accordance with ASTM specifications. All remotely operated valves in these systems are of the backseating type and equipped with stem leak-off provisions.

Seismic Design

Components in the emergency core cooling system are designated as Class I equipment and are designed to maintain their functional integrity during earthquake (2.6).

Instrumentation

The Engineered Safeguards Actuation instrumentation for the emergency core cooling system is provided with redundant channels and signals as described in Section 7. The control room layout is arranged so that all indicators and alarms are grouped in one sector at a convenient location for viewing. Switches and controls are also located conveniently.

Quality Control

Quality Standards for the emergency core cooling system components are given in Table 6-5.

6.1.2.5 Coolant Storage

The letdown storage tank has a total coolant volume of 600 ft^3 and normally contains approximately 2,600 gallons of water. This tank provides water to the high pressure injection pumps until the borated water storage tank outlet valves are opened. The letdown storage tank is designed and inspected in accordance with the requirements of ASME III-C.

Each unit is provided with a borated water storage tank as described in Section 9.

Provisions are made for sampling the water and adding concentrated boric acid solution or demineralized water.

Each core flooding tank contains approximately 7,000 gallons of borated water at a concentration of 2,270 ppm of boron.

6.1.2.6 Pump Characteristics

Curves of total dynamic head and NPSH versus flow are shown in Figure 6-6 for the high pressure pumps and in Figure 6-7 for the low pressure pumps.

6.1.2.7 Heat Exchanger Characteristics

The LPI coolers are designed to remove the decay heat generated during a normal shutdown. In addition, each cooler is capable of cooling the injection water during the recirculation mode following a loss of coolant accident to provide for removal of decay heat which provides adequate core cooling. The heat transfer capability of the low pressure injection cooler as a function of recirculated water temperature is illustrated in Figure 6-8.

6.1.2.8 Relief Valve Settings

Relief valves are provided to protect the low pressure piping and components from overpressure. These relief valves will be set at 350 psig, the system design pressure.
9. (Reference Supplement 6 revisions for Oconee 1.)

6.1.2.9 Reliability Considerations

System reliability is assured by the system functional design including the use of normally operating equipment for safety functions, testability provisions, and equipment redundancy; by proper component selection; by physical protection and arrangement of the system; and by compliance with the intent of the AEC General Design Criteria. There is sufficient redundancy in the emergency core cooling system to assure that no credible single failure can lead to significant physical disarrangement of the core. This is demonstrated by the single failure analysis presented in Table 6-2. This analysis was based on the assumption that a major loss-of-coolant accident had occurred and coincidentally an additional malfunction or failure occurred in the engineered safeguards system. For example, the analysis included malfunctions or failures such as electrical circuit or motor failures, valve operator failures, etc. It was considered incredible that valves would change to the opposite position by accident if they were in the required position when the accident occurred. Table 6-2 also presents an analysis of possible malfunctions of the core flooding tanks that could reduce their post-accident availability. It is shown that these malfunctions result in indications that will be obvious to the operators so appropriate action can be taken. In general, failures of the type assumed in this analysis are considered highly improbable since a program of periodic testing will be incorporated in the station operating procedures. The adequacy of equipment sizes in the ECCS is demonstrated by the post-accident performance analysis described in Section 14. This analysis shows that only one high pressure injection pump, one low pressure injection pump, and one low pressure injection cooler in combination with the core flooding tanks is required to protect against the full spectrum of break sizes (Figure 14-55). Two of each are normally available.

6.1.2.10 Missile Protection

Protection against missile damage is provided by either direct shielding or by physical separation of duplicate equipment. For most of the routing inside the reactor building, the ECCS piping will be outside the primary and secondary shielding, and hence, protected from missiles originating within these areas. The portions of the injection lines located between the primary reactor shield and the reactor vessel wall are not subject to missile damage because there are no credible sources of missiles in this area.

The high pressure injection lines enter the reactor building via penetrations on opposite sides of the building. Each injection line splits into two lines inside the reactor building, but outside the secondary (missile) shield, to provide four injection paths to the reactor coolant system. The four connections to the reactor coolant system are located between the reactor coolant pump discharge and the reactor inlet nozzles. There are four injection lines penetrating the missile shield, minimizing the effect on injection flow in the unlikely event of missile damage to the injection lines inside the secondary shield.

Protection from missiles is given to the low pressure injection lines within the reactor building. The portion of the low pressure injection system located in the reactor building consists of two redundant injection lines which are connected to injection nozzles located on opposite sides of the vessel. Both redundant suction lines from the sump are missile protected. The sump suction is located outside of the secondary shielding and is additionally protected by a grating.

The entire core flooding system is located within the reactor building. The core flooding tanks and two of the three valves in each core flooding line are located outside of the secondary shield.

6.1.2.11 Actuation

The high pressure injection system is actuated automatically by a low reactor coolant system pressure of 1,500 psig or by a reactor building pressure of 4 psig. All of the pumps and valves can also be remotely operated from the control room.

The low pressure injection system is automatically actuated by a low reactor
8. coolant system pressure of 500 psig or reactor building pressure of 4 psig. All of the pumps and automatic valves can also be remotely operated from the control room.

The core flooding system is actuated at a reactor coolant system pressure of 600 psig. At this point the differential pressure across the inline check valves allows them to open releasing the contents of the tanks into the reactor vessel.

A description of the Engineered Safeguards Protective System is given in Section 7.1.3. Table 7-2 gives actuation set points for all of the systems discussed.

Table 6-2

Single Failure Analysis - Emergency Core Cooling System

Component	Malfunction	Comments
A. High Pressure Injection System		
1. Suction valve for high pressure injection pump from borated water storage tank.	Fails to open.	The parallel valve will supply the required flow to one pump string.
2. High pressure injection valve.	Fails to open.	The alternate line will provide the total amount of flow required for protection.
3. High pressure injection pump (operating).	Fails (stops).	Two backup pumps are available to deliver the flow; however, only one is required for protection.
4. High pressure injection pump.	Fails to start.	Two backup pumps are available to deliver the flow; however, only one is required for protection.
5. Seal return line isolation valve.	Fails to close on ES signal.	The other isolation valve will close eliminating this fluid path.
6. Letdown cooler isolation valve.	Fails to close on ES signal.	The other isolation valve will close the flow path.

Table 6-2 (Cont'd)
Single Failure Analysis - Emergency Core Cooling System

Component	Malfunction	Comments
B. Low Pressure Injection System		
(INJECTION FROM BORATED WATER STORAGE TANK)		
1. Low pressure injection pump.	Fails to start.	Adequate injection is provided by the other pump.
2. Low pressure injection isolation valve.	Fails to open.	Other line admits necessary flow. Shutoff pump in associated line until valve can be opened.
3. Valve in suction line from BWST.	Fails to open.	Other line admits necessary flow.
(RECIRCULATION FROM REACTOR BUILDING EMERGENCY SUMP)		
1. Valve in suction line from emergency sump.	Fails to open.	Other line admits necessary flow.
2. Valve in suction line from BWST.	Fails to close after initiating recirculation.	Check valve prevents flow into BWST.
3. Low pressure injection pump.	Loss of pump.	Reactor core protection will be maintained by alternate pump and low pressure injection string.
C. Core Flooding System		
1. Isolation valve in discharge line.	Closes during normal operation.	If the valve cannot be manually opened, the reactor must be shut down or operations limited as specified in Technical Specifications.
2. Tank relief valve.	Opens during normal operation.	Loss of nitrogen pressure and consequent loss of ability of tank to perform. Reactor must be shut down or operations adjusted to Technical Specification limits and relief valve must be repaired.

Table 6-2 (Cont'd)
Single Failure Analysis - Emergency Core Cooling System

Component	Malfunction	Comments
3. Check valves in discharge line.	Excessive leak detected during normal reactor operation.	It is extremely unlikely that both check valves would permit excessive leakage. Leakage would be indicated by core flooding tank pressure and level changes. If leakage becomes progressively worse or is unacceptably high, reactor must be shut down while the check valves are repaired.

6.1.2.12 Environmental Considerations

The major operating components of the emergency core cooling system are external to the reactor building and will not be exposed to the post-accident building environment.

Electrical and mechanical equipment within the reactor building which are required to be operable during and subsequent to a LOCA and/or a steam line break are:

a. Reactor coolant system pressure transmitters.

b. Reactor building isolation valves and associated position indications.

c. Reactor building air cooling unit fans and cooling coils.

5. d. Instrument cables for radiation, pressure, level and valve position instruments.

e. Power cables for the reactor building fan motors and isolation valves.

Non-nuclear instrumentation (Item a) in the Reactor Protection System and the Engineered Safeguards System located inside the reactor building are qualified in accordance with Criteria for Nuclear Power Plant Protection Systems, IEEE No. 279 dated August 30, 1969, to establish the adequacy of equipment performance in the LOCA environment.

A valve operator similar to those being used (Item b) was satisfactorily tested for performance under conditions expected to exist in the reactor building after the LOCA. The operator was tested in accordance with Level 4 of the Standard Draft, dated June 7, 1968, prepared by Sub-Committee 2 (Equipment Qualification
5. Testing) of the IEEE/NSG/Technical Committee for Standards.

A scaled down reactor building cooling coil unit, (a 24 x 24 inch section identical in construction with the full-size unit), has been satisfactorily tested under post-accident conditions. The maximum test conditions were 70 psig, 286 F and 100% relative humidity.

Other equipment and components located in the primary containment or elsewhere in the plant must be operable during and subsequent to a loss-of-coolant or steam-line-break accident and are as follows:

a. Equipment Outside Containment

Safety related equipment and components which are located outside the containment and which therefore are not subject to the abnormal environmental conditions present within the containment during an accident are given operational performance tests on either the actual equipment
4. or prototype units. A list of the equipment located outside the containment is tabulated in Table 6-2a.

b. Equipment Inside Containment

Safety related equipment located within the containment is qualified for the application by tests to demonstrate operability under the accident environment. A list of this equipment along with a brief description of the qualification tests is tabulated in Table 6-2b.

Instrument transmitters and electric motor valve operators in the reactor protection system and the engineered safeguards sytem located inside the reactor building were designed to withstand the potential effects of radiation due to normal and accident conditions. Non-metallic materials and lubricants were selected on the basis of their susceptability to radiation damage demonstrated by irradiation tests. The instrument transmitters were successfully irradiation tested at the Babcock & Wilcox Nuclear Development Center (NDC). The transmitters with two dosimeters attached to each were placed in a sealed aluminum box and positioned near fuel elements in the NDC storage pool. The test was conducted in two parts; the first part simulated the environmental dose to the transmitters associated with the 40-year design plant lifetime, and the second part simulated the maximum expected dose to the transmitters associated with a LOCA. The non-metallic materials selected for the electric motor valve operators based on irradiation testing are: melamine used in the limit switches (all plastic material used in melamine), viton for all seals, Humble Nebula EP #1 as the lubricant, and Class "H" insulation for the motor.

TABLE 6.2a

Equipment Operation During An Accident and Located Outside Containment

4160 Volt Station Auxiliary Switchgear

600 Volt Load Centers

600 Volt and 208 Volt Motor Control Center

Batteries

Chargers and Inverters

Panelboards

Low Pressure Injection Pump Motors

High Pressure Injection Pump Motors

Reactor Building Spray Pump Motors

Low Pressure Service Water Pump Motors

Cables

TABLE 6.2b

Equipment Operational During an Accident and Located Within the Containment

Equipment	Accident Environmental Tests
Reactor Building Cooling Fan and Motors	After preaging a prototype motor for an extrapolated 40 years insulation life and testing it for operation under seismic conditions, the motor and fan assembly was placed in a pressure vessel. Steam is then injected into the chamber to a pressure of 70 to 80 psia and chemicals similar to those used in the spray system are introduced. The pressure cycle is repeated four additional times and then pressure is reduced to the level to be expected following the accident. The motor then is run continuously for a minimum of 7 days in the test chamber.
Cables	Representative samples of preaged cables are tested under high pressure, temperature and humidity conditions equal to or exceeding those specified for the LOCA. These cables are preaged for forty years of radiation and temperature prior to testing.
Valves	A typical production valve and actuator was tested as follows under simulated accident conditions. After preaging heat test and shock and vibration tests, the production valve and actuator was subjected to the following environmental tests: a) Saturated steam at 90 psig for one hour. b) Boric acid spray for the next two hours at 70 psig followed by a pressure drop to 40 psig with the spray continuing for an additional half hour.

Equipment	Accident Environmental Tests
	c) Steam pressure was maintained at 40 psig for a period of 1-1/2 hours followed by a pressure dropoff to 20 psig. d) Steam pressure was maintained at 20 psig for the remaining nineteen hours of the first day followed by a pressure decrease to 10 psig. e) Steam pressure at 10 psig was then maintained for six days yielding a total test time of seven days. f) Valve operation was conducted at the beginning and the end of each level of pressure in a, b, c, d, and e above.
Penetrations	Qualification tests have been performed on one production assembly of each type that is required to function during or following the loss-of-coolant accident to verify its functional capability. The interior end of the penetration assemblies were subjected to the following emergency conditions at 100% relative humidity in an autoclave built to duplicate Oconee reactor building concrete and nozzle design. 1. First fifteen minutes: Pressure of 65 psig at a temperature of 300°F Rise time for normal operating conditions - less than ten seconds. 2. Next forty-five minutes: Pressure of 40 psig at a temperature of 260°F. 3. Next twenty-three hours: Pressure of 35 psig at a temperature of 250°F. During the environmental tests, functional capability was demonstrated by applying rated current to conductors in series at 600 volts r.m.s. above ground. The temperature along the nozzle and in the wire bundles was

TABLE 6.2b Cont'd

Equipment | **Accident Environmental Tests**

monitored throughout the test. Leak rate was measured and recorded during the test.

The following tests were performed before and after the autoclave test.

1. Connector and conductor resistance test. Measured ohmic resistance of each conductor.
2. Dielectric withstand tests.
   Conductor to ground and conductor to conductor.
3. Insulation resistance.
   Conductor to ground.
4. Leak rate test.

Reactor Coolant System Pressure Transmitters.

Reactor coolant pressure transmitters required for use within the reactor building following an accident have been conservatively tested under conditions simulating the environment expected after the design base 14.1 ft^2 LOCA. The results of these tests show that the transmitters are acceptable for the required functions.

A three (3) phase test was performed to simulate the post-LOCA reactor building environment. The respective phases are given below:

9.

Phase I - Pre-accident test at the Babcock & Wilcox Nuclear Development Center (NDC) to simulate the environmental dose to the transmitters associated with the 40-year plant design lifetime.

Phase II - Environmental autoclave test at Franklin Institute Research Laboratory to simulate the reactor building pressure and temperature history for an LOCA.

TABLE 6.2b Cont'd

Equipment

Reactor Coolant System Pressure Transmitters (Cont'd)

Accident Environmental Tests

Phase III - Post-Accident test at NDC to simulate the maximum expected dose to the transmitters after an LOCA.

Phase I consisted of irradiating the transmitters while the units were in a nonoperating mode. The transmitters were placed in a sealed aluminum box with two dosimeters attachment to each transmitter and positioned over two reactor fuel elements in the NDC storage pool.

Phase II consisted of exposing the transmitters in the operating mode to a steam environment in a test autoclave for 24 hours. The units were supplied with a constant input of approximately 2/3 of full range and the resultant output/input ratio was measured for the test duration.

Phase III consisted of irradiating the transmitters while the units were in the operating mode in much the
9. same manner as Phase I except that the box was lowered into position beside one fuel element from the reactor. A constant input of approximately 2/3 of full range was maintained throughout the test.

The resulting output signal inaccuracies for each test phase were analyzed and found to be acceptable.

6.1.3 DESIGN EVALUATION

In establishing the required component redundancy for the emergency core cooling system, several factors related to equipment availability were considered:

a. The probability of a major reactor coolant system failure is very low; i.e., the probability that the equipment will be needed to serve its emergency function is low.

b. The fractional part of a given component lifetime for which the component is unavailable due to maintenance is estimated to be very small. On this basis, the probability that a major reactor coolant system accident would occur while a component from the emergency core cooling system was out of service for maintenance is several orders of magnitude below the low basic accident probability.

c. The maintenance period for important equipment can usually be scheduled for a period of time when the reactor is shut down. Where maintenance of an engineered safety feature component is required during operation, the periodic test frequency of the similar redundant components can be increased to insure availability.

d. Where the systems are designed so that the components serve a normal function in addition to the emergency function or where meaningful periodic tests can be performed, there is also a low probability that the required emergency action would not be performed when needed; i.e., equipment reliability is improved by using the equipment for other than emergency functions.

6.1.3.1 High Pressure Injection System (HPI)

17. One high pressure injection string can deliver 450 gpm at 585 psig reactor
vessel pressure. The safety analysis in Section 14 has shown that one high
pressure injection pump is sufficient to prevent core damage for those smaller
leak sizes which do not allow the reactor coolant system pressure to decrease
rapidly to the point where the low pressure injection system is initiated.
17. After receiving an actuation signal, the HPI system valves for injection will
reach full open within 14 seconds and the HPI pumps will reach full speed
within 6 seconds. One of the three high pressure injection pumps is normally
in operation and a positive static head of water assures that all pipe lines
are filled with coolant. The high pressure injection lines contain thermal
sleeves at their connections into the reactor coolant piping to prevent over
stressing the pipe juncture.

Operation of this system does not depend on any portion of another engineered
safety feature. The system can be operated in conjunction with the Low Pressure
Injection System if the HPI system must be operated in the recirculation mode.

6.1.3.2 Low Pressure Injection and Core Flooding Systems

Two pumps will deliver 6,000 gpm to the reactor vessel through two separate
injection lines. One pump can deliver approximately 3,000 gpm to the reactor
17. vessel at 100 psig. Assuming the reactor had been operating at full power prior
to the accident, the decay heat being generated in the core at 30 minutes after
the accident is approximately 1.8% of full power, or 160 x 10^6 Btu/h. One low
pressure injection pump and cooler combination is capable of removing the heat
energy generated after loss-of-coolant accident.

After receiving an actuation signal, the low pressure injection valves will
3,17. reach full open within 15 seconds and the low pressure injection pumps will
reach full speed within 8 seconds.

Injection response of the core flooding system is dependent upon the rate of
reduction of reactor coolant system pressure. For the maximum pipe break (14.1
ft^2), the core flooding system is capable of reflooding the core to the hot
spot in less than 25 seconds after a rupture has occurred.

Special attention has been given to the design of core flooding nozzles to assure
that they will take the differential temperature imposed by the accident con-
dition. Special attention has also been given to the ability of the injection
lines to absorb the expansion resulting from the recirculating water temperature.

The low pressure injection system is connected with other safeguards systems
in three respects, i.e., (1) the high pressure and low pressure injection sys-
tems and the reactor building spray system take their suction from the borated
water storage tank; (2) the low pressure injection pumps and the reactor
building spray pumps share common suction lines from the reactor building sump
during the coolant recirculation mode; and (3) the low pressure injection system
and the core flooding system utilize common injection nozzles on the reactor
vessel.

Rev. 3. 3/16/70
Rev. 17. 12/17/71

6.1.3.3 Loss of Normal Power Source

Following a loss-of-coolant accident assuming a simultaneous loss of normal power sources, the emergency power source and both the low pressure and high pressure injection systems will be in full operation within 25 seconds after actuation. All calculations for the Oconee Units have assumed a 25 second delay from receipt of the actuation signal to start of flow for both the HPI and LPI systems. Upon loss of normal power sources including the startup
21. source and initiation of an engineered safeguards signal, the 4160 volt engineered safeguards power line is connected to the underground feeder from
17. Keowee hydro (Section 8.2.3). The Keowee hydro unit will startup and accelerate to full speed in 23 seconds or less. An analysis has shown that by energizing the HPI and LPI valves (which have opening times of 14 seconds and 15 seconds respectively at normal bus voltage) and pumps at less than 100% voltage and frequency the design injection flow rate (HPI - 450 gpm, LPI-3000 gpm) will be obtained within 25 seconds.

6.1.4 TESTS AND INSPECTIONS

3.

All active components, listed in Table 6-3, of the emergency injection system will be tested periodically to demonstrate system readiness. The high pressure injection system will be inspected periodically during normal operation for leaks from pump seals, valve packing, and flanged joints. During operational testing of the low pressure injection pumps, the portion of the system subjected to pump pressure will be inspected for leaks. Items for inspection will be pump seals, valve packing, flange gaskets, heat exchangers, and safety valves for leaks to atmosphere.

Table 6-3

Emergency Core Cooling Systems Performance Testing

High Pressure Injection Pumps	One of two pumps operates continuously. The other pump will be operated periodically.
High Pressure Injection Line Valves	The remotely operated stop valves in each line are opened partially one at a time. The flow monitors will indicate flow through the lines.
18. High Pressure Injection Pump Suction Valves	The valves are opened and closed individually and console lights monitored to indicate valve position.
Low Pressure Injection Pumps	Pumps are used in normal service for shutdown cooling. These pumps are tested singly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.
Borated Water Storage Tank Outlet Valves	The operational readiness of these valves is established in completing the pump operational test discussed above. During this test, each valve is tested separately.

Rev. 3. 3/16/70
Rev. 17. 12/17/71
Rev. 18. 3/10/72
Rev. 21. 7/26/72

Table 6-3 (Cont'd)

Emergency Core Cooling Systems Performance Testing

Low Pressure Injection Valves	With pumps shut down and borated water storage tank outlet valves closed, these valves can be opened and reclosed by operator action.
Sump Recirculation Suction Valves	With low pressure injection pumps shut down, operation of these valves can be checked.
Check Valves in Core Flooding Injection Lines	With the reactor shut down, the check valves in each core flooding line are checked for operability by closing the isolation valves, reducing the reactor coolant system pressure to provide a ΔP slightly above the check valve opening pressure, and opening the isolation valves. Check valve operability is shown by tank pressure and level changes.

6.2 REACTOR BUILDING SPRAY SYSTEM

6.2.1 DESIGN BASIS

The reactor building spray system is designed to provide building atmosphere cooling to limit post-accident building pressure to less than the design value and to reduce it to nearly atmospheric pressure.

6.2.2 SYSTEM DESIGN

6.2.2.1 Piping and Instrumentation Diagram

A schematic diagram of the system is shown in Figure 6-3. The system serves no function during normal operation.

Removal of post-accident energy is accomplished by directing borated water spray into the reactor building atmosphere. The system consists of two pumps, two reactor building spray headers, isolation valves, and the necessary piping, instrumentation and controls. The pumps and remotely operated valves for each unit can be operated from the control room. The reactor building spray system is sized to furnish 100% of the design cooling capacity (240×10^6 Btu/h) with both of the spray paths in operation. Both paths operate independently, and the reactor building spray system also operates separately from the reactor building cooling units (Section 6.3) which independently possesses full post-accident cooling capability.

A high reactor building pressure signal of 10 psig from the engineered safeguards protective system initiates operation. The two pumps start, taking suction initially from the borated water storage tank through the intertie with the low pressure injection system, and initiate building spray through the spray headers and nozzles. After the water in the borated water storage tank reaches a low level, the spray pump suction is transferred to the reactor building sump automatically when the operator places the low pressure injection system in the recirculation mode. The reactor building emergency sump water is cooled by the low pressure injection system as described in 6.1.

6.2.2.2 Codes and Standards

The equipment is designed to the applicable codes and standards given in Section 9.

6.2.2.3 Material Compatibility

All materials are compatible with the reactor coolant. The major components of the system are constructed of stainless steel. Minor parts such as pump seals utilize other corrosion resistant materials.

6.2.2.4 Component Design

Pumps

The reactor building spray pumps are similar to those used in refinery service. These pumps are liquid-penetrant tested by methods described in the ASME Boiler and Pressure Vessel Code Section VIII and are hydrotested and qualified to be able to withstand pressures greater than 1.5 times the design pressure. The pumps are designed so that periodic testing may be performed to assure operability at all times.

Valves

The remotely operated valves of the reactor building spray system are designed and manufactured to the same requirements as the valves in the emergency core cooling systems. Refer to Section 6.1.2.4.

Spray Headers and Nozzles

120 full core spray nozzles are arranged on each of the two reactor building spray headers. The spray nozzles are spaced in the headers to give uniform spray coverage of the reactor building volume above the operating floor.

Piping

Except for the sections of lines requiring flanged connections for maintenance, the entire system is welded construction. Table 6-6 lists the design conditions for this system.

Quality Control

Quality standards for the reactor building spray system components are given in Table 6-5.

6.2.2.5 Coolant Storage

This system shares borated storage tank capacity with the low pressure injection system and the high pressure injection system.

6.2.2.6 Pump Characteristics

Curves of total dynamic head and NPSH versus flow are shown in Figure 6-9.

6.2.2.7 Reliability Considerations

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is shown in Table 6-4.

Table 6-4
Single Failure Analysis Reactor Building Spray System

Component	Malfunction	Comments
1. Reactor building spray pump.	Fails to start.	Since each of the two strings of the reactor building spray system is equally sized, the remaining string will provide heat removal capability at a reduced rate. In combination with the reactor building cooling system, heat removal capability in excess of the requirements will be provided.
2. Building isolation valve.	Fails to open.	(Same as above)
3. Check valve in suction or discharge line.	Fails to open.	(Same as above)

6.2.2.8 Missile Protection

Protection against missile damage is provided by direct shielding or by physical separation of duplicate equipment. The spray headers are located outside and above the primary and secondary concrete shield.

6.2.2.9 Actuation

The reactor building spray system will be activated at a reactor building pressure of 10 psig (Section 7). The system components may also be actuated by operator action from the control room for performance testing.

6.2.2.10 Environmental Considerations

None of the active components of the reactor building spray system are located within the reactor building, so none are required to operate in the steam-air environment produced by the accident.

6.2.3 DESIGN EVALUATION

The reactor building spray system, acting independently of the reactor building cooling system, is capable of limiting the containment pressure after a loss-of-coolant accident to a level below the design pressure. The reactor building spray system is at least equivalent in heat removal capacity to the cooling system. The reactor building spray system is designed for long term post-accident operation. In combination with cooling units, it affords redundant alternative methods to maintain containment pressure at a level below design pressure. Any of the following combinations of equipment will provide sufficient heat removal capability to accomplish this:

a. The reactor building spray system alone.

b. Three cooling units alone.

c. Two cooling units and the reactor building spray system at one-half capacity.

The reactor building spray system will deliver 3,000 gpm through the spray nozzles within 37.5 seconds after the reactor building reaches 10 psig.

6.2.4 TESTS AND INSPECTION

The active components of the reactor building spray system can be tested as follows:

Reactor Building Spray Pumps

The delivery capability of one pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

Borated Water Storage Tank Outlet Valves

These valves will be tested in performing the pump test above.

Reactor Building Spray Injection Valves

With the pumps shut down and the borated water storage tank outlet valves closed, these valves can each be opened and closed by operator action.

Reactor Building Spray Nozzles

With the reactor building spray inlet valves closed, low pressure air or fog can be blown through the test connections. Visual observation will indicate flow paths are open.

During these tests, the equipment can be visually inspected for leaks. Valves and pumps will be operated and inspected following maintenance on the system to assure proper operation.

6.3 REACTOR BUILDING COOLING SYSTEM

6.3.1 DESIGN BASIS

The reactor building cooling systems are designed to remove the heat in the containment atmosphere after an accident to prevent the building pressure from exceeding the design pressure.

6.3.2 SYSTEM DESIGN

6.3.2.1 Piping and Instrumentation Diagram

Figures 6-4 and 6-10 illustrate the cooling systems. Each cooling unit consists of a fan, a tube cooler, and the required distribution duct work. The reactor building atmosphere is circulated past the cooling tubes by the fan and returned to the building. Cooling water for the cooling units is supplied by the low pressure service water system. During normal operation these units, with two fans operating, serve to cool the reactor building atmosphere. Upon receipt of the signal from the Engineered Safeguards Actuation System, the two operating fans reduce speed and the third fan starts at reduced speed.

Performance of the cooling system is monitored by flow instrumentation in the service water return line from each cooler and by the reactor building temperature and pressure instrumentation.

A more complete description of the reactor building normal ventilation system is given in Section 5.3.

6.3.2.2 Codes and Standards

The cooling surfaces are constructed in accordance with TEMA guidelines. The header system for the coolers is designed and fabricated to the requirements of USAS B31.1. The low pressure service water system is designed to USAS B31.1.

6.3.2.3 Materials Compatibility

The materials for the reactor building coolers have been selected to be compatible with the use of untreated service water to minimize corrosion in accordance with TEMA guidelines.

6.3.2.4 Component Design

Coolers

The cooling surface of the cooling units has been designed for and satisfactorily tested under simulated post-accident conditions. A conservative design has resulted in a heat exchanger which has heat transfer capability in excess of the expected heat transfer requirements.

Ductwork

The ductwork which must function during normal operations has additional features to assure that it is operable during the accident. In the event that the accident imposes severe stresses on the lower portion of the ductwork, causing possible collapse or deformation, the upper sections are equipped with "drop-out" plates.

6.3.2.5 Reactor Building Circulating Fan Characteristics

Circulation of the reactor building atmosphere under accident conditions is by the same fans used for normal ventilation. Upon actuation by an engineered safeguards signal, the fan motors switch from full speed to half speed and the spare unit is started at half speed (6.3.2.9). Prototype fan motors combination testing has demonstrated the capability to supply design flow of steam-air mixture through the coolers.

6.3.2.6 Reactor Building Cooler Characteristics

The reactor building cooler is located in the discharge ducting for the fan. The air-stream mixture flows across the tube bank, resulting in condensation of a portion of the steam and removal of sensible heat from the air. The rated capacity of each unit is 80 x 10^6 Btu/h. Figure 6-11 shows the design heat transfer capability of each unit at various reactor building temperature conditions. Figure 6-11 is based on a low pressure service water temperature of 75 F. Actually, the cooling water is drawn from a point near the bottom of the lake and the anticipated service water temperature would be in the range of 45 to 76 F. Therefore, the curve shown in Figure 6-11 is conservative.

5.

Figure 6-11a shows how the reactor building cooling rate varies with the air-steam mixture flow rate. It can be seen that even if the mixture flow rate decreases by 40 percent, the cooling capability decreases by less than 7 percent.

6.3.2.7 Reliability Considerations

Inside the reactor building, the cooling units are located outside the secondary shield at an elevation above the water level in the bottom of the reactor building during post-accident conditions. In this location the units are protected from being flooded.

The major equipment of the reactor building cooling units is arranged in three independent strings with three duplicate service water supply lines. In the unlikely event of a failure in one of the three cooling units, the reactor building spray system independently, or half of the reactor building spray system capacity combined with the remaining two cooling units, will provide cooling capacity in excess of that required. Fan-motor operation under design LOCA condition has been demonstrated by prototype test.

A failure analysis of the cooling units is presented in Table 6-7.

6.3.2.8 Missile Protection

The cooling units and associated piping are located outside the secondary concrete shielding. The ductwork required to operate during an accident is located outside of the secondary shielding.

6.3.2.9 Actuation

In the event of a loss-of-coolant accident, the cooling system is initiated at a reactor building pressure of 4 psig. The cooling units are placed in operation as follows:

a. The low pressure service water valves at the discharge of the coolers open wide. Normally, these valves are operating with an intermediate setting (235 gpm/loop). The loop flow under accident conditions is 1,400 gpm.

b. The spare cooling unit fan is started; and the speed of all fans is switched to reduced speed to reduce the horsepower requirements generated by the denser building atmosphere.

c. The fusible dropout plates in the duct work melt and drop off, assuring that a positive path for recirculation of the reactor building atmosphere is available.

d. Depending upon the severity of the accident, the blowout plates at the top of the downcomer will be forced out, allowing attenuation of any possible shock waves before they reach the cooling coils.

6.3.2.10 Environmental Considerations

Figure 6-12 depicts the reactor building post-accident steam-air conditions. The fans and motors are designed for operation in the post-accident conditions. Cooling capability of the coolers has been satisfactorily tested in this environment.

6.3.3 DESIGN EVALUATION

The reactor building cooling system provides the design heat removal capacity following a loss-of-coolant accident with all three coolers operating by continuously circulating the steam-air mixture past the cooling tubes to transfer heat from the containment atmosphere to the low pressure service water.

Building pressure is limited below the design pressure. The design heat load at these conditions is 240 x 10^6 Btu/h. The design inlet cooling water is 75 F, although the expected cooling water range is 45 - 75 F. The heat removal capacity for these units is shown in Figure 6-11. The safety analysis given in Section 14 demonstrates system effectiveness.

6.3.4 TESTS AND INSPECTION

The equipment, piping, valves, and instrumentation are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

The cooling units will be tested periodically as follows:

a. The fans can be started and inspected for proper operation.

b. The return line service water valves will be opened, and the lines checked for flow.

Table 6-5
Quality Control Standards for Engineered Safeguards Systems

Summary of Requirements
Core Flooding Tanks

CLASSIFICATION: ASME III, Class C, Paragraph N-2113 and the requirements of ASME VIII, Paragraph UW-2(a) (lethal substances)

Inspection Requirements	Acceptance Standard
1. Inspection of raw materials and review of material certificates	ASME III
2. Hydro test	ASME III
3. Radiograph	ASME VIII

Summary of Requirements for Low Pressure
Injection Heat Exchanger

CLASSIFICATION: Shell ASME VIII, Tube ASME III, Class C (lethal)

Inspection Requirements	Acceptance Standard
1. Inspection of raw materials and review of material certificates	ASME II, III, VIII
2. Seal weld on tube-to-tube sheet	TEMA-R-7 and additional requirements
3. Liquid penetrant inspection on tube-to-tube sheet weld	ASME III, N-627 and additional requirements
4. Hydro test	ASME III, VIII, TEMA-R
5. Leak test and seal weld (air)	

Summary of Requirements for Valves

Inspection Requirements	Acceptance Standard
Class I and II Valves	
1. Radiographic inspection of the body casting	USAS B31.7
2. Inspection of material and review of material certificates	USAS B31.7
3. Liquid penetrant inspection of the valve body	USAS B31.7

Table 6-5 (Cont'd)
Quality Control Standards for Engineered Safeguards Systems

Inspection Requirements	Acceptance Standard
4. Hydro test of valve assembly	USAS B16.5 and additional requirements
5. Seat leakage test	MSS-SP-61 and additional requirements
Class III Valves	
1. Inspection of material and review of material certificates	USAS B31.7
2. Hydro test of valve assembly	USAS B16.5
3. Seat leakage test	MSS-SP-61 and additional requirements

In addition to these inspections listed above, all valve materials must meet the ASTM material specification.

Summary of Requirements for
Engineered Safeguards Systems Pumps

Inspection Requirements	Acceptance Standard
1. Inspection of materials and review of material certificates	ASTM
2. Liquid-penetrant inspection of castings	ASME VIII
3. Performance test	Hydraulic Institute Standard

Additional requirements:

Low Pressure Injection Pumps

1. Hydrotest casing to 600 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This exceeds the hydrotest requirements of ASME VIII Paragraph UG-99 (>1.5 x design pressure).

Reactor Building Spray Pumps

1. Hydrotest casing to 1,200 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This exceeds the hydrotest requirements of ASME VIII Paragraph UG-99 (>1.5 x design pressure).

Table 6-5 (Cont'd)
Quality Control Standards for Engineered Safeguards Systems

High Pressure Injection Pumps

1. Ultrasonic examination of pump barrel.

3. 2. Hydrotest nozzle head and pump barrel to 4,575 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This meets the hydrotest requirements of ASME VIII Paragraph UG-99
3. (1.5 x design pressure of 3,050 psig).

Low Pressure Service Water Pumps

1. Documented quality control records and certified caliper measurements of the entire casing thickness will be furnished.

2. Witness performance test will be performed. Acceptance standards are per Hydraulic Institute Standard.

3. A documented, non-witness, hydro-test will be performed. Acceptance standards are per ASME code.

Table 6-6
Engineered Safeguards Piping Design Conditions

		Temp (°F)	Press. (psig)
	1. High Pressure Injection System		
	a. From the pump discharge to upstream of the stop check valves inside the secondary shielding.	200	3,050
3.	b. High pressure injection pump.	200/150	2,800/3,050
	c. From upstream of the stop check valves to the reactor inlet line.	650	2,500
16.	2. Low Pressure Injection System (Reference Supplement 9 Revisions for Oconee 3)		
	a. From the borated water storage tank to upstream of the borated water storage tank outlet valves.	150	Static
	b. From upstream of the borated water storage tank outlet valve to upstream of the electric motor operated valves in the borated water feed lines.	200	100

Rev. 3. 3/16/70
Rev. 16. 7/30/71

			Temp (^oF)	Press. (psig)
	c.	From upstream of the electric motor operated valves in the borated water feed lines to upstream of the valves at the pump inlets.	300	200
	d.	From upstream of the system inlet valves at the pump inlets to downstream of the manual cooler isolation valves.	300/250	470/505
	e.	From downstream of the manual cooler isolation valves to upstream of the throttle valve at the cooler discharge.	300	370
16.	f.	From upstream of the throttle valves at the cooler discharge to upstream of the reactor building isolation valves.	300/250	470/505
	g.	From upstream of the system inlet valves to upstream of the check valves in the core flooding lines.	300	2,500
	h.	From upstream of the check valves in the core flooding lines to the reactor vessel.	650	2,500
	i.	From the reactor building emergency sump to upstream of the valves in the recirculation lines.	300	59
	3.	Reactor Building Spray System		
	a.	From downstream of the pump inlet valves to downstream of the reactor building valves.	300	500
	b.	From downstream of the inlet valves through the nozzles.	300	200
	4.	Low Pressure Service Water System		
5.	a.	Condenser circulating water crossover to low pressure service water pump suction.	100	50
	b.	Pump discharge	100	150

Table 6-7
Single Failure Analysis for Reactor
Building Cooling System

	Component	Malfunction	Comments
1.	Circulating fan	Fails to operate.	The cooling capacity of the cooling units is reduced; however, the reactor building spray system provides separate full capacity cooling.
2.	Cooler service water outlet valve.	Fails to open fully.	Valve will normally be partially open. If the valve fails to open fully, the unit will operate under reduced heat removal capability. The reactor building spray system provides full heat removal capability.
3.	Cooler service water inlet valve.	Inadvertently left closed.	The flow through this string will be unavailable for cooling. It is unlikely that this condition would occur during an accident since the position and flow are monitored during normal operation. The reactor building spray system will provide the required cooling.
4.	Service water pump.	Fails to operate.	The two remaining pumps will provide full low pressure service water flow to all components.

6.4 REACTOR BUILDING PENETRATION ROOM VENTILATION SYSTEM

6.4.1 DESIGN BASES

This system is designed to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post-accident reactor building leaks. Experience(2) has shown that reactor building leakage is more likely at penetrations than through the liner plates or weld joints.

6.4.2 SYSTEM DESIGN

6.4.2.1 Piping and Instrumentation Diagram

The system schematic and characteristics are shown on Figures 6-5 and 6-5a,
5. respectively. Penetration rooms are formed adjacent to the outside surface of each reactor building by enclosing the area around the majority of the penetrations. The only penetrations which do not pass through this area are:

a. Two main steam lines.

b. Permanent equipment hatch which contains a double gasketed closure.

c. Normal personnel access lock.

d. Emergency personnel access lock.

e. Embedded lines (normal sump drain, emergency sump lines).

f. Refueling tube.

The main steam lines are not considered a source of significant leakage because they are welded to the liner plate. The access openings can be tested during normal operation and are not considered sources of significant leakage. There are double seals at each of these access openings, and the space between these double seals is connected to the penetration room. The refueling tube is equipped with a blind flange which is only opened during shutdown for transfer of fuel to the spent fuel pool.

The main function of the system is to control and minimize the release of radioactive materials from the reactor building to the environment in post-
5. accident conditions. When the system is in operation, a negative pressure will be maintained in the penetration room to assure inleakage.

Leakage into each of the penetration rooms is discharged to the unit vent through a pair of filter assemblies each consisting of a prefilter, an absolute filter and a charcoal filter in series. The entire system is designed to operate under negative pressure up to the fan discharge.

5. The design flow rate from the penetration room far exceeds the maximum antici-
18 pated reactor building leakage. The design leak rate of 1/4% per day amounts to approximately 15 scfm compared to a design evacuation rate of 1000 scfm for each half of the system. The three valves in each purge line penetration

will be closed by reactor building isolation signal. The reactor building purge equipment, if running, will be shut down from an interlock on the reactor building purge isolation valves. After closing of the external valves, a small normally open valve vents the leakage, if any, from the two outermost valves into the penetration room.

Following a loss-of-coolant accident, a reactor building isolation signal will
place the system in operation by starting both full-size fans. Two power-
operated butterfly valves which open when the fans start are provided at the
discharge of each fan. This valve will be closed to prevent recirculation if
one fan fails. A check valve is also provided at the discharge of each fan to
prevent recirculation on failure of a fan. In the event of a fan failure, the
8. normally closed tie valve (PR-20) can be opened from its remote manual station
to maintain adequate cooling air through the idle filter train.

The penetration room ventilation system is not vulnerable to control malfunctions since it is controlled manually. Instrumentation is used only to monitor system performance and has no control function other than to guide the operator in adjusting the final control elements.

16. The system utilizes remote manual control valves PR-13 and PR-17 in conjunction
with constant speed fans to provide the proper negative pressure in the pene-
tration room. Locations of penetrations and openings in the penetration room
are shown on Figures 6-5B and 6-5C. The system is designed so that each filter
20. 5. train will maintain a maximum negative pressure of 1.73" H_2O, assured by the
16. two redundant 8" vacuum relief valves. If during operation the leakage increases
causing a decrease in negative pressure below 0.06" H_2O, the remote manual
control valve will be adjusted or leaks will be repaired to bring the negative
pressure to greater than .06" H_2O.

The remote manual control valve is also used to compensate for filter loading.
16. Initially, it will be partially closed and as the filter loads up causing a
decrease in flow and negative penetration room pressure, the valve will gradually
be opened so that the pressure drop across the filter-valve combination remains
16. constant. By periodically adjusting the remote manual control valve to offset
the effect of increased leakage and filter loading the system characteristic
remains constant.

The communicative paths between various parts of the penetration room are very large in comparison with the minute leakage that might exist due to imperfect seals. It therefore can be assumed that no pressure differentials exist in the room so that an instrument string sensing pressure at a single point can be used. Penetration room pressure is displayed in the control room and excessive and insufficient vacuum are annunciated.

Fan status and radiation level of filter effluent are displayed in the control
room and excessive radiation is annunciated. Filter ΔP is displayed locally.
16. Filter flow is displayed remotely adjacent to the remote manual control valves
PR-13 and PR-17 remote control stations.

The system may be actuated by an operator during normal operation for testing.

Particulate filtration is achieved by a medium efficiency pre-filter and a high efficiency (HEPA) filter.

The pre-filter consists of multiple horizontal tubular bags attached to a vertical metal plate header. The bags are made of ultra fine glass fibers and are supported so that adjacent bags do not touch and reduce the flow area. At the filter train design flow of 1000 cfm the pre-filter is operating at one-half its rated flow.

The HEPA filter will intercept any particulates that pass through the pre-filter. The filter consists of a single cell of fiber glass media mounted in a metal frame. The cell has face dimensions of 24" x 24" and a depth of 11-1/2" and is rated at 1150 scfm.

5. Absorption filtration is accomplished by an activated charcoal filter. The filter consists of three horizontal removable type double tray carbon cells. Flow through the trays is essentially vertical. Each tray has a face area of 4.2 sq. ft. and a bed depth of 2 inches. At rated flow (167 cfm) the average fact velocity is 40 ft./min. and the residence time is 0.25 seconds. Each tray contains 40 lbs. of carbon. The carbon is impregnated so that it will absorb methyl iodide as well as elemental iodine. It is derived from new, hard coconut shells.

The design basis for this filter was a requirement to remove 25 percent of the core iodine inventory. The 25 percent was derived using the standard assumption that during an MHA 50 percent of the halogens are released from the core and that 50 percent of the iodine released plates out within the reactor building. The initial inventory of the individual isotopes in terms of Curies/MWt is given in Table 1 below:

Table 1

Inventory of Iodine Isotopes in Reactor Building (at t = o)

Isotope	Initial Inventory Curies/MWt
Iodine 131	2.51×10^4
8. Iodine 132	3.81×10^4
Iodine 133	5.63×10^4
Iodine 134	6.58×10^4
Iodine 135	5.10×10^4

From time to time the system will be activated to purge the filters of any
5. moisture that may accumulate. The air will be taken from the penetration room where it will be sufficiently warm to accomplish this purpose.

6.4.2.2 Codes and Standards

The equipment in this system is designed and rated in accordance with the following standards:

6.4.2.2 Codes and Standards

The equipment in this system is designed and rated in accordance with the following standards:

Pre-Filter - Filter efficiency is determined by the "American Filter Institute Dust Spot Test" utilizing atmospheric dust.

Absolute Filter - The basic design criteria for this filter is set forth in AEC Health and Safety Bulletin 212 (6-25-65) which incorporates U.S. Military Specification MIL-F-51068A captioned "Filter, Particulate, High Efficiency, Fire Resistant".

In addition, the dust holding capacity is determined by utilizing the test procedures of AFI "Code for Testing Air Cleaning Devices Used in General Ventilation", Section I (1952).

Absorptive (Carbon) Filter - The specified ignition temperature of the carbon is checked using the procedure set forth in USAEC Report DP-1075, "High Temperature Absorbents of Iodine", by R.C. Milhaus. This test is conducted on one sample from each lot of carbon.

Fans - Fan performance is determined by prototype test according to procedures set forth by the Air Moving and Conditioning Association (AMCA) 1960 Standard Test Code.

6.4.2.3 Material Compatibility

Since this system will not experience high temperature or corrosive fluid service, it utilizes carbon steel and suitable coatings to obtain desired service life.

6.4.2.4 Equipment Accessibility

The system equipment is fully accessible during all normal station operation for maintenance and performance testing.

6.4.2.5 Reliability Considerations

Each unit's penetration room is provided with two fans and two filter assemblies.
8. 5. Both fans, discharging through a single line to the unit vent, are controlled from the main control room.

During normal operation, this system is held on standby with each fan aligned
5. with a filter assembly. The engineered safeguards signal from the reactor building will actuate the fans. Control room instrumentation monitors
5. operation. The system can be tested during normal operation.

6.4.3 DESIGN EVALUATION

A single failure analysis of the various portions of this system is presented in Table 6-8.

Table 6-8
Single Failure Analysis for Reactor Building Penetration Room Ventilation System

Component	Malfunction	Comments
1. Fan	Fails	The other fan retains full capacity.
5. 2. Fan Discharge Valve	Fails to open	The other fan retains full capacity.
3. Fan Discharge Valve	Fails to close	Check valve prevents recirculation.
4. Vacuum Relief Valve	Failure to open.	The other vacuum relief valve opens.
8. 16. 5. PR-13 and PR-17	Loss of air to remote manual loaders.	Valves go open.
6. PR-20	Loss of air to remote manual loaders.	Valve stays shut.

Redundant fans, cross connected piping, and locked open filter inlet valves render incredible a loss of cooling air flow to the filters. However, for the postulated case of loss of air flow through a filter, the heatup time until charcoal ignition temperature is reached was determined using the following conservative assumptions:

(a) MHA iodine release to Reactor Building
(b) Iodine input to Penetration Room filter based on Reactor Building leak rate of one-fourth percent per day for the first day and one-eighth percent per day thereafter. Iodine evenly distributed over a single filter (120 lbs. charcoal)
(c) No heat loss from filter
(d) Peak heating rate of 2630 btu per hour
(e) Specific heat for charcoal equals 0.2 btu per pound F
(f) Initial charcoal temperature equals 104F; charcoal ignition temperature equals 660F.

10.

An analysis based on the following relationship shows that there is at least five hours between loss of air flow and the time at which charcoal ignition temperature is reached. This is ample time to start the stand-by fan and restore air flow.

$Q = W\ c\ \Delta T$

Where: Q = Btu required to increase charcoal temperature by ΔT
W = 120 lbs. charcoal
c = 0.2 Btu per pound F
ΔT = 660 F - 104 F = 556 F

Rev. 5. 5/25/70
Rev. 8. 7/23/70
Rev. 10. 8/28/70
Rev. 16. 7/30/71

Adequate instrumentation is provided to detect loss of air flow through either filter. Reduction in air flow below a preset minimum would result in low Penetration Room vacuum and cause an alarm in the control room. Flow indication
10. with readout outside the penetration filter area is furnished for filter. The operator will monitor instrumentation periodically following an accident, as stated in answer to Question 5, FSAR Supplement 6.

5. The reactor building penetration room is maintained at a negative pressure of greater than 0.06" H_2O when the penetration room fans are in operation.

4. Even in the event of unfiltered leakage of all the iodine input to the
20. penetration room due to high wind velocity, the improvement in atmospheric dilution more than compensates for bypassing of the penetration room filter by this portion of the iodine. At a wind velocity of greater than 8.1 mph, the improvement in X/Q compensates for the complete loss of the filtering system in the calculation of offsite dose. A wind velocity of 8.1 mph will cause a reduction in pressure of .032" H_2O along the penetration room wall. (This assumes that the wind velocity is exactly parallel to the wall which is the worst case assumption). By maintaining the penetration room at a negative pressure of 0.06" H_2O, a conservative margin of pressure is established.

6.4.4 TESTING

The high efficiency particulate air (HEPA) filters and the charcoal iodine filters are tested to assure that they are able to remove airborne materials from penetration leakage.

The following is a description of the efficiency tests. The field tests will be made prior to initial unit operation, and annually thereafter and any time a filter is replaced.

Before shipment each absolute filter will be given a leak test. The filters will be challenged with DOP (DI-OCTYL Phthalate) smoke particles having a mean diameter of 0.3 microns present in a concentration of 80 micrograms per liter at design flow. If the efficiency defined as the change in concentration divided by the upstream concentration is less than 99.97% the filter will be rejected. The test will last about 30 seconds.

5. After the filter is installed, the above test will be repeated. The test will last about 2 minutes. Permanently installed injection and sampling probes will be provided. Provisions have been made to insure adequate mixing of the injection and penetrating DOP so that fixed probes can be used. Upstream mixing is done by an injection probe in the form of a perforated ring and a diffuser plate. Downstream mixing is achieved by a long flow path with a pronounced converging section ahead of the probe.

The DOP leak test will not have any effect on subsequent filter performance because of the low concentration and short test period.

The filter manufacturer will perform radioiodine absorption efficiency tests on samples from each lot of carbon. The samples will have the same physical characteristics as the actual filters.

One sample will be challenged with elemental iodine (I_2-131) in a background of nonradioactive elemental iodine and methyl iodine at system design conditions. After a loading period of 2 hours, the feeding of I_2 and CH_3I will be halted and elution air will be drawn through the sample for 4 hours. If the integrated I_2-131 removal efficiency including both feed and elution periods is less than 95.0%, the lot will be rejected.

5. A second sample will be challenged with radioactive methyl iodide (CH_3I-131) in a background of nonradioactive elemental iodine and methyl iodine at system design conditions. After a loading period of 2 hours, the feeding of I_2 and CH_3I will be halted and elution air will be drawn through the sample for 4 hours. If the integrated CH_3I-131 removal efficiency including both feed and elution periods is less than 70.0%, the lot will be rejected.

Prior to shipment each carbon filter cell will be subjected to a leak test. The test will consist of challenging the cell with 50 ppm of refrigerant R-112 for 5 minutes at system design conditions. If the cell efficiency defined as the change in concentration divided by the initial concentration is less than 99.8% (maximum let through concentration is 0.1 ppm) the cell will be rejected.

The above test will be repeated after the filter is installed at the site. The permanently installed test probes for the particulate filter will be used for testing the carbon filter.

The R-112 leak test will not have any effect on subsequent filter performance because of the low concentration and high volatility of the R-112 and the short loading period.

6.5 ENGINEERED SAFEGUARDS LEAKAGE AND RADIATION CONSIDERATIONS

6.5.1 INTRODUCTION

The use of normally operating equipment for engineered safeguards functions and the location of some of this equipment outside the reactor building require that consideration be given to direct radiation levels after fission products have accumulated in these systems and leakage from these systems.

The shielding for components of the engineered safeguards is designed to meet the following objectives in the event of a maximum hypothetical accident:

a. To provide protection for personnel to perform all operations necessary for mitigation of the accident.

b. To provide sufficient accessibility in all areas around the station to permit safe continued operation of the unaffected nuclear units.

6.5.2 SUMMARY OF POST-ACCIDENT RECIRCULATION

Following a loss-of-coolant accident, flow is initiated in the low pressure injection system from the borated water storage tank to the reactor vessel. Flow is also initiated by the reactor building spray systems to building spray headers. When the borated water storage tank level reaches the low level alarm, recirculation from the reactor building emergency sump is initiated by the operator for both the reactor core cooling flow and the reactor building sprays. The operator will maintain the 3,000 gpm design flow rates of the low pressure injection pumps, but will throttle the reactor building spray pumps from the 1,500 gpm design flow rate of 1,000 gpm in order to ensure adequate NPSH. The post-accident recirculation system includes all piping and equipment both internal and external to the reactor building as shown on Figures 6-2 and 6-3 up to the stop and test line valves leading to the borated water storage tank.

The NPSH available to the low pressure injection and reactor building spray pumps during the post-LOCA recirculation phase has been calculated based on:

a. "As Built" piping drawings.
b. Pipe and fitting losses calculated using the information in Crane Technical Paper No. 410.
c. Total flow in a single string (i.e., consisting of one low pressure injection pump and one reactor building spray pump served by a single sump suction line) is 4,000 gpm. This consists of 3,000 gpm to the low pressure injection pump and 1,000 gpm to the reactor building spray pump.
d. Sump water temperatures and reactor building pressures given in Table 6-8a.
e. RB spray pump shaft center line at elevation 760 ft. 1 in.
f. LP injection pump shaft center line at elevation 761 ft. 1 in.
g. Water level in the reactor building sump is 783 ft. 9 in. based on the following assumptions: (height above RB basement level is 6.50 ft.)

    (1) The Technical Specification minimum levels were used for the BWST and the CFT's, with three feet of level remaining in the BWST at time of switchover.

    (2) Some water is maintained in the RB atmosphere as vapor. The quantity was determined using the results of a CONTEMPT Computer Run for a 5.0 ft^2 break with 2 fan coolers and one reactor building spray pump operating.

    (3) The break is conservatively assumed to occur at the top of the hot leg, thereby keeping the RC System full.

The NPSH available to each pump, based upon the above data, are given in Table 6-8a. The calculated NPSH available to each pump in the "B" string, LP-P1B and BS-P1B (worst case), assuming a saturated sump, are compared below with the NPSH required.

Rev. 19 5/5/72
(Entire Page Revised)

Pump	LP Injection	RB Spray
Flow Rate, gpm	3,000	1,000
NPSH, ft. H_2O		
Available	18.0	19.8
Required	12.0	17.0

The required NPSH's indicated above reflect the manufacturer's certified test.

6.5.3 BASES OF LEAKAGE ESTIMATE

While the reactor auxiliary systems involved in the recirculation complex are closed to the auxiliary building atmosphere, leakage is possible through component flanges, seals, instrumentation, and valves.

The leakage sources considered are:

a. Valves

(1) Disc leakage when valve is on recirculation system boundary.

(2) Stem leakage.

(3) Bonnet flange leakage.

b. Flanges

c. Pump shaft seals

While leakage rates have been assumed for these sources, maintenance and periodic testing of these systems will preclude all but a small percentage of the assumed amounts. With the exception of the boundary valve discs, all of the potential leakage paths may be examined during periodic tests or normal operation. The boundary valve disc leakage is retained in the other closed systems, and therefore, will not be released to the auxiliary building.

While valve stem leakage has been assumed for all valves, the manual valves in the recirculation complex are backseating and do not rely on packing alone to prevent stem leakage.

6.5.4 LEAKAGE ASSUMPTIONS

Source	Quantities
a. Valves - Process	
(1) Disc leakage	10 cc/h/in. of nominal disc diameter
(2) Stem leakage	1 drop/min.
(3) Bonnet flange	10 drops/min.

Table 6-8a

NPSH Available to ES Pumps During Recirculation
(1-3000 gpm LPI pump, and 1-1000 gpm BS pump)

Period	Time, (a) seconds	Reactor Building Total Pressure, (a) psig	Sump Temp, (a) F	LPI Pump NPSH (b) Available, ft	BS Pump NPSH (b) Available, ft	LPI Pump NPSH Available Saturated Sump, (c) ft	BS Pump NPSH Available Saturated Sump, (c) ft
Initial (Before Accident)	0	0	--	--	--	--	--
Beginning of Recirculation	4,550	4.41	213.6	27.5	29.3	18.0	19.8
Maximum Recirculation Sump Temperature	8,620	5.26	227.7	18.1	19.9	18.0	19.8

(a) Data obtained from the results of a reactor building pressure analysis for 2 Fan Coolers and 1 Reactor building spray pump shown in Figures 14-63i and 14-63j (see Section 14).

(b) Credit taken for Reactor Building Pressure

(c) Reactor Building total pressure assumed to be equal to the saturation pressure of the sump water (boiling sump assumption).

b. Valves - Instrumentation

Bonnet flange and stem — 1 drop/min.

c. Flanges — 10 drops/min.

d. Pump seals — 50 drops/min.

For the analysis, it was assumed that the water leaving the reactor building was at 252 F. This assumption is conservative as this peak temperature would only exist for a short period during the postaccident condition. Water downstream of the coolers was assumed to be 115 F. The auxiliary building was assumed to be at 70 F and 30 per cent relative humidity. Under these conditions, approximately 22 per cent of the leakage upstream of the coolers and 4 per cent of the leakage downstream of the coolers would flash into vapor. For the analysis, however, it was assumed that 50 per cent of the leakage upstream of the coolers would become vapor because of additional heat transfer from the hot metal.

6.5.5 DESIGN BASIS LEAKAGE

The design basis leakage quantities are tabulated in Table 6-9.

Table 6-9

16. Leakage Quantities to Auxiliary Building

(Reference Supplement 9 Revisions for Oconee 3)

Leakage Source	No. of Sources	Estimated Quantities: Leakage Per Source (drops/min.)	Estimated Quantities: Total Leakage (cc/h)
Low Pressure Injection System			
a. Pump Seals			
Low Pressure Injection pump	2	50	300
Spray Pump	2	50	300
b. Flanges(*)	10	10	300
c. Process Valves	28	1	84
d. Instrumentation Valves	40	1	120
9. e. Valve Seats at Boundaries	31	(**)	1070
16.		Total	2174

(*) Only the pumps are flanged.

(**) Assuming 10 cc/h/in. of nominal disc diameter.

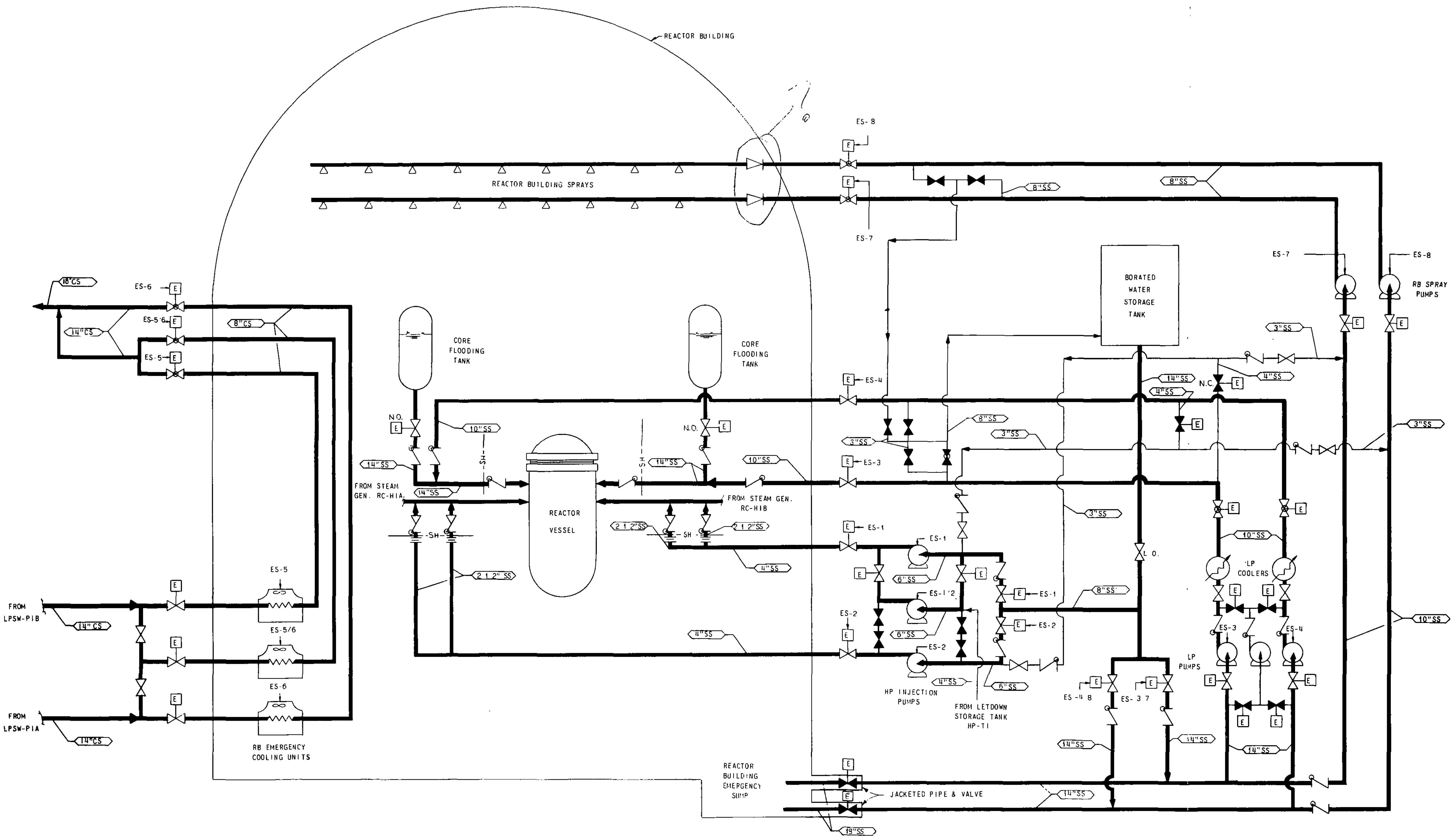
6.5.6 LEAKAGE ANALYSIS CONCLUSIONS

It was concluded from this analysis (in conjunction with the discussion and analysis in 14.2.2.4.4) that leakage from engineered safeguards systems outside the reactor building does not pose a public safety problem.

6.6 REFERENCES

(1) Durametallic Corporation Research Report No. 1200, "Seal Test for Nuclear Power Plant Application," 4/24/68.

(2) Cottrell, W. B. and Savolainen, A. W., Editors, U.S. Reactor Containment Technology, ORNL-NSIC-5, Volume II.



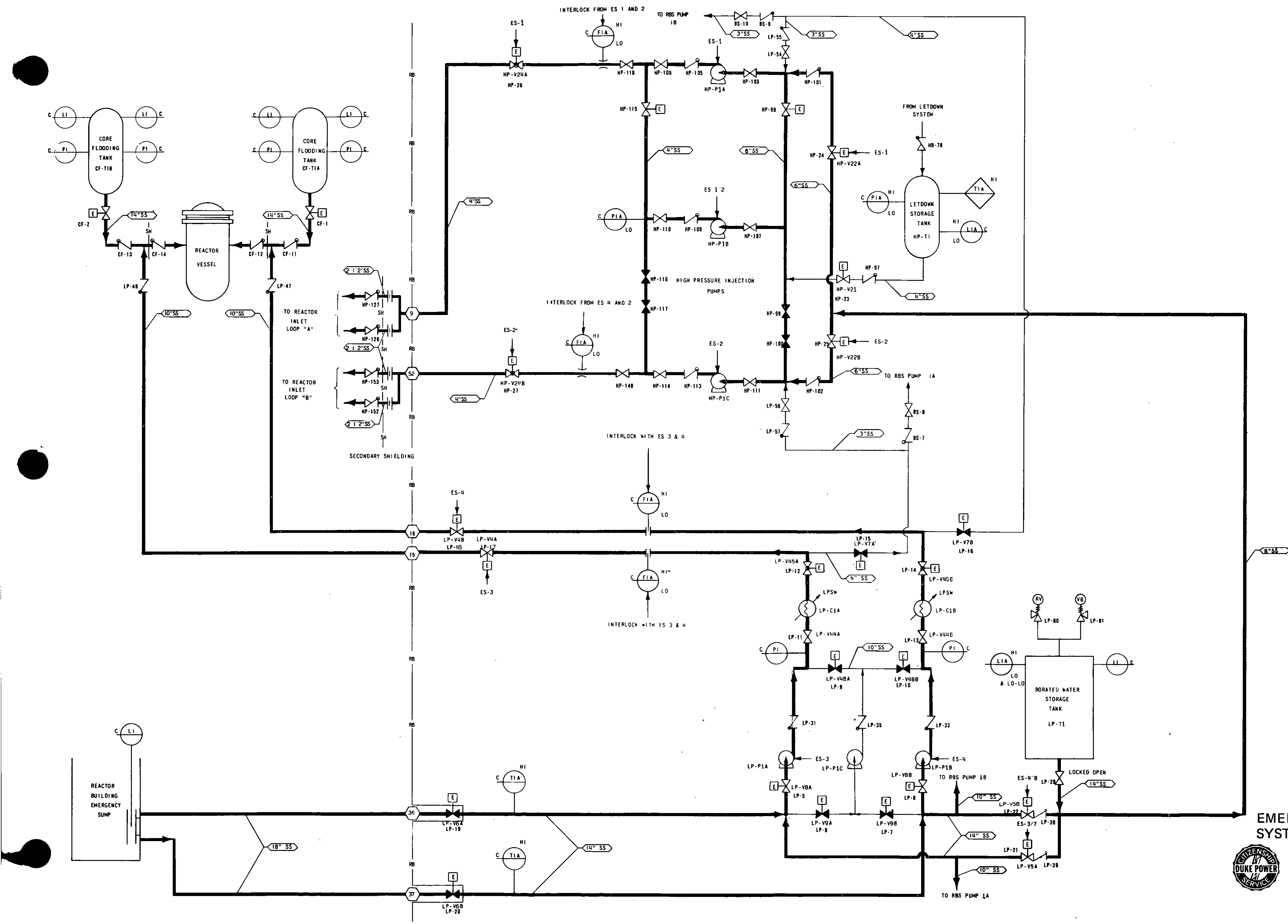
NOTES

1. UNITS 2 & 3 ARRANGEMENT DUPLICATE OF UNIT 1 SHOWN ABOVE
2. FOR LEGEND NOMENCLATURE SEE FIGURE 9-1.
3. VALVES ARE SHOWN IN ES POSITION

SIMPLIFIED SCHEMATIC DIAGRAM OF ENGINEERED SAFEGUARDS SYSTEM FOR CORE AND BUILDING PROTECTION

OCONEE NUCLEAR STATION

Figure 6 - 1



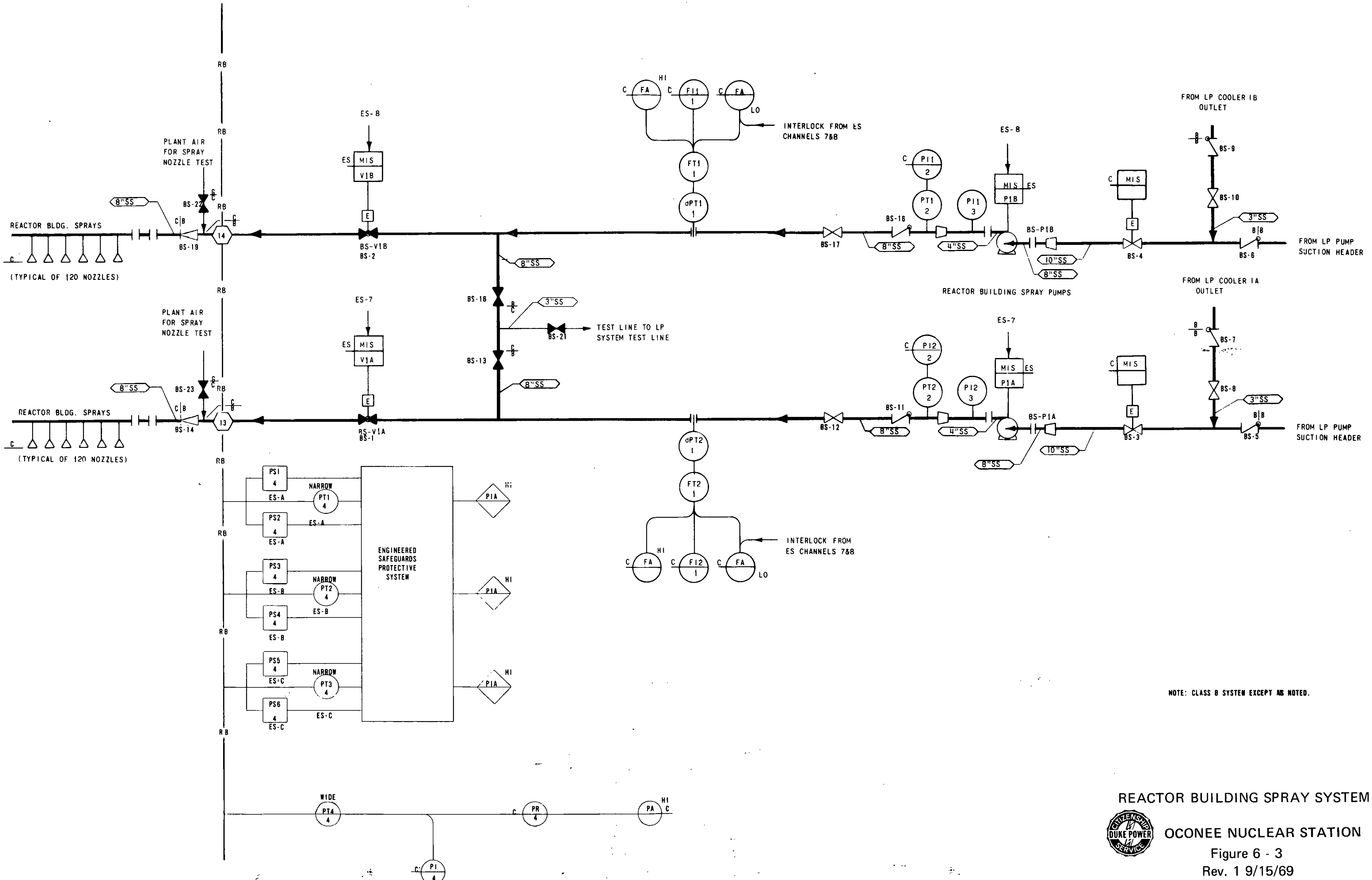
EMERGENCY CORE COOLING SYSTEM

OCONEE NUCLEAR STATION

Figure 6 - 2

Rev. 16 7/30/71

Rev. 26 1/29/73



REACTOR BUILDING SPRAY SYSTEM

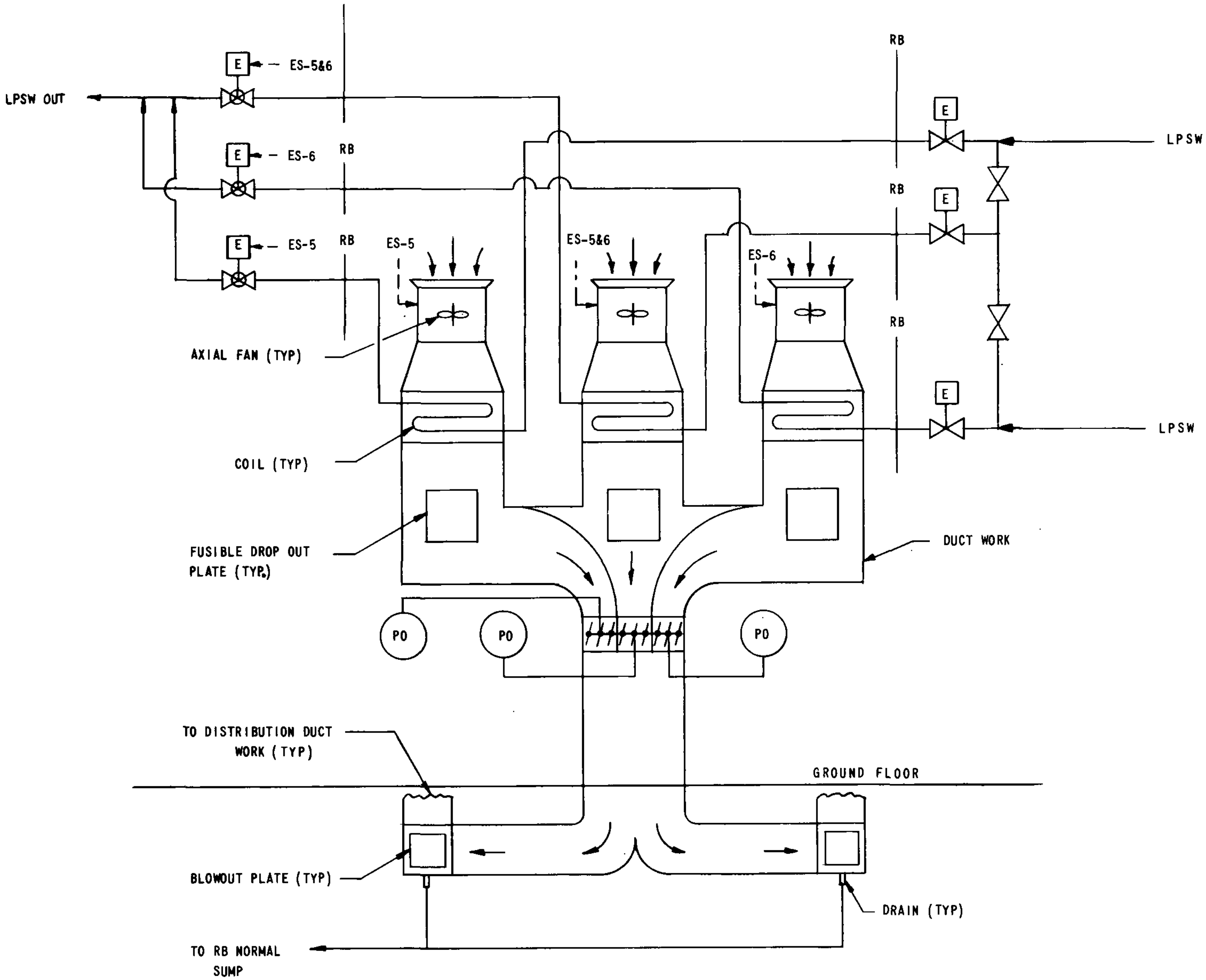
OCONEE NUCLEAR STATION

Figure 6 - 3

Rev. 1 9/15/69

Rev. 5 5/25/70

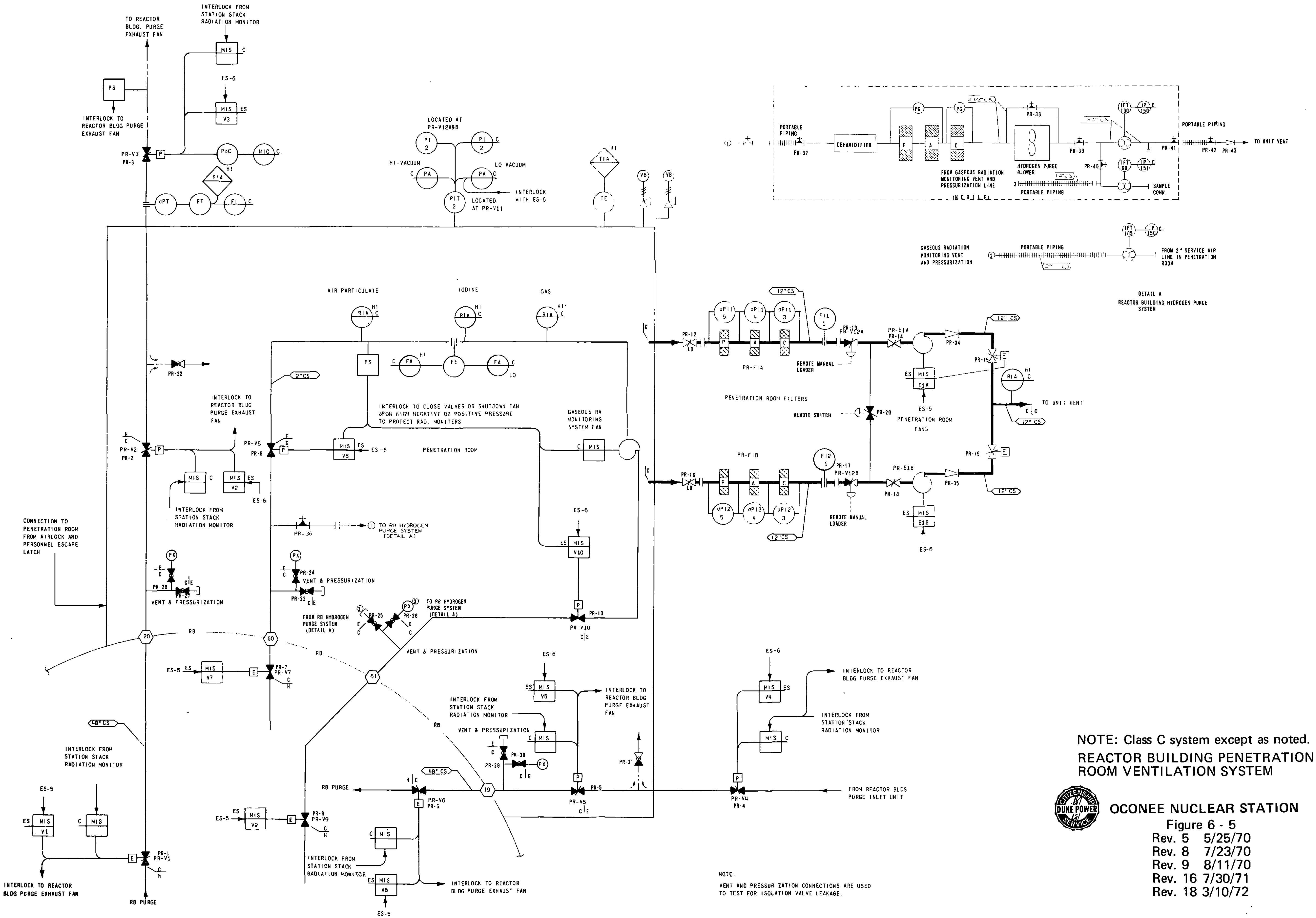
Rev. 16 7/30/71



REACTOR BUILDING COOLING SCHEMATIC

OCONEE NUCLEAR STATION

Figure 6 - 4



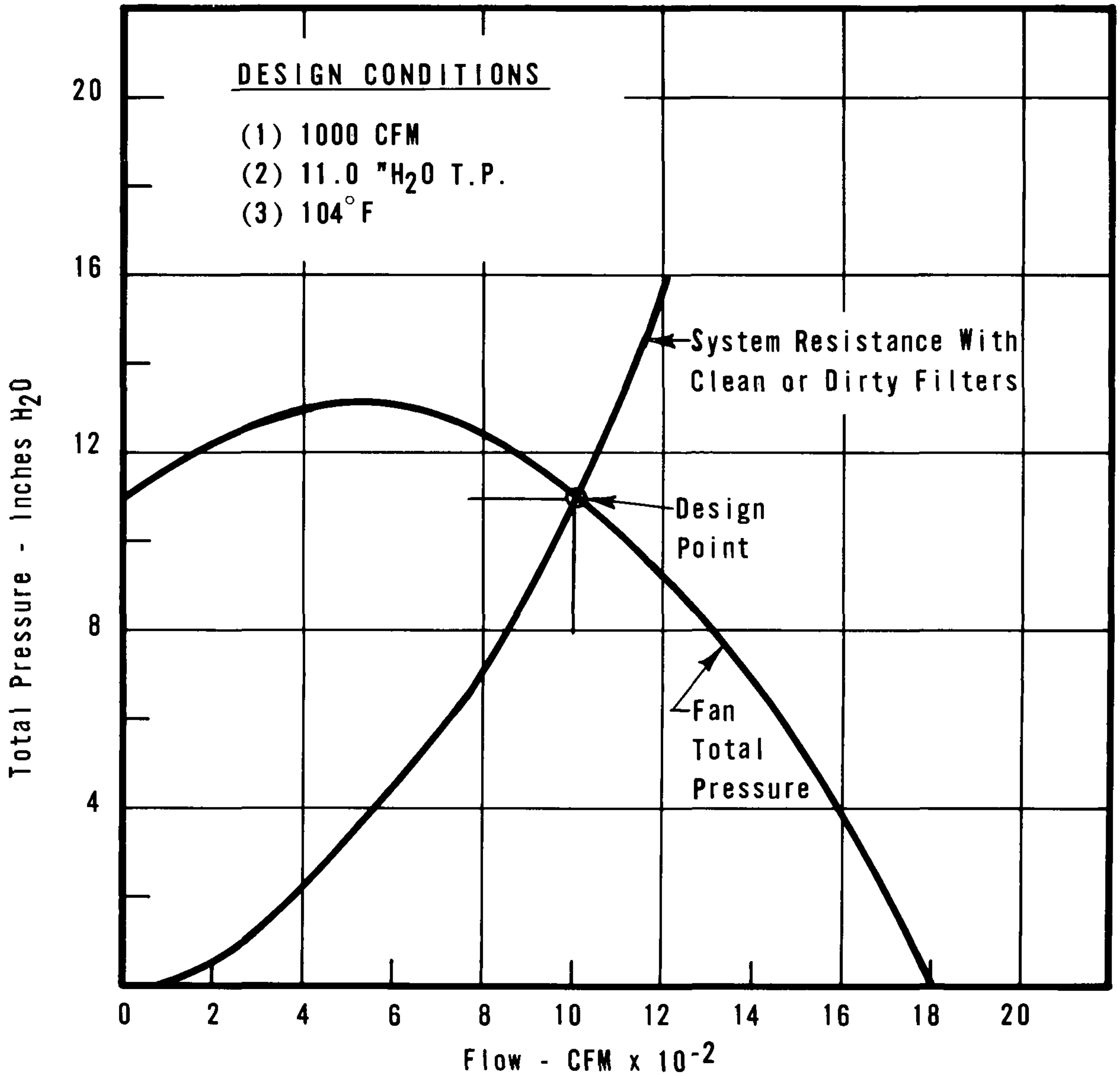
NOTE: Class C system except as noted.

REACTOR BUILDING PENETRATION ROOM VENTILATION SYSTEM

OCONEE NUCLEAR STATION

Figure 6 - 5

Rev. 5 5/25/70
Rev. 8 7/23/70
Rev. 9 8/11/70
Rev. 16 7/30/71
Rev. 18 3/10/72

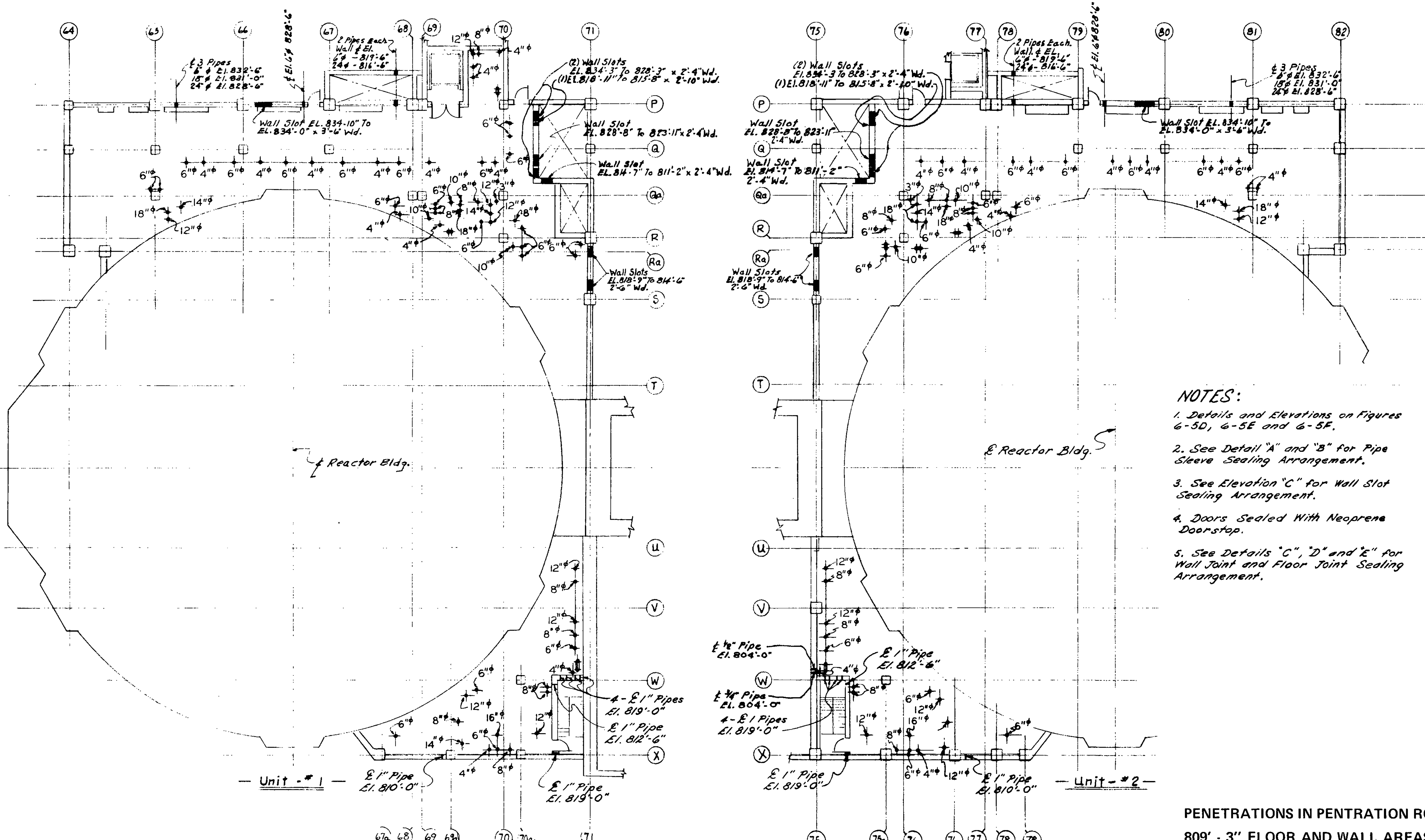


PENETRATION ROOM VENTILATION FAN AND SYSTEM CHARACTERISTICS

OCONEE NUCLEAR STATION

Figure 6 - 5A

(New) Rev. 5 5/25/70



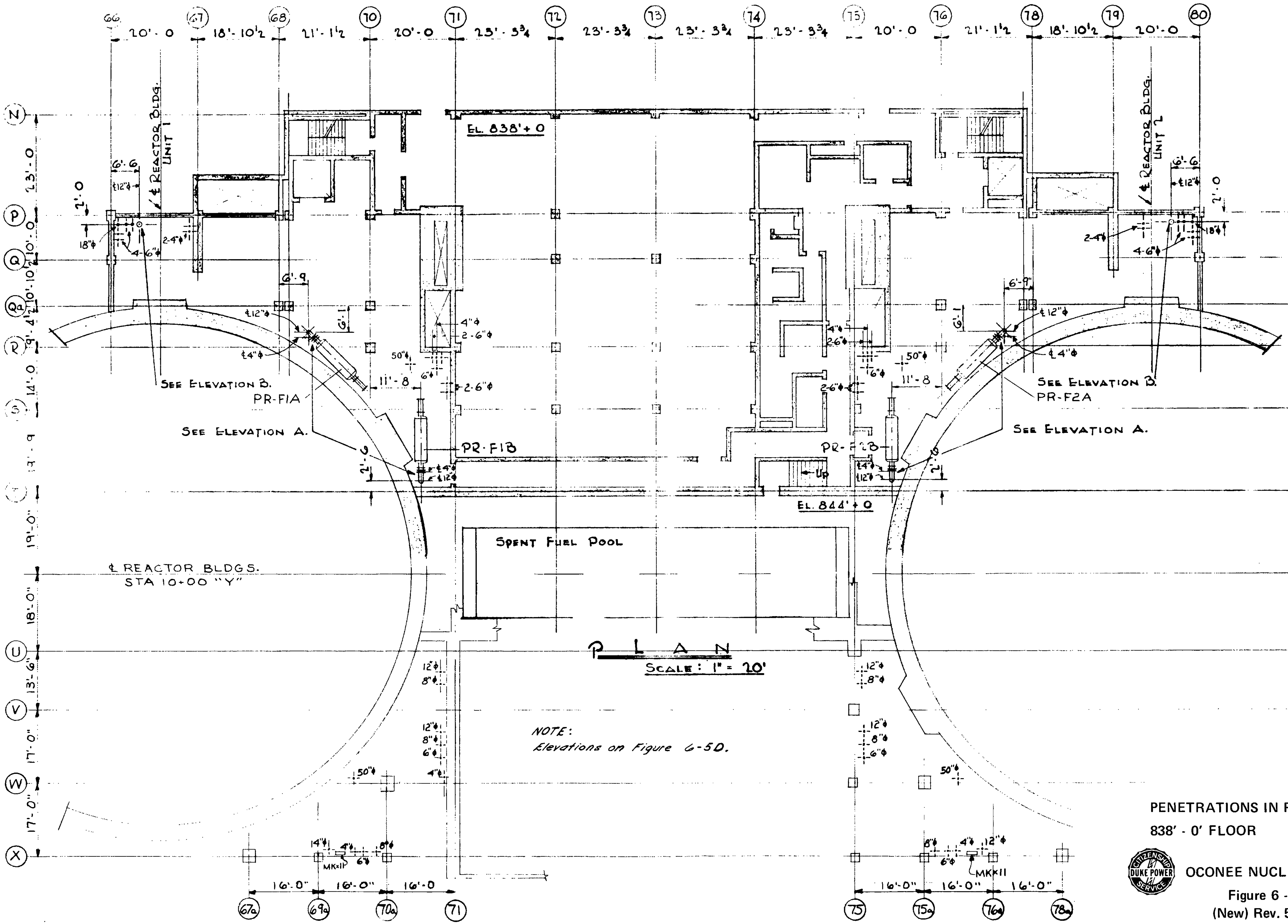
PENETRATIONS IN PENTRATION ROOMS

809' - 3" FLOOR AND WALL AREAS

OCONEE NUCLEAR STATION

Figure 6 - 5B

(New) Rev. 5 5/25/70



PENETRATIONS IN PENTRATION ROOMS

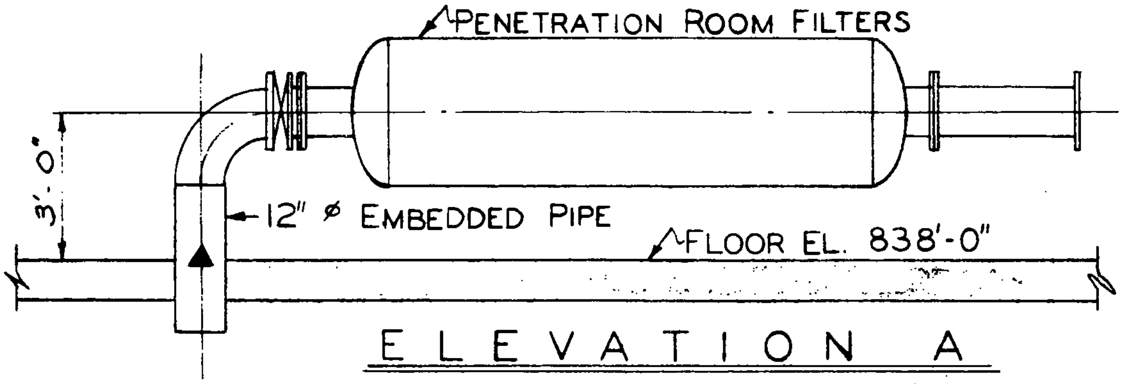
838' - 0' FLOOR

OCONEE NUCLEAR STATION

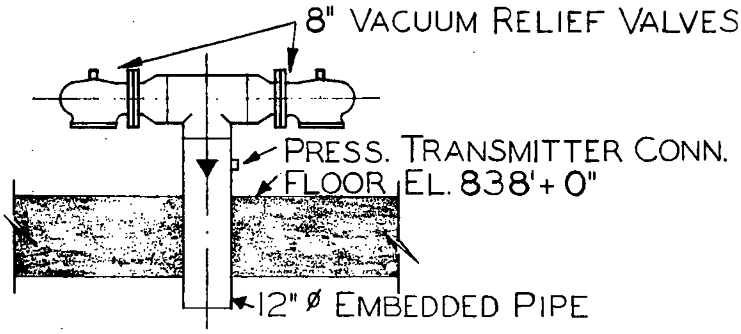
Figure 6 - 5C

(New) Rev. 5 5/25/70

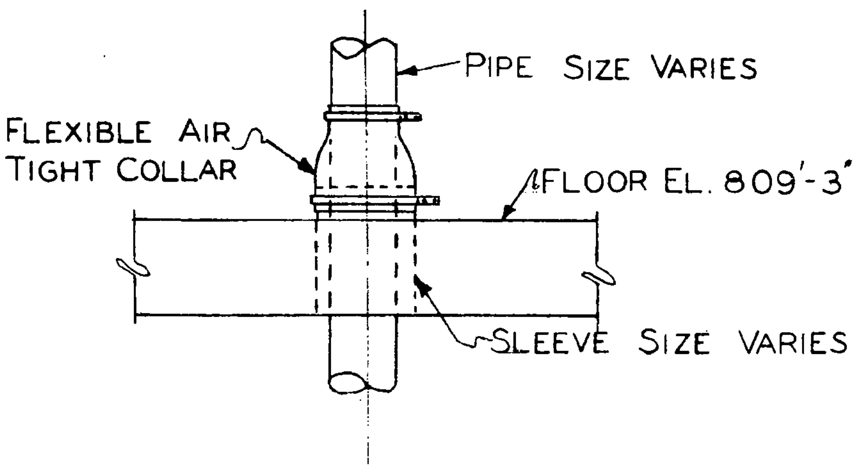
Rev. 16 7/30/71



ELEVATION A



ELEVATION B



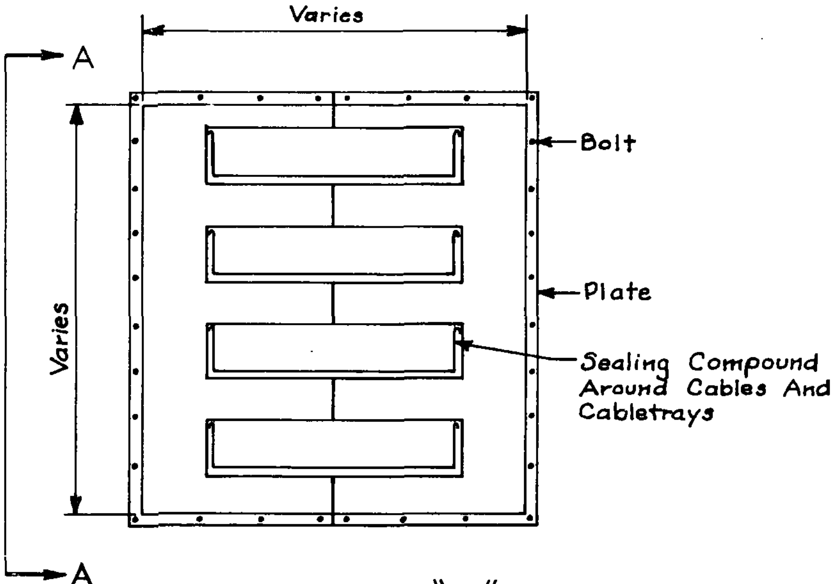
DETAIL A

Rev. 8 Elevation B revised

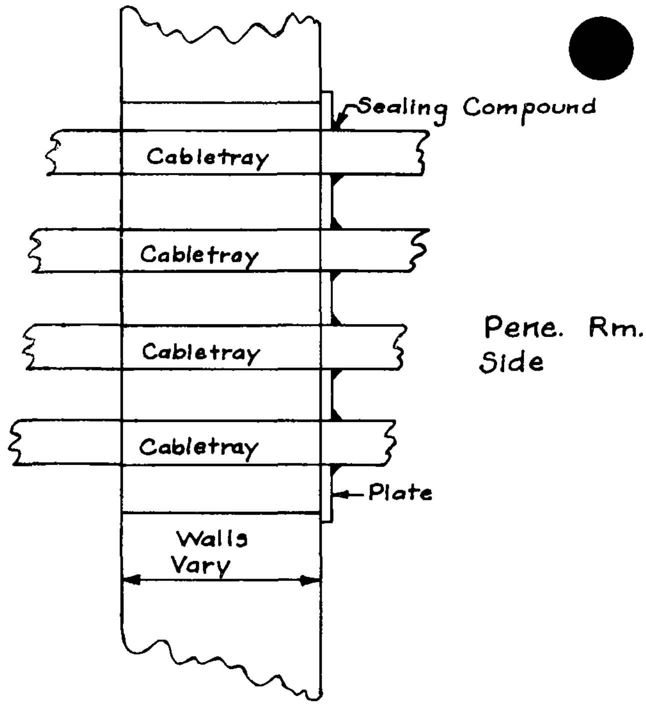
PENETRATION ROOMS DETAILS, MECHANICAL OPENINGS

OCONEE NUCLEAR STATION

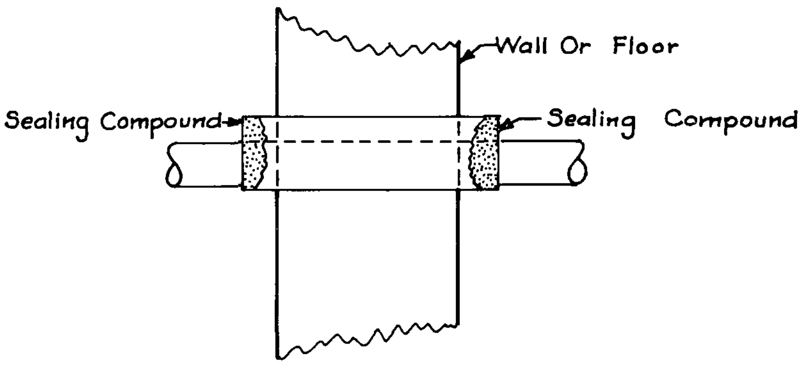
Figure 6 - 5D
(New) Rev. 5 5/25/70
Rev. 8 7/23/70



ELEVATION "C"



SECTION A-A



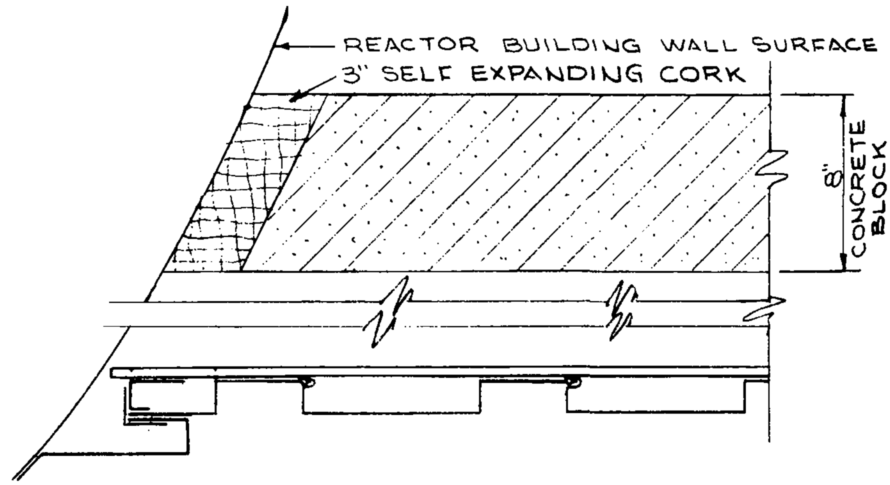
DETAIL "B"

PENETRATION ROOMS DETAILS,
ELECTRICAL OPENINGS

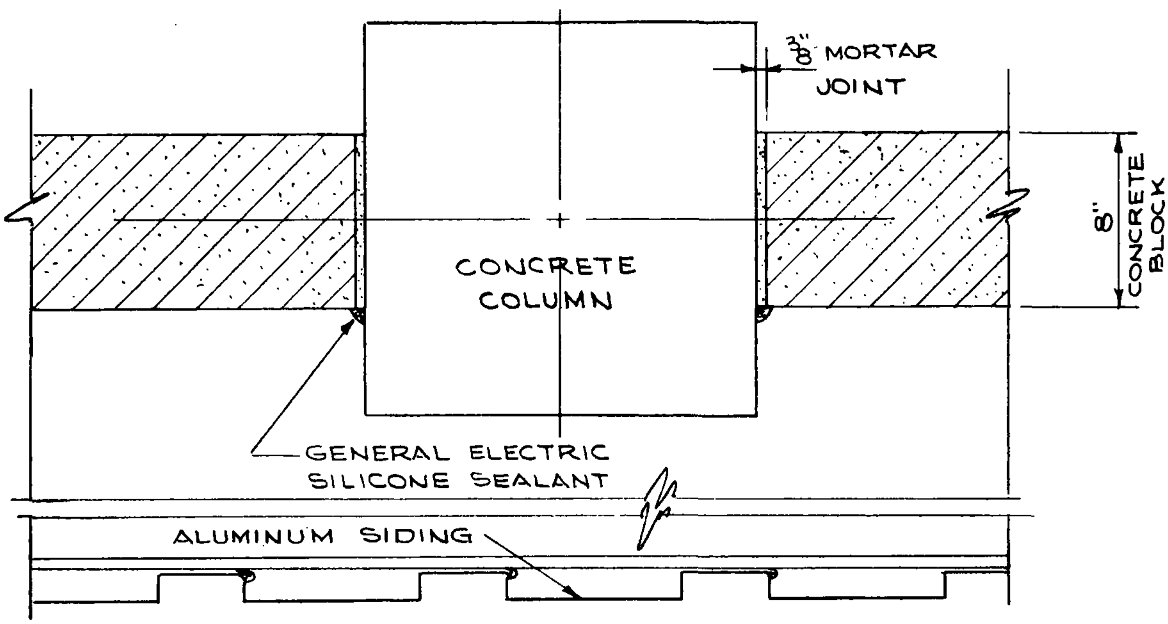
OCONEE NUCLEAR STATION

Figure 6 - 5E

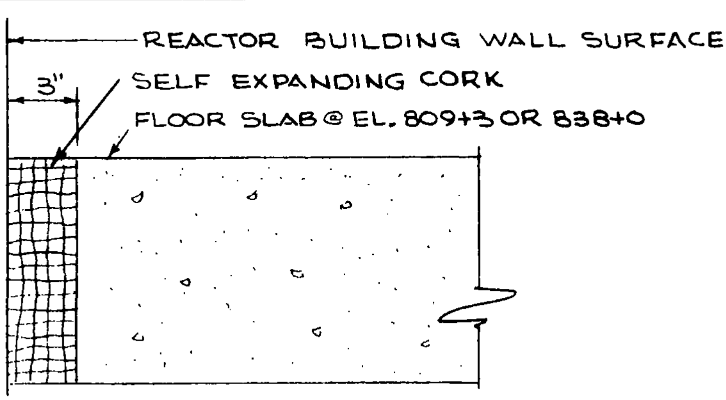
(New) Rev. 5 5/25/70



DETAIL "C" - PLAN OF WALL DETAIL AT REACTOR BUILDING



DETAIL "D"-PLAN OF WALL DETAIL AT COLUMN



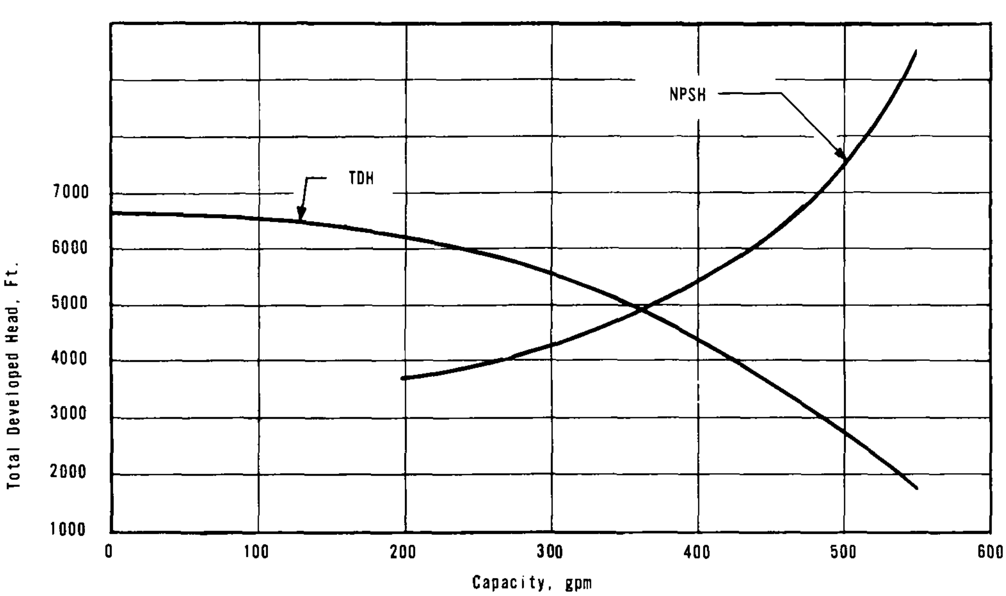
DETAIL "E" - SLAB INTERSECTING WITH REACTOR BUILDING

PENETRATION ROOMS CONSTRUCTION DETAILS

OCONEE NUCLEAR STATION

Figure 6 - 5F

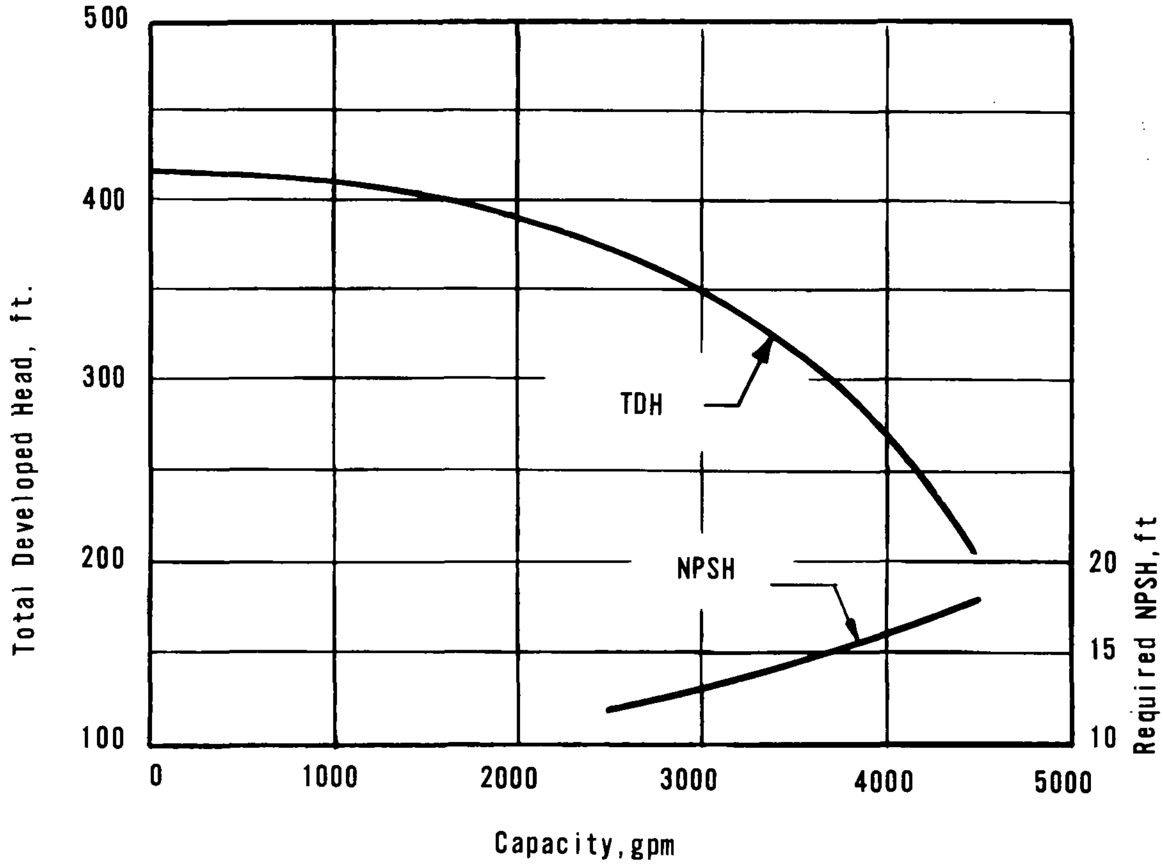
(New) Rev. 5 5/25/70



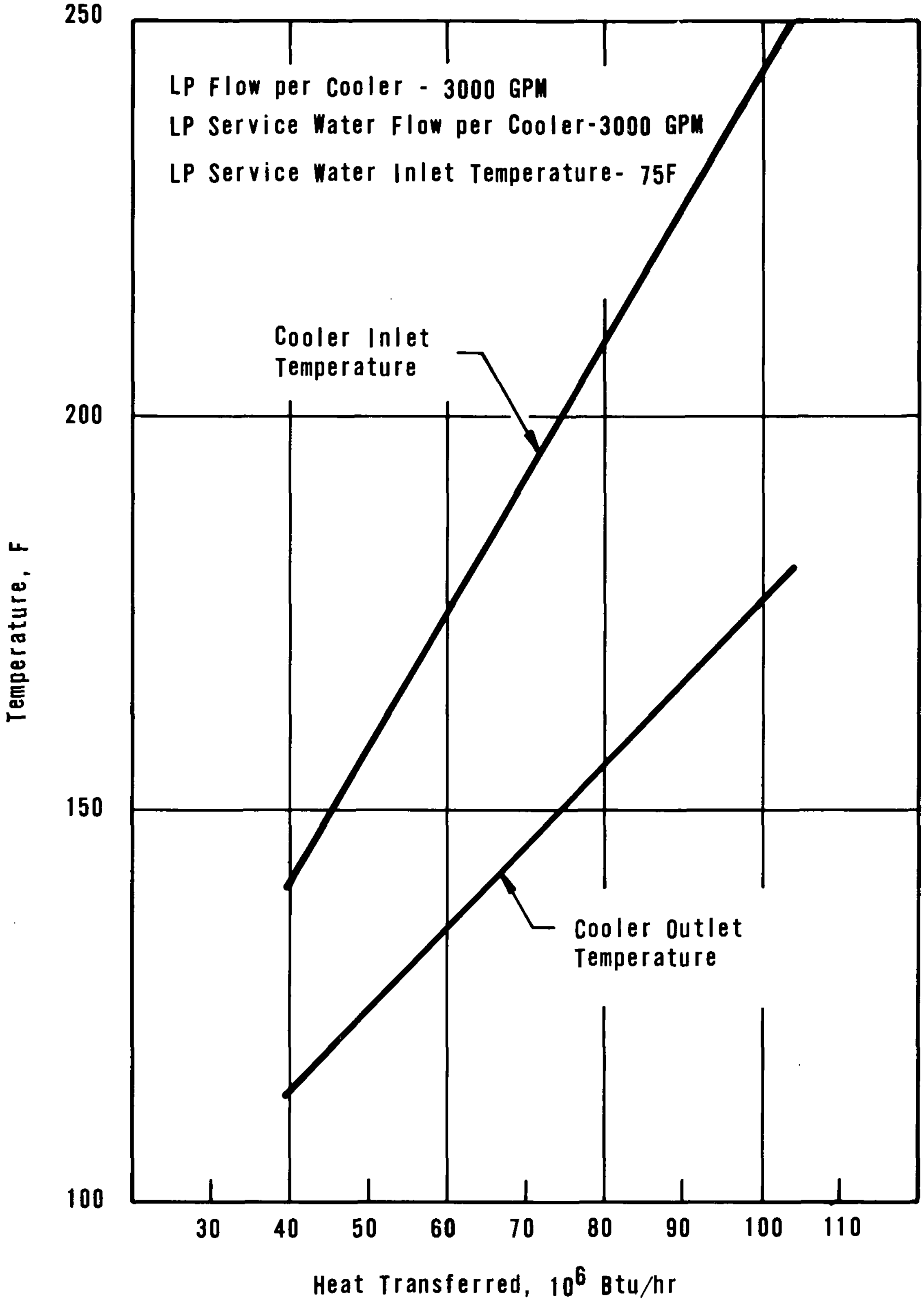
HIGH PRESSURE INJECTION
PUMP CHARACTERISTICS

OCONEE NUCLEAR STATION

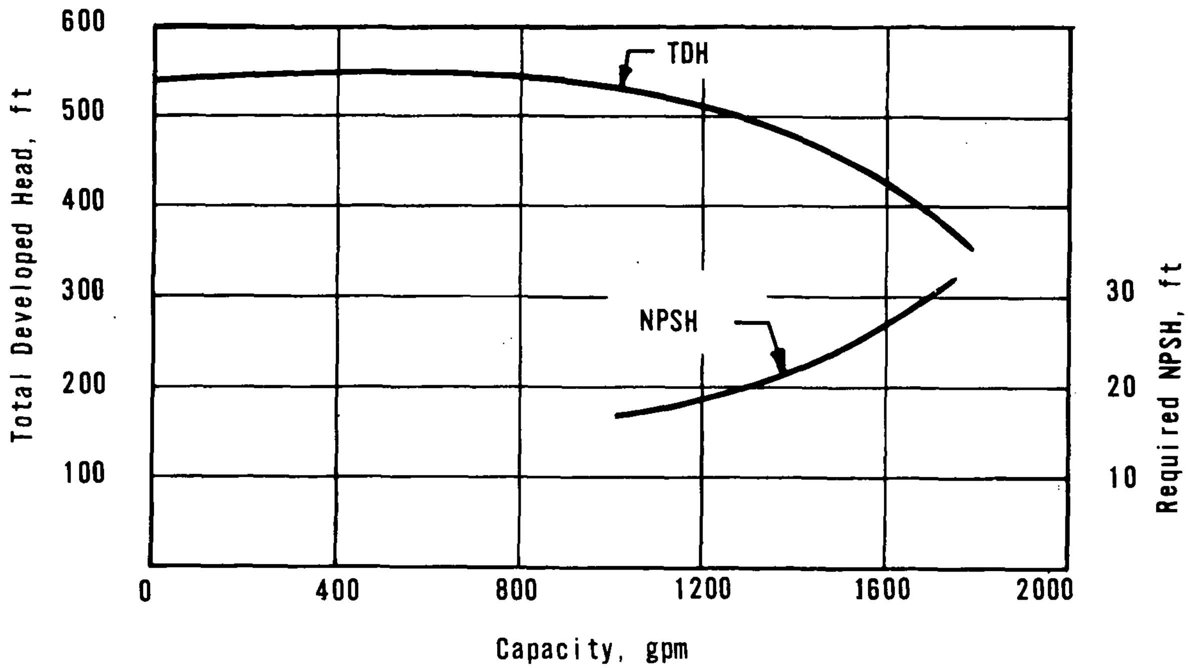
Figure 6 - 6



LOW PRESSURE INJECTION
PUMP CHARACTERISTICS



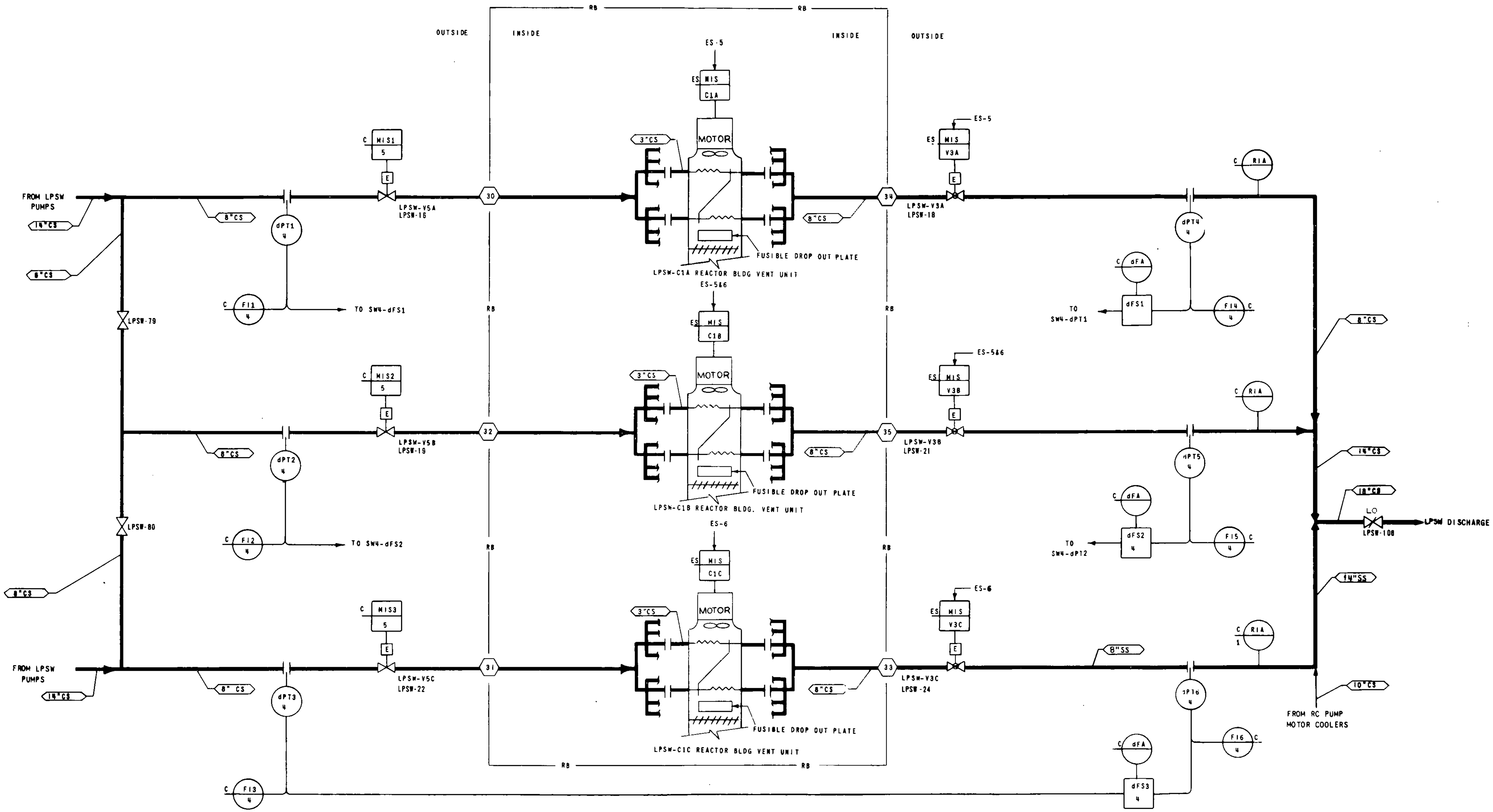
LOW PRESSURE INJECTION COOLER CAPACITY



REACTOR BUILDING SPRAY PUMP
CHARACTERISTICS

OCONEE NUCLEAR STATION

Figure 6 - 9



NOTE:
LO - LOCK OPEN

REACTOR BUILDING COOLING UNITS

OCONEE NUCLEAR STATION
Figure 6 - 10
Rev. 16 7/30/71

a. Packaged

b. Shipped

(f) Environmental Monitoring

(1) For each medium sampled during the six-month period, the following information shall be provided.

a. Number of sampling locations.

b. Total number of samples

c. Number of locations at which levels are found to be significantly greater than local backgrounds.

d. Highest, lowest, and the average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(2) If levels of station contributed radioactive materials in environmental media indicate the likelihood of public intakes in excess of 3 percent of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided. (These values are comparable to the top of Range I, as defined in FRC Report No. 2.)

(3) If statistically significant variations in off-site environmental concentrations with time are observed and are attributed to station releases, correlation of these results with effluent releases shall be provided.

6.6.1.2 Personnel Exposure and Monitoring Reports

A. This report shall be submitted to the Directorate of Licensing, USAEC, Washington, D. C., 20545 within the first quarter of each calendar year.

(1) A report of the total number of individuals for whom personnel monitoring was provided during the calendar year.

(2) A report of individuals, 18 years of age or older whose annual radiation dose exceeded the applicable quarterly numerical values, and for each individual under 18 years of age whose annual dose exceeded 10 percent of the applicable quarterly numerical values will be submitted.

B. A report shall be submitted to the employee and the Directorate of Licensing, USAEC, Washington, D. C., within 30 days after the exposure determination or 90 days from the termination of employment, whichever comes first, on the total exposure to radiation and radioactive material received during the period of employment.

6.6.1.3 Material Status

A. The licensee will file Form AEC-742 within 30 days of December 31 and June 30 to report the status of all special nuclear materials.

B. The licensee will file Form AEC-741 within 10 days of shipping or receiving special nuclear material.

30.| 6.6.2 Non-Routine Reports

6.6.2.1 Reporting of Abnormal Occurrences & Unusual Events

A. Events requiring notification within 24 hours (by telephone or telegraph to the Director of Region II Regulatory Operations Office followed by a written report within 10 days to the Directorate of Licensing, USAEC, Washington, D. C. 20545 (copy to the Directorate of Regulatory Operations, Region II, Atlanta, Georgia).

(1) Abnormal occurrences specified in Section 1.8 of the Technical Specifications.

(2) Any significant variation of measured values in a non-conservative direction from corresponding predicted values of safety connected parameters during initial criticality.

The written report, and to the extent possible the preliminary telephone or telegraph report, shall describe, analyze, and evaluate safety implications, and outline the corrective actions and measures taken or planned to prevent recurrence of (1) and (2) above.

B. Unusual Events as defined in Section 1.9 of the Technical Specifications shall be reported within 30 days to the Directorate of Licensing the Directorate of Regulatory Operations, Region II, Atlanta, Georgia.

6.6.2.2 Radiation Exposure and Monitoring

The licensee will report any over exposure, excessive radiation level or concentration to the Directorate of Regulatory Operations, USAEC, Washington, D. C. 20545, and Regulatory Operations, Region II, Atlanta, Georgia, per 10CFR20.

6.6.2.3 Loss of Licensed Material

A. The licensee will report immediately by telephone or telegraph the theft or loss of any licensed material in such quantities and under such circumstances that a substantial hazard may result to persons in an unrestricted area.

(6) Bioassays and/or whole body counts of individuals, and other surveys, as appropriate, to evaluate individual exposures and to assess protection actually provided.

(7) Records sufficient to permit periodic evaluation of the adequacy of the respiratory protective program.

(e) The licensee uses equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.7-1 below. Equipment not approved under U. S. Bureau of Mines Approval Schedules may be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type, as specified in Table 6.7-1 below.

(f) Unless otherwise authorized by the Commission, the licensee does not assign protection factors in excess of those specified in Table 6.7-1 below in selecting and using respiratory protective equipment.

(g) These specifications with respect to the provisions of 20.103 shall be superseded by adoption of proposed changes to 10 CFR 20, Section 20.103, which would make this specification unnecessary.

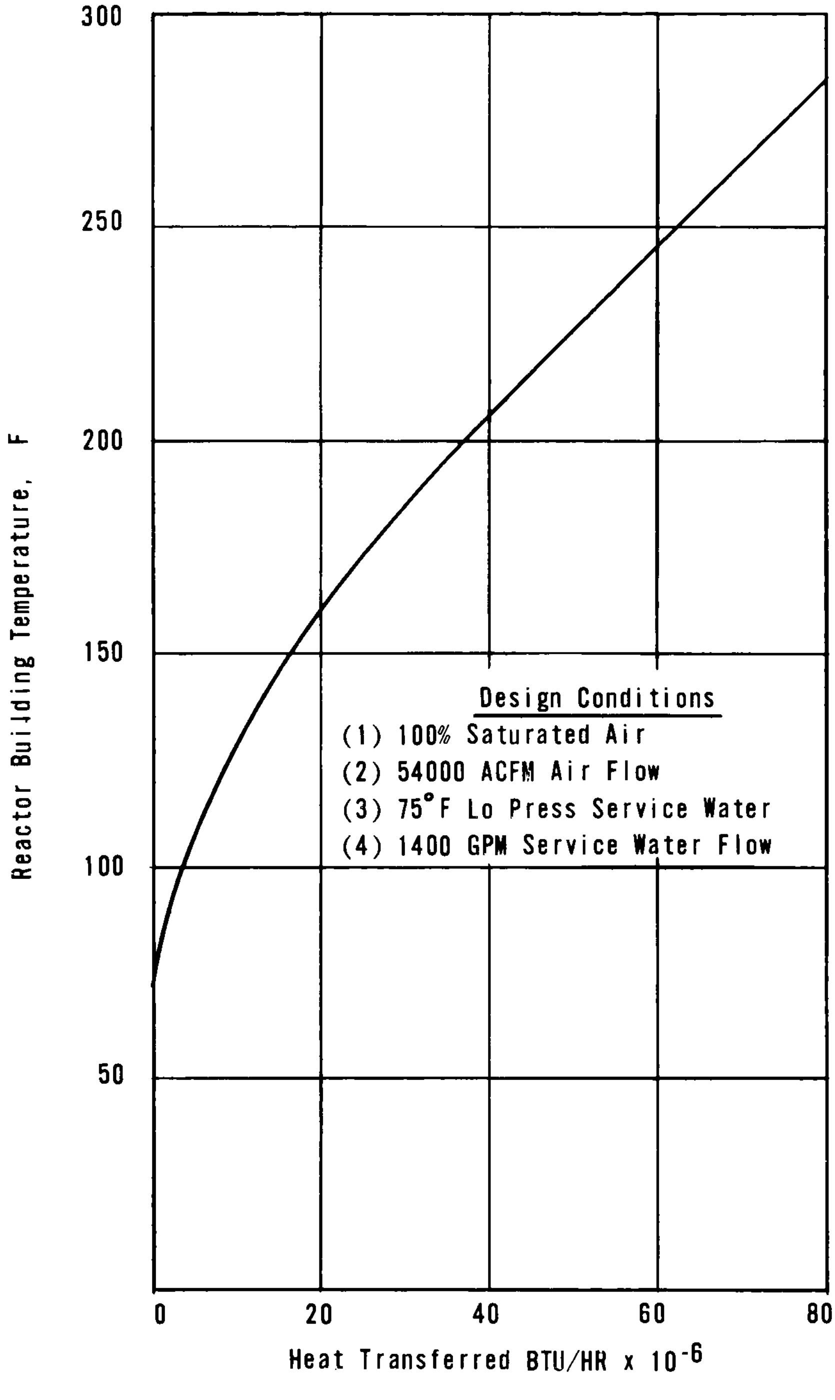
30. 6.7.2 Exposure of individuals to concentrations of radioactive noble gases may be controlled in accordance with the dose limits and requirements of Section 20.101, instead of 20.103 consistent with requirements of Specification 6.7.1a.2.a and footnote 2 referenced therein.

TABLE 6.7-1

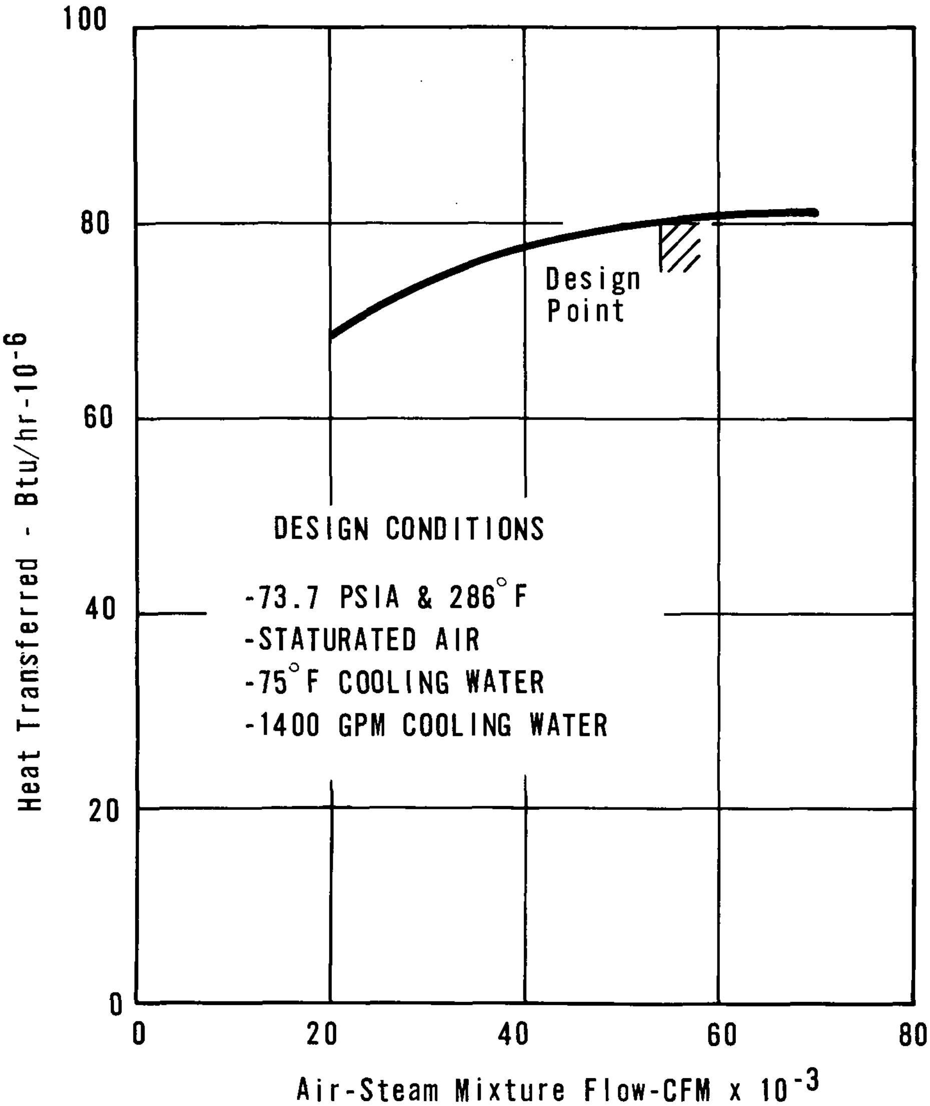
PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES 1/	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE 3/	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 7/	NP	5	21B 30 CFR 14.4(b)(4)
Facepiece, full 7/	NP	100	21B 30 CFR 14.4(b)(5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR 12.2(c)(2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR 12.2(c)(2) Type C(i)
Facepiece, full 7/	D	100	19B 30 CFR 12.2(c)(2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR 12.2(c)(2) Type C(iii)
Hood	CF	5/ See note	6/
Suit	CF	5/ See note	6/
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full 7/	D	100	13E 30 CFR 11.4(b)(2)(i)
Facepiece, full	PD	1,000	13E 30 CFR 11.4(b)(2)(ii)
Facepiece, full	R	1,000	13E 30 CFR 11.4(b)(1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19 B CFR 12.2(e) or applicable schedules as listed above

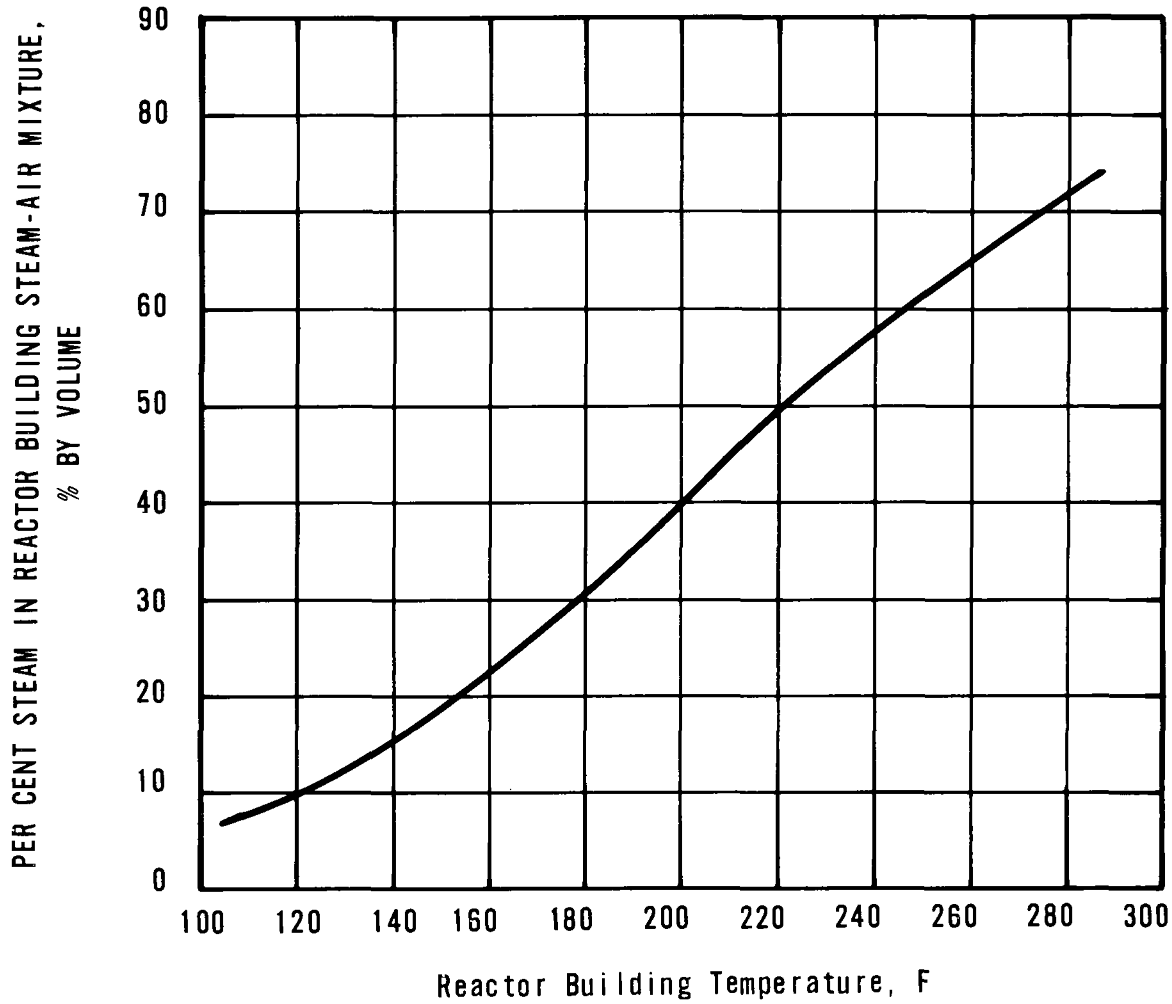
1/, 2/, 3/, 4/, 5/, 6/, 7/, [These notes are on the following pages]



REACTOR BUILDING COOLER
HEAT REMOVAL CAPACITY



REACTOR BUILDING COOLER HEAT REMOVAL CAPABILITY AS A FUNCTION OF AIR-STEAM MIXTURE FLOW



REACTOR BUILDING POST-ACCIDENT
STEAM-AIR MIXTURE COMPOSITION

LIST OF EFFECTIVE PAGES
FSAR SECTION 7

Instrumentation and Control

Page	Revision
LOEP-- 1 of 3	Rev. 37
LOEP-- 2 of 3	Rev. 37
LOEP-- 3 of 3	Rev. 26
7-i	Rev. 15
7-ii	Rev. 4
7-iii	Rev. 4
7-iv	Rev. 4
7-v	Rev. 4
7-vi	Rev. 6
7-vii	Rev. 16
7-1	Original
7-2	Rev. 5
7-2a	Rev. 4
7-2b	Rev. 6
7-3	Rev. 3
7-4	Rev. 16
7-5	Rev. 16
7-6	Rev. 15
7-6a	Rev. 15
7-6b	Rev. 15
7-7	Rev. 3
7-8	Original
7-9	Rev. 29
7-10	Rev. 4
7-10a	Rev. 19
7-10b	Rev. 4
7-11	Rev. 6
7-12	Rev. 4
7-13	Rev. 18
7-14	Rev. 18
7-14a	Rev. 4
7-15	Rev. 4
7-16	Rev. 4
7-16a	Rev. 6
7-17	Original
7-17a	Rev. 4
7-18	Original
7-19	Original
7-20	Rev. 5
7-20a	Rev. 5
7-21	Rev. 5
7-21a	Rev. 5
7-22	Rev. 19
7-23	Rev. 4
7-23a	Rev. 4
7-23b	Rev. 4
7-23c	Rev. 4
7-24	Rev. 4
7-25	Rev. 4

LIST OF EFFECTIVE PAGES
FSAR SECTION 7 (CONT'D)

Instrumentation and Control

Page	Revision	Page	Revision
7-26	Original	7-43a	Rev. 4
7-27	Original	7-44	Rev. 18
7-28	Original	7-44a	Rev. 4
7-29	Rev. 4	7-45	Rev. 6
7-29a	Rev. 4	Fig. 7-1	Rev. 16
7-30	Original	Fig. 7-2	Rev. 9
7-31	Rev. 3	Fig. 7-2A	Rev. 16
7-32	Rev. 37	Fig. 7-3	Rev. 19
7-32a	Rev. 8	Fig. 7-4	Original
7-33	Rev. 4	Fig. 7-5	Original
7-33a	Rev. 4	Fig. 7-6	Rev. 4
7-34	Original	Fig. 7-6A	Rev. 5
7-35	Rev. 37	Fig. 7-7	Rev. 4
7-35a	Rev. 6	Fig. 7-8	Original
7-35b	Rev. 4	Fig. 7-9	Original
7-36	Original	Fig. 7-10	Original
7-37	Original	Fig. 7-11	Original
7-38	Original	Fig. 7-12	Original
7-39	Original	Fig. 7-13	Original
7-40	Original	Fig. 7-14	Rev. 3
7-41	Original	Fig. 7-15	Original
7-42	Original	Fig. 7-16	Rev. 37
7-43	Rev. 4	Fig. 7-17	Original

LIST OF EFFECTIVE PAGES
FSAR SECTION 7 (CONT'D)

Instrumentation and Control

Page	Revision	Page	Revision
Fig. 7-18	Original		
Fig. 7-19	Original		
Fig. 7-20	Original		

TABLE OF CONTENTS

CONTENTS (Cont'd)

CONTENTS (Cont'd)

CONTENTS (Cont'd)

CONTENTS (Cont'd)

LIST OF TABLES

LIST OF FIGURES

(at rear of section)

7 INSTRUMENTATION AND CONTROL

Instrumentation and control systems include the Reactor Protective System, the Engineered Safeguards Protective Systems, the Rod Drive Control System, the Integrated Control System, the Nuclear Instrumentation System, the Non-Nuclear Instrumentation System, and the Incore Monitoring System.

7.1 PROTECTIVE SYSTEMS

The protective systems, which consist of the Reactor Protective System and the Engineered Safeguards Protective Systems, perform important control and safety functions. The protective systems extend from the sensing instruments to the final actuating devices, such as circuit breakers and pump or valve motor contactors.

7.1.1 DESIGN BASIS

The protective systems are designed to sense plant parameters and actuate emergency actions in the event of abnormal plant parameter values. They meet the intent of the proposed IEEE "Criteria for Nuclear Power Plant Protection Systems" dated August 1968, (IEEE No. 279).

7.1.1.1 Single Failure

The protective options meet the single failure criterion of IEEE No. 279 to the extent that:

a. No single component failure will prevent a protective system from fulfilling its protective functions when action is required.

b. No single component failure will initiate unnecessary protective system action where implementation does not conflict with the criterion above.

7.1.1.2 Redundancy

All Reactor Protective System functions are implemented by redundant sensors, measuring channels, logic, and action devices. These elements combine to form the protective channels as defined in Section 15.

7.1.1.3 Independence

Redundant protective channels are electrically independent and packaged to provide physical separation.

7.1.1.4 Separation

Protective channels are physically separate and are electrically isolated from regulating instrumentation. Only one string of instrumentation may be selected at a given time for use in a system control function, and electrical isolation is assured through the use of isolation amplifiers. A fifth channel of regulating instrumentation not associated with protection is normally employed for control purposes.

7.1.1.5 Manual Trip

Manual trip switches, independent of the automatic trip instrumentation are provided.

7.1.1.6 Testing

Manual testing facilities are built into the protective systems to provide for:

a. Preoperational testing to give assurance that a protective system can fulfill its required protective functions.

b. On-line testing to prove operability and to demonstrate reliability.

7.1.1.7 Environment

3. Protective instrumentation within the reactor building is designed for continuous operation in an ambient of 40 F to 120 F, 60 psig, and 100 percent relative humi-
4. dity, except that neutron detectors are designed for 90 percent relative humidity.
5. All instrumentation in the reactor buildings required for safe shutdown and operation of the nuclear steam supply systems will be operable during the periodic integrated leak rate tests. Selected sensors in the Engineered Safeguards Systems will withstand the building environmental conditions during the loss-of-coolant accident.

The following design bases apply to control and electrical equipment located in the control area (control room, cable room, and electrical equipment room.) Reactor Protective System and Engineered Safeguards Protection System equipment is designed for continuous operation in a room environment of 40F to 110F and up to 75 RH. All modules are designed to operate at temperatures in the range 40F to 140F. The increased upper limit allows for a 30F rise inside the equipment cabinets over the ambient room temperature. Each module has been tested to confirm proper operation under design environmental conditions by placing it in a controlled environment and
4. observing deviations in output voltages with known inputs. The results of these tests indicate the two systems will withstand 90 RH for 24 hours and 89 RH continuously.

All other safety related control and electrical equipment is designed to function in its environment of 110F ambient temperature in the control room and 120F ambient temperature in the cable and equipment rooms. The equipment will be tested to verify proper operation in its environment.

The control area air conditioning and ventilation systems (9.8.2.2 and Table 9-13) are conservatively designed to provide a suitable environment for the control and electrical equipment. In addition, redundant air conditioning and ventilation equipment is provided, as summarized below, to assure that no single failure of an active component within these systems will prevent proper control area environmental control.

(1) Two 100 percent capacity supply fans with filter banks and chilled water coils.

(2) Two 100 percent capacity central station type chilled water systems.

(3) Two 50 percent capacity outside air booster fans.

Rev. 3. 3/16/70
Rev. 4. 4/20/70
Rev. 5. 5/25/70

In the unlikely event that both supply fans or both chilled water systems are inoperative, the temperature in the control area would exceed temperature limitations, requiring the affected unit(s) to be shut down. The limiting temperature would be reached in the following approximate time intervals:

control room	110F	28 min
cable room	120F	13 min
electrical equipment room	120F	27 min

With the low latent heat gain from the outside air supply, the relative humidity in the area would not exceed 50 percent and the dewpoint would be less than 74 F; therefore,the safety related equipment will not be adversely affected by airborne moisture.
The above times were conservatively calculated as follows:

(1) The heat gain is based on all equipment loaded to its rating with no diversity or demand factor.

(2) No credit was taken for heat transfer to cold heat sinks such as masonry walls, etc.

(3) It was assumed that the booster fans were inoperative; therefore, the calculations assume no cooling from outside air.

(4) The booster fans were considered to be in operation only when considering the effects of high humidity.

All other safety related control and electrical equipment is designed to function in its environment of 110 F ambient temperature in the control room and 120 F ambient temperature in the cable and equipment rooms. The equipment will be tested to verify proper operation in its environment.

7.1.1.8 Seismic

The protective systems are designed to function normally during and after either a maximum hypothetical earthquake (MHE) or design earthquake. The nuclear instrumentation detectors and all equipment mounted in the Nuclear Instrumentation Reactor Protective System cabinets and in the Engineered Safeguards Protective System cabinets are being dynamically tested to show normal operation during excitation in excess of the maximum predicted accelerations at their locations through the frequency range expected during either earthquake.

The seismic design basis for other instrumentation and controls, including final actuation devices, for automatic initiation of engineered safeguards including residual heat removal system is based on the design or maximum hypothetical earthquake. This equipment is designed to assure it will not lose its capability to perform its intended function during and following the design bases event. If a seismic disturbance occurred after a major accident the engineered safety features will perform their intended function.

These instrumentation and control devices are being evaluated to show their ability to withstand the seismic conditions in accordance with the design objective by either predicting the equipment's performance by analysis, testing the equipment to determine its resonance frequency, and applying the applicable building response curves or testing the equipment under simulated seismic conditions. To assure that the seismic design is met on the equipment, the equipment or components comprising the equipment assemblies will be
4. analyzed or tested by the vendors, private testing laboratories, or Duke Power Company personnel using seismic load as obtained from the building response calculations to show that stresses are within allowable limits and will not result in loss of function. Testing of this equipment will be done and reports compiled to verify the seismic design objectives.

6. A summary of the seismic design bases which apply to the Reactor Protective System and Engineered Safety Features is given in Table 7-8, page 7-45.

7.1.2 REACTOR PROTECTIVE SYSTEM

The Reactor Protective System (RPS) monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage. It also assists in protecting against reactor coolant system damage caused by high system pressure by limiting energy input to the system through reactor trip action.

7.1.2.1 Design Basis

The Reactor Protective System (RPS) includes all design basis features of Section 7.1.1 with the following additions:

7.1.2.1.1 Loss of Power

A loss of power to a Reactor Protective channel will cause that protective channel to trip.

7.1.2.1.2 Equipment Removal

The Reactor Protective System initiates a protective channel trip whenever a module or subassembly is removed from the equipment cabinet. Provisions are made in each protective channel to supply an input signal which leaves the channel in a non-tripped condition for testing and maintenance.

Rev. 4. 4/20/70
Rev. 6. 6/22/70

7.1.2.2 System Design

7.1.2.2.1 System Logic

The system as shown in Figure 7-1 consists of four identical protective channels, each terminating in a trip relay within a Reactor Trip (RT) Module. In the normal untripped state, each protective channel functions as an AND gate, passing current to the terminating relay and holding it energized as long as all inputs are in the normal energized (untripped) state. Should any one or more inputs become de-energized (tripped), the terminating relay in that protective channel de-energizes (trips). Thus, for trip signals each protective channel becomes an OR gate.

Each of the four protective channels terminates in a channel trip relay within a reactor trip module. There are four such modules. Each protective channel trip relay has four contacts, each controlling a logic relay in one reactor trip module. Therefore, each reactor trip module has four logic relays controlled by the four protective channels. The four logic relays combine to form a 2-out-of-4 coincidence network in each reactor trip module. The coincidence logics in all reactor trip modules trip whenever any two of the four protective channels trip.

The reactor trip modules are given the same designation as the protective channel whose trip relay they contain and in whose cabinet they are physically located. Thus, the protective channel-A reactor trip module is located in protective channel A cabinet, etc. (Fig. 7-1). The coincidence logic in each reactor trip module controls one or more breakers in the control rod drive power system.

The coincidence logic contained in the Reactor Protective System channel A RT module controls breaker A in the control rod drive system as shown in Figure 7.1, channel B RT module controls breaker B, channel C RT module controls breaker C
3. and contactor E, and channel D RT module controls breaker D and contactor F. Breakers A and B control all the 3 phase primary power to the rod drives; breakers C and D control the d-c power to rod groups 1 through 4; and breakers
3. E and F control the gating power to rod groups 5 through 8.

The control rod drive circuit breaker combinations that initiate a reactor trip can best be stated in logic notation as: AB + ADF + BCE + CDEF. This is a 1-out-of-2 logic used twice and is referred to as a 1-out-of-2 x 2 logic. It should be noted that when any 2-out-of-4 protective channels trip, all reactor trip module logics trip commanding all control rod drive breakers to trip.

The undervoltage coils of the control rod drive breakers receive their power from the protective channel associated with each breaker. The manual reactor trip switch is interposed in series between each RT module logic and the assigned breakers undervoltage coil.

7.1.2.2.2 Summary of Protective Functions

The four Reactor Protective System protective channels are identical in their functions, which combine in the system logic to trip the reactor automatically and protect the reactor core for the following conditions:

a. When the reactor power, as measured by neutron flux, exceeds a fixed maximum limit.

11. b. When the reactor power, as measured by neutron flux, exceeds the limit set by the reactor coolant flow and power imbalance.

10.

c. When the reactor power exceeds the limit set by the number and combination of reactor coolant pumps in operation.

d. When the reactor outlet temperature exceeds a fixed maximum limit.

e. When a specified reactor pressure-outlet temperature relationship is exceeded.

f. When the reactor pressure falls below a fixed minimum limit.

15. g. When reactor building pressure exceeds a fixed maximum limit.

The RPS also automatically trips the reactor to protect the reactor coolant system whenever the reactor pressure exceeds a fixed maximum limit.

The abnormal conditions that initiate a reactor trip are keyed to the above listing and tabulated in Table 7-1.

Table 7-1
Reactor Trip Summary
16. (Reference Supplement 9 Revisions for Oconee 3)

	Trip Variable	No. of Sensors	Steady-State Normal Range	Trip Value or Condition for Trip
	Over Power	4 Flux Sensors	0-100%	107.5% of rated power.
11.	Nuclear Over Power Based on Flow and Imbalance	4 Two-Section Flux Sensors 8 ΔP Flow	NA	1.10 times flow minus reduction due to imbalance
10.				
4.	Power/RC Pumps	4 Pump Monitors	2 to 4 pumps	Loss of one operating coolant pump motor in each loop and reactor neutron power exceeds 55% rated power. Loss of two operating reactor coolant pump motors in one loop.

Rev. 4. 4/20/70 Rev. 16. 7/30/71
Rev. 10. 8/28/70
Rev. 11. 9/8/70
Rev. 15. 12/30/70

Table 7-1 (Cont'd)

Trip Variable	No. of Sensors	Steady-State Normal Range	Trip Value or Condition for Trip
4.		2 Pumps	Loss of one of two operating reactor coolant pump motors in one loop
Reactor Outlet Temperature	4 Temperature Sensors	532-604 F	619 F
Pressure/Temperature	4 Pressure Sensors 4 Temperature Sensors	NA	$(13.26T_{out}-5989*) \geq P$
9. Reactor Coolant Pressure	4 Pressure Sensors	2,090-2,220 psig	2,355* psig (high) 1,800 psig (low)
15. Reactor Building Pressure	4 Pressure Sensors	0 psig	4 psig

*These are actual instrumentation settings chosen to protect the core and allow for the difference between core outlet pressure and measured system pressure.

16. 7.1.2.2.3 Description of Protective Channel Functions (Reference Supplement 9 Revisions for Oconee 3)

The functions of the RPS described below apply to each protective channel.

1. Over Power

   The nuclear instrumentation provides a linear neutron flux signal in the power range as an indication of reactor power to a protective system bistable trip module.

   When the neutron flux signal exceeds the trip point of the bistable, the bistable trips, de-energizing the associated protective channel trip relay.

11. 2. Over Power Trip Based on Flow and Imbalance

   Neutron flux and the reactor coolant flow are continuously monitored. A linear neutron flux signal is received from the nuclear instrumentation and a total reactor coolant flow signal is received from the flow tubes. A power level trip setpoint is established for a bistable trip module as the percentage reactor coolant flow rate multiplied by 1.10. The reactor power imbalance (power in the top half of the core
10. minus the power in the bottom half of the core) reduces the power level trip setpoint such that the four pump power-imbalance boundaries in Figure 7-2a are not exceeded. Less than four pump power-imbalance protection is provided by the power level trip setpoint decrease due

to flow decrease. When the neutron flux signal exceeds the power
10. level trip setpoint established by the total reactor coolant flow
and the reactor power imbalance the bistable trips, de-energizing
the associated protective channel trip relay.

All flow ΔP cells for a single loop are connected to common 1-inch
"low" and "high" lines from the flow tube in that loop. Severance
of the "low" line will result in maximum indicated flow for the loop
4. in all four protective channels. All console indicators for that
loop will go to 110 percent full flow. Severance of the "high" line
will result in zero indicated flow for the loop and possibly a power/
flow reactor trip.

3. Power/RC Pumps Trip

3. The reactor coolant (RC) pumps are monitored to determine that they
are running. Loss of a single pump initiates four independent
signals, one to each protective channel. This information is re-
ceived by a pump monitor logic which counts the number of RC pumps
in operation and identifies the coolant loop in which the pumps are
operating. The pump monitor logic output controls the trip point of
a power/pump comparator, and initiates a protective channel trip for
the conditions in Table 7-1. Normally, the trip point corresponding
to only two pumps in one loop in operation is set at 0% FP. If two
9. pumps in one loop are lost, a reactor trip will be initiated. Prior
to startup with two pumps in one loop, the Power/RC Pumps trip point
corresponding to only two pumps in one loop in operation must be
increased to 55% FP.

4. Reactor Outlet Temperature Trip

The reactor outlet temperature is measured by resistance elements. The bridge for each resistance element is considered a part of, and is located within, its associated protective system channel cabinet.

3. The reactor outlet temperature signal from the temperature bridge
passes through a signal converter and then is applied to a bistable
trip module. When the temperature exceeds the trip point of the
bistable, the bistable trips, de-energizing the protective channel
trip relay. Prior to startup with two pumps in one loop, one of the
9. two reactor protective channels receiving outlet temperature infor-
mation from the idle loop must be tripped.

5. Pressure Temperature Trip

Figure 7-2 shows the operating reactor coolant pressure-temperature
boundaries formed by the combined reactor high temperature, high
pressure, low pressure, and the pressure-temperature comparator trip
settings. The pressure-temperature comparator trips whenever the
3. specified reactor pressure-outlet temperature relationship is ex-
ceeded. The comparator forms the boundary line A-B in Figure 7-2.

Rev. 3. 3/16/70
Rev. 4. 4/20/70
Rev. 9. 8/11/70
Rev. 10. 8/28/70
Rev. 15. 12/30/70
(Carry-Over)

6. Reactor Pressure Trip

The reactor coolant pressure signal from the pressure transmitter is received by an isolation module in the associated protective channel cabinet. This module acts as a signal conditioner and isolation unit.

Pressure signals go to a high pressure bistable trip module and a low pressure trip module. When the pressure exceeds the trip point of the high pressure bistable, the bistable trips de-energizing the protective channel trip relay.

The low pressure bistable trips when the pressure falls below the trip point, tripping the protective channel trip relay.

7. Reactor Building Pressure Trip

Each of the four protective channels receives reactor building pressure information from an independent pressure switch. A contact
15. buffer in each protective channel continuously monitors the state of the associated pressure switch. When the state of the pressure switch changes to that corresponding to a reactor building pressure exceeding the trip point specified in Table 7-1, the contact buffer de-energizes the protective channel's trip relay.

7.1.2.2.4 Set Point Adjustments for Single Loop Operation

Prior to startup with two (2) pumps in one (1) loop, the following adjustments must be made to the Reactor Protective System:

(1) The power/RC pumps trip point corresponding only to two pumps in one loop in operation must be increased to 55% rated power.

9.

(2) One of the two reactor protective channels receiving outlet temperature information from the idle loop must be tripped.

15. 7.1.2.2.5 Availability of Information

The modules, logic, and analog equipment associated with a single protective channel are contained wholly within two Reactor Protective System cabinets. Within these cabinets, there is a meter for every analog signal employed by the protective channel, and a visual indication of the state of every logic element. At the top of one cabinet, and easily visible at all times, is a protective channel status panel. Lamps on this panel give a quick visual indication of the trip status of the particular protective channel and of the
3. RT module associated with it. Additional lamps on the panel give visual indication of a channel bypass or a fan failure.

In addition to the visual indications and readouts within the protective channel cabinets, each trip function, power supply, and analog signal is monitored by the plant computer. Separate from the computer, trip actions are sequence-annunciated in the plant status annunciator. Such sequencing permits the operator to identify readily the protective channel trip actions. Process
10. instrumentation including power, imbalance, flow, temperature, and pressure is indicated on the main control console.

Plant annunciator windows provide the operator with immediate indications of changes in the status of the reactor protective system. The following conditions are annunciated for each reactor protective system channel:

a. channel trip

b. fan failure in channel

c. channel on test

4. 6. d. shutdown bypass initiated

e. manual bypass initiated

8. f. dummy bistable installed

Any time a test switch is in other than the operate position, annunciator (c) will be lit and the associated protection channel will be tripped. Under this condition, annunciator (a) will be lit unless annunciator (e) is lit (i.e., the channel is bypassed).

Rev. 4. 4/20/70
Rev. 6. 6/22/70
Rev. 8. 7/23/70
Rev.15. 12/30/70
(Carry-Over)

7.1.2.3 System Evaluation

7.1.2.3.1 System Logic

The RPS is a four-channel, redundant system in which the four protective channels are brought together in four identical 2-out-of-4 logic networks of the RT modules. A trip in any 2 of the 4 protective channels initiates a trip of all four logic networks. The system to this point has the reliability and advantages of a pure 2-out-of-4 system.

Each of the reactor trip modules (2-out-of-4 logic networks) controls a control
3. rod drive breaker or contactor. Thus, a trip in any 2 of the 4 protective channels
initiates a trip of all the breakers and contactors. The breakers and contactors,
however, are arranged in what is effectively a 1-out-of-2 logic (Figure 7-6).
This system combines the advantages of the 2-out-of-4 and the 1-out-of-2 x 2
systems, while eliminating some of the disadvantages of the 1-out-of-2 x 2 system
3. alone. The combination results in a system that is considered superior to either
of the basic systems alone.

In evaluating system performance, it is arbitrarily assumed that "failure" can either prevent a trip from occurring or can initiate trip action.

3. The redundant Reactor Protective System inputs operate in a true 2-out-of-4 logic mode so that the failure of an input leaves the system in either a 2-out-of-3 or a 1-out-of-3 logic mode, with either state providing sufficient redundancy for reliable performance.

The system can tolerate several input function failures without a reduction in performance capability provided the failures occur in unlike variables in different protective channels, or are of a different mode in different protective channels, or all occur within one protective channel. When a single protective channel fails, the system is left in either a 2-out-of-3 or 1-out-of-3 logic mode as explained below.

The protective channel trip relay of each channel is located in the reactor trip
3. module associated with the channel. Within each reactor trip module there is a
logic relay for each protective channel. These combine in each module to form the 2-out-of-4 logic. A Failure Mode and Effects analysis of the reactor trip module has demonstrated that single failures within the module or in its interconnections can produce only the following effects:

1. Trip the breaker associated with the module.
2. Place the system in a 2-out-of-3 mode, as if the associated protective channel had suffered a can't trip failure.
3. Place the system in a 1-out-of-3 mode, as if the associated protective channel had tripped.

The combination of reactor trip modules and control rod drive breakers and contactors
form a 1-out-of-2 x 2 logic. At this level the system will tolerate a "cannot
trip" type of failure of one reactor trip module, or of the breaker and/or con-
3. tactors associated with one reactor trip module without degrading the system's
ability to trip all control rods. The failure analysis demonstrates that no single failure involving a reactor trip module will prevent its associated breakers and contactors from opening.

7.1.2.3.2 Redundancy

The redundancy of the Reactor Protective System could be demonstrated by physically removing all the components associated with a single protective channel. Doing so would have all the remaining components and protective channels operational in a 1-out-of-3 system.

7.1.2.3.3 Electrical Isolation

All signals leaving the Reactor Protective System are isolated from the system either by the use of isolation amplifiers for analog signals, or by relay contacts (in the case of digital signals). The effect of this isolation is to prevent faults occurring to signal lines outside of the Reactor Protective System cabinets from being reflected into more than one protective channel. The isolation thus provided also assures that two or more protective channels cannot interact through the cross-coupling or faulting of related signal lines.

Faults such as short, open, or grounded circuits and cross-coupling of analog output signals from two or more channels have no effect upon the protective channels or their functions.

The isolation amplifier circuits have been prototype tested to assess their effectiveness to isolate the input signal from output circuit faults. They are capable of blocking a direct connection (i.e., a hot short) across their output of 410 vdc (300 v rms) without effecting the input source. The redundancy and coincidence logic of the system permits the system to tolerate failures and thus reduces the chance of an inadvertent reactor trip.

7.1.2.3.4 Periodic Testing and Reliability

The use of 2-out-of-4 logic between protective channels permits a channel to be tested on-line without initiating a reactor trip. Maintenance to the extent of removing and replacing any module within a protective channel may also be accomplished in the on-line state without a reactor trip.

To prevent either the on-line testing or maintenance features from creating a means for unintentionally negating protective action, a system of interlocks initiates a protective channel trip whenever a module is placed in the test mode or is removed from the system. However, provisions are made in each protective channel to supply an input signal which leaves the channel in a non-tripped condition for testing or maintenance.

The test scheme for the reactor protective system is based upon the use of comparative measurements between like variables in the four protective channels, and the substitution of externally introduced digital and analog signals as required, together with measurements of actual protective function trip points. A digital voltmeter is provided for making accurate measurements of trip point and analog signal voltages.

On-line testing may be performed at different intervals and levels within the system consistent with satisfactory system reliability characteristics. The reliability of the system for random failures has been assured by careful selection of components, failure testing of logic elements, environmental testing of the system's modules, and long-term prototype proof-testing with the Babcock and Wilcox Test Reactor (BAWTR).

The reliability of the system logic, primarily the relays and coincidence networks in the RT modules, has been made very high so as to eliminate the need for frequent tests of the logic. The logic relays are of two classes; one class designed for high speed, light electrical loads, and more than 10^7 operations under load; and the other class for switching electric loads of up to 10 amperes and more than 10^6 operations. Confirmation tests of operational reliability of these two types of relays, operated under load as they are used in the RPS, have been performed with no sign of failure or wear to 5×10^6 and 1.2×10^6 operations respectively.

The system test scheme includes frequent visual checks and comparisons within the system on a regular schedule in which all protective channels are checked at one time, together with less frequent electrical tests conducted on a rotational plan in which tests are conducted on different protective channels at different times.

A regular check of all Reactor Protective System indications is required. The check includes such things as comparing the value of the analog variables between protective channels and observing that the equipment status is normal. In addition, power-range protective channel readings are compared with a thermal calculation of reactor power. These checks are designed to detect the majority of failures that might occur in the analog portions of the system as well as the self-annunciating type of failure in the digital portions of the system. The electrical tests are designed to detect more subtle failures that are not self-evident or self-annunciating and are detectable only by testing.

Electrical tests are conducted on a rotational basis in accordance with a preliminary test scheme as follows:

29. 1. **Prior to startup (if the reactor has been shut down for greater than seven days), all Reactor Protective System channels, logic, and control rod drive power breakers are electrically trip tested to prove their operability.**

2. One week after startup, protective channel A is electrically trip tested for every input up to and including the channel trip relay.

3. Two weeks after startup protective channel B is similarly tested.

4. Three weeks after startup protective channel C is similarly tested.

5. Four weeks after startup protective channel D is similarly tested.

6. Five weeks after startup, protective channel A is similarly tested.

19. The rotational cycle is repeated so that each protective channel is electrically trip tested every 4 weeks.

19. The control rod drive power breaker with a reactor trip module will be tested monthly.

Rotational testing has several advantages. It significantly reduces the probability of system failure as compared to testing all protective channels at one time. It also reduces the chance of systematic errors entering the system.

7.1.2.3.5 Physical Isolation

The need for physical isolation has been met in the physical arrangement of the protective channels within separate cabinets and wiring within the cabinets separating power and signal wiring so as to reduce the possibility of some physical event impairing system functions. The systems sensors are separated from each other. There are four pressure taps for the reactor coolant pressure measurements to reduce the liklihood of a single event effecting more than one sensor. Outside the Reactor Protective System cabinets, vital signals and wiring are separated and physically protected to preserve protective channel independence and maintain system redundancy against physical hazards.

Redundant detectors and transmitters applied in the Reactor Protective System are located to provide physical separation. Redundant out of core nuclear detectors are located in separate quadrants around the reactor vessels. Two resistance thermometers assigned to the RPS are located on each reactor coolant outlet header. Cables approach redundant temperature detectors from opposite directions. Redundant pressure transmitters are located outside the secondary shield in four separate quadrants of the reactor buildings. Two reactor coolant pressure transmitters for RPS are connected to each of the two loops. Separate flow transmitters for each RPS channel are applied to sense the flow in each loop. This arrangement results in detectors and transmitters associated with one RPS channel being located in essentially (the reactor vessels are not in the center of the reactor buildings) the same quadrant of a reactor building, and
4. with redundant detectors and transmitters located in another quadrant of the reactor building. Since each RPS channel receives a flow signal from both loops, one of the flow transmitters for each channel is not located with the other RPS transmitters for that channel. Location and cable routing for these transmitters is such that separation of at least seven feet is provided between redundant channels inside the reactor buildings. Cables for redundant RPS and ES detectors and transmitters are routed in separate directions in trays carrying no other cables to four separate reactor building penetrations. These penetration assemblies are assigned to nuclear instrumentation, ES instrumentation, and RPS cables exclusively. Two of these penetration assemblies are located sixty feet apart in separate quadrants of each reactor building. One is used for RPS and ES channel A instrumentation; the other for RPS and ES channel B instrumentation. A penetration assembly for RPS and ES channel C instrumentation and one for RPS channel D are located on the opposite side of the reactor buildings thirty feet apart. Located under the control rooms between the outside of the reactor buildings and the cable and equipment rooms, four separate trays are provided per unit which carry nothing but nuclear, RPS and ES instrumentation cables. Three separate routes are followed by these trays. RPS channel C and RPS channel D follow the same route but are separated vertically by 1-1/2 feet. A detailed review of cable tray and pipe routing in this area indicates that no more than two RPS channels could be damaged by a single pipe failure or missile. Equipment locations in the auxiliary building provide the basis for vertical arrangement of trays following the same route from the reactor buildings. Switchgear for power equipment is located at lower elevations and instrumentation cabinets are located at higher elevations. Therefore, vertical separation of classes of cables in trays is as follows from top trays down:

a. Instrumentation cable trays
b. Control cable trays
c. Power and control cable trays
d. Power cable trays

Inside the cable rooms cables from each protective channel are routed in trays separate from those carrying cables from any other protective channel. Included in these trays are instrumentation cables from the reactor building, control and interconnecting cables associated with that protective channel, and non-protective
4. instrumentation and control cables. Both protective and non-protective cables are individually armored and are flame retardant.

Reactor trip cables from the four RPS cabinets are routed separately to a reactor trip switch located on the main control board. From the trip switch the cables follow four separate paths to the reactor trip breakers and the control rod drive cabinets.

7.1.2.3.6 Primary Power

The primary source of 120V ac power for the Reactor Protective System comes from four vital buses, one for each protective channel, as described in 8.2.2.8.

7.1.2.3.7 Manual Trip

Manual trip may be accomplished from the control console by a trip switch. This trip is independent of the automatic trip system. Power from the control rod drive power breakers' undervoltage coils comes from the RT modules. The manual trip switches are between the reactor trip module output and the breaker undervoltage coils. Opening of the switches opens the lines to the breakers, tripping them. There is a separate switch in series with the output of each reactor trip module. All switches are actuated through a mechanical linkage from a single pushbutton.

7.1.2.3.8 Bypassing

Each protective channel is provided with two key-operated bypass switches, a channel bypass switch and a shutdown bypass switch.

The channel bypass switch enables a protective channel to be bypassed without initiating a trip. Actuation of the switch initiates a visual alarm on the main console which remains in effect during any channel bypass. The key switch will be used to bypass one protective channel during on-line testing. Thus, during on-line testing the system will operate in 2-out-of-3 coincidences. The use of the channel bypass key switch is under administrative control.

3. The shutdown bypass switch enables the power/imbalance/flow, power/RC pumps, low pressure, and pressure-temperature trips to be bypassed allowing control rod drive tests to be performed after the reactor has been shut down and depressurized below the low reactor coolant pressure trip point. Before the bypass may be initiated, a high pressure trip bistable - which is incorporated in the shutdown
19 bypass circuitry - must be manually reset. The set point of the high pressure bistable (associated with shutdown bypass) is set below the low pressure trip point. If pressure is increased with the bypass initiated, the channel will trip when the high pressure bistable (associated with shutdown bypass) trips. The use of the shutdown bypass key switch is under administrative control.

For maintenance purposes a bistable may be removed from the system and a dummy bistable inserted in its place, thus bypassing the original function. This operation forces the protective channel into a trip state upon removal of the bistable. Thus, the removal and substitution cannot be performed without passing through a tripped condition. The use of dummy bistable modules is under administrative control.

7.1.3 ENGINEERED SAFEGUARDS PROTECTIVE SYSTEM

The Engineered Safeguards Protective System (ESPS) monitors parameters to detect the failure of the reactor coolant system and initiates operation of the high and low pressure injection systems, the building isolation, the reactor building cooling and the building spray systems. In addition, the signal is used to start the standby power source and initiate a transfer to the standby power source when required as described in 8.2.3.1 and 8.2.3.2.

7.1.3.1 Design Basis

The design basis of the system includes the items of Section 7.1.1 with the following additions:

Carry-over from page 7-10

Rev. 3. 3/16/70
Rev. 4. 4/20/70
(Carry-Over)

7.1.3.1.1 Loss of Power

a. The loss of vital bus power to the instrument strings will, with the
6. exception of reactor building spray, initiate a trip of that portion of the logic associated with the affected instrument string.

b. The loss of vital bus power to the system logic will not initiate system actuation.

7.1.3.1.2 Equipment Removal

a. Removing modules from an instrument string will, with the exception of
6. reactor building spray, initiate a trip in that portion of the logic associated with the affected instrument string.

b. Removing logic modules from one protective channel does not affect any other protective channel and does not initiate system action.

7.1.3.2 System Design

7.1.3.2.1 System Logic

The Engineered Safeguards Protective System is a basic 2-out-of-3 coincidence logic system. Each input variable is measured three times, the three redundant signals terminate in three bistables as shown in Figure 7-3.

The Engineered Safeguards Protective System consists of eight 2-out-of-3 coincidence logic networks for actuating the equipment in four safeguards systems, thus each system is actuated by a pair of 2-out-of-3 logic and its outputs are referred to as an Engineered Safeguards Protective System channel. Each safeguards system is therefore actuated by two redundant coincident logics or protective channels.

The coincidence logic output is normally de-energized. Trip action consists of closing the electrical path through the logic.

The output of the protective channel coincidence logic is connected to the
4. channel's unit control module (UC modules). There is one UC module for every item, (pump, valve, etc.) controlled by the protective channel. A protective channel's UC modules are connected in parallel with the output of the coincidence logic.

The output of the coincidence logic follows a normally closed path in each UC module, finally terminating in an output relay, R_O, within each module. The
4. R_O relays of a protective channel's UC modules are driven in parallel with the output of the protective channel coincidence logic.

The contacts of the R_O relay are normally open across a control line terminating in a control relay, CR, in the controller of the equipment assigned to
4. the individual UC module. Power for operating the CR relay is taken from the equipment controller in series with the R_O relay contacts. Trip action involves energizing the R_O relay, closing its contacts which in turn energizes the CR relay actuating the assigned equipment.

Each protective channel is equipped with a logic test module (LT module). The LT module, together with the UC module, provides the necessary circuitry to permit trip testing of an individual protective device without tripping an entire protective system or channel.

The UC module also provides a means whereby following a system trip an individual protective device may be removed from the control of the Engineered Safeguards Protective System and returned to manual control. This block action cannot be initiated prior to a system trip.

The design of the system's logic can be summarized in terms of the systems operation as follows:

a. Each protective action is initiated by either of two protective channels with 2-out-of-3 coincidence between input signals.

b. Protective action is initiated by applying power from the protective channel logic to the individual R_O relays in the UC modules, which in turn energize the CR relays in each protective device controller.

c. There is a UC module for every safeguards system component (valve, pump, etc.).

7.1.3.2.2 High and Low Pressure Injection Systems

The instrumentation, logic, and actuation of the High and the Low Pressure Injection systems are identical in design. The systems differ only in their actuation set point.

There are three independent reactor coolant pressure sensors. The output of each sensor terminates in an isolation amplifier which provides individually isolated outputs. One output of each pressure measurement goes to the plant computer for monitoring. One output goes to bistables, for initiating high pressure injection action and for low pressure injection action. The bistables are identical except for their set point. Bistable action is initiated when the low reactor coolant pressure set points are reached.

The output of the three high pressure injection system bistables is combined in series with the trip outputs of three reactor building pressure bistables. The combination of reactor coolant pressure and reactor building pressure bistables outputs allows either variable to initiate high pressure injection.

The series outputs of the bistables are brought together in two identical 2-out-of-3 coincidence logics which form two Engineered Safeguards Protective System channels. Either of the two protective channels is independently capable of initiating the required protective action through redundant high pressure injection system equipment.

The outputs of the three low pressure injection system bistables are also combined in series with the independent trip outputs of the three reactor building pressure bistables. The combination functions in identically the same way as described for the high pressure injection system, with two 2-out-of-3 coincidence logics and protective channels.

7.1.3.2.3 Reactor Building Cooling and Reactor Building Isolation Systems

There are three reactor building pressure sensors. The output of each sensor terminates in an input isolation amplifier, which provides individually isolated outputs. One isolated output of each pressure measurement goes to the plant computer for monitoring. One output of each pressure measurement goes to a bistable which initiates action when its high building pressure trip point is exceeded. Each input isolation amplifier module contains an analog meter for indicating the measured pressure. Each of the three bistables has contact outputs that are combined in series with the output of the high and low pressure injection system bistables as previously described.

The outputs of the three bistables are brought together in two identical 2-out-of-3 coincidence logics which provide two Engineered Safeguards Protective System channels. Either of the two channels is independently capable of initiating the required protective action. Each protective channel uses redundant protective system devices.

7.1.3.2.4 Reactor Building Spray System

The Engineered Safeguards Protective System channels of the reactor building
spray system are formed by two identical 2-out-of-3 logic networks with the
6. 4. active elements originating in six reactor building pressure sensing pressure
switches.

The independent pressure switches containing normally open contacts form one
6. protective channel's 2-out-of-3 logic inputs. Three other identical pressure
4. switches form the 2-out-of-3 logic inputs of the second protective channel.
Either of the two protective channels is capable of initiating the required
protective action.

7.1.3.2.5 Availability of Information

All system analog signals are indicated within the system cabinets and are
4. monitored by the plant computer. All bistable outputs are indicated within
the cabinets. Logic outputs are indicated within the cabinets and their state
monitored by the plant computer.

Plant annunciators provide the operator with immediate indication of changes in
4. the status of the ESPS. Points annunciated are illustrated in Figure 7-3. In-
18. cluded are all test switches, except those that are spring loaded to return to
the operate position.

7.1.3.2.6 Summary of Protective Action

Actions initiated by the engineered safeguards actuation system are tabulated
in Table 7-2. The devices actuated by the engineered safeguards actuation sys-
4. tem are listed in Table 7-3. Channels indicated may be referred to applicable
systems as shown in Figure 7-3.

Table 7-2
Engineered Safeguards Actuation Conditions

Channel No.	Action	Trip Condition	Steady State Normal Value	Trip Point
1, 2	High-Pressure Injection	Low Reactor Coolant Pressure or	2,120-2,250 psig	1,500 psig
		High Reactor Building Pressure	Atmospheric	4 psig
3, 4	Low-Pressure Injection	Very Low Reactor Coolant Pressure or	2,120-2,250 psig	500 psig
		High Reactor Building Pressure	Atmospheric	4 psig
5, 6	Start Reactor Building Cooling & Reactor Building Isolation	High Reactor Building Pressure	Atmospheric	4 psig
7, 8	Reactor Building Spray	High Reactor Building Pressure	Atmospheric	10 psig

Table 7-3
Engineered Safeguards Actuated Devices

Channel 1	Channel 2	Channels 1&2	Channel 3	Channel 4	Channels 3&4
HP-P1A	HP-P1C	HP-P1B	LP-P1A	LP-P1B	LPSW-P1C (3)
HP-24	HP-25		LP-17	LP-18	
HP-26	HP-27		LP-21 (Also chan 7)	LP-22 (Also chan 8)	
HP-3	HP-5		LPSW-4	LPSW-5	
HP-4	HP-21		LPSW-P1A (1)	LPSW-P1B (2)	
HP-20	KEOWEE START (Channel B)				
KEOWEE START (Channel A)	LOAD SHED & STBY. BKR. 2				
LOAD SHED & STBY. BKR. 1	Standby BUS FEED BKR.2				
Standby BUS FEED BKR.1					

Channel 5	Channel 6	Channels 5&6	Channel 7	Channel 8
RC-5	RC-7	LPSW-15	BS-1	BS-2
RC-6	FDW-106	LPSW-6	LP-21 (Also chan 3)	LP-22 (Also chan 4)
FDW-105	FDW-108	LPSW-21	BS-P1A	BS-P1B
FDW-107	CC-8	RBCU-F1B		
CC-7	LPSW-24			
LPSW-18	RBCU-F1C			
RBCU-F1A	GWD-13			
GWD-12	LWD-2			
LWD-1	CS-6			
CS-5	PR-2			
PR-1	PR-3			
PR-6	PR-4			
PR-E1A	PR-5			
PR-7	PR-E1B			
PR-9	PR-8			
	PR-10			
	FDW-103			
	FDW-104			

(1) LPSW-P1C for Unit 2
LPSW-P3B for Unit 3
(2) LPSW-P1B for Unit 2
LPSW-P3A for Unit 3
(3) LPSW-P1A for Unit 2

Rev. 4. 4/20/70
Rev. 6. 6/22/70
Rev. 16. 7/30/71
Rev. 18. 3/10/72

7.1.3.3 System Evaluation

The ESPS is a basic three-channel redundant system employing 2-out-of-3 coincidence between measured variables.

The system will tolerate the failure of one of three variables among either the reactor coolant pressure measurements or reactor building pressure measurements without losing its ability to perform its intended functions.

The high and low pressure injection systems are actuated by either reactor coolant pressure or reactor building pressure, thus providing diversity in actuation. The system will tolerate single or multiple failures within one protective channel without affecting the operation of other protective channels. This is the result of keeping each of the protective channel logics independent of every other protective channel. The independence is carried through the protective channel logic and up to the final actuating CR control relay. This is best illustrated by considering the actuation arrangement for the high pressure injection pumps (Figure 7-3).

There are three high pressure injection system pumps which operate in the event of an accident. HP-P1A is under the control of protective channel 1, HP-P1C is under the control of protective channel 2, while HP-P1B is under the control of both channels. Within the motor controller of HP-P1A and HP-P1C there is a single CR control relay controlled by the R_O relay in the pump's associated Test and Block module. The operation of the protective channel logic, the R_O relay in relation to the CR relay, was described previously. Should any two of the three reactor coolant pressure variables drop below their bistable set point, both protective channel 1 and 2 logics will trip, energizing the appropriate CR relays, and start the pumps.

Within the motor controller of HP-P1B there are two independent CR relays, each controlled by separate R_O relays in separate Test and Block modules, one in channel 1 and one in channel 2. The arrangement is identical to the way a channel would control any device since all elements are independent and duplicated through the CR relay. The only common element is the power source for the CR relays which is common to the motor controller. Loss of this power prevents the motor control from operating as well as the pump. Relays that monitor actuator coils for each motor or valve control detect either an open coil or a loss of control power.

The example just presented shows the independence and redundancy of the system. There is redundancy of sensors, logic, and equipment. The redundancy is preserved and kept effective by independence of sensors, instrument strings, logic, and control elements in the final actuator. These characteristics enable the system to tolerate single failures at all levels.

The system protective devices (pumps, valves, etc.) require electrical power in order to operate and perform their functions. The power for operating the CR relays is taken from the power source of the associated device. Loss of power to a CR relay or device does not impair the system functions since there is a second redundant device for each required function. The power for the R_O relays, logic, and instruments is taken from the plant's system of battery backed vital buses since loss of power at this level could affect the performance capability of the system. The system will tolerate the loss of one vital bus without loss of protective capability.

7.1.3.3.1 Redundancy and Diversity

The system as evaluated above is shown to have sufficient diversity and redundancy to withstand single failures at every level.

7.1.3.3.2 Electrical Isolation

The use of isolation amplifiers will effectively prevent any faults (shorts, grounds, or cross connection of signals), on any analog signal leaving the system from being reflected into or propagating through the system. The direct connection of any analog signal to a source of electrical power can, at worst, negate information from the measured variable involved. The use of individual R_O relays for each controlled device effectively preserves the isolation of each device, and of elements of one protective channel from another. Faults in the control wiring between an R_O relay and its CR relay in the controller of a protective device will not affect any other device or protective channel action.

Separation of redundant Engineered Safeguards (EG) functions is accomplished
by assigning the eight actuation channels (Table 7-2) to three groups. Isola-
4. tion for power, control, equipment location, and cable routing is maintained
throughout. Channels 1, 3, 5 and 7 are assigned to one group (odd actuation
channels). Channels 2, 4, 6 and 8 are assigned to a second group (even actua-
tion channels). Equipment which is actuated by both the even and odd actuation
channels is assigned to a third group. All equipment required to per-
form a specific ES function is assigned to the same group. For example, a
pump motor and all valves required for that pump to perform its function are
assigned to the same group.

For Unit 1, AC power for equipment controlled by the odd numbered actuation
channels is supplied from switchgear group 1TC (4KV), motor control center 1XS1
(600 and 208 volts), actuation power from vital bus A, and DC control power
from DC panel A. ES functions which are redundant to those controlled by the
4. odd numbered actuation channels are controlled by the even numbered actuation
channels. AC power for this equipment in Unit 1 is supplied from switchgear
group 1TD (4KV), motor control center 1XS2 (600 and 208 volts), from vital bus
B, and DC control power from DC panel B. Where a third unit of ESG equipment
is used to provide additional redundancy, it is actuated by both the even and
odd actuation channels. AC power for this equipment in Unit 1 is supplied from
switchgear groups 1TE or 2TC (4KV), motor control center 1XS3 (600 and 208 volts),
actuation power from either vital bus A for odd channel actuation or vital bus B
for even channel actuation, and DC power from DC panel C. Similar arrangements
are employed for ES equipment in Units 2 and 3 with different power and control
sources for each unit. These are described in Section 8.

7.1.3.3.3 Physical Isolation

The arrangement of modules within the system cabinets is designed to reduce the
chance of physical events impairing system operation. Control wiring between
4. the UC modules and the final actuating devices is physically separated and pro-
tected against damage which could impair system operation.

Separation between redundant channels of equipment, control cables, and power
cables provides independence of redundant ES functions. Power and control
cables for each group of ES equipment are routed in cable trays that contain
4. no cable for redundant equipment. Cables for reactor building cooling units
enter each reactor building through three separate penetrations located at
least 25 feet apart, and are routed in three different directions to the cooling
units. The only other ESG equipment located inside the reactor buildings are
electric motor operated isolation valves which are all common to the odd numbered
actuation group discussed above.

7.1.3.3.4 Periodic Testing and Reliability

The number of elements which can fail in a single instrument string is small as the system coincidence logic is not complex. The redundancy of the logic and the division of protective devices between logics forms a system having two parallel protective channels either of which is capable of performing the required functions. These characteristics are basic to an inherently reliable system. Logic elements are relays which have been selected for reliability and subjected to confirming tests under loads identical to those encountered in the system. The resultant calculated probability of logic failure is several orders of magnitude less than the known or estimated probability of failure of all other system elements.

The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the isolation amplifiers.

When an analog instrument string is placed in test, all associated analog sub-
4. system outputs go to the trip state. This assures that protective action can-
not be defeated by placing analog instrument strings in test.

To avoid a full protective channel or system trip, the logic is tested in parts,
one element at a time. The continuity of the electrical connections from the
output of the coincidence logic up to each R_o relay is tested by means of the
4. LT and UC modules. A LT module can neither prevent a trip of the associated
protective channel when protective action is called for nor initiate a trip of
the associated protective channel.

An individual protective device may be actuated by means of the UC module manual
switch. Operating this switch energizes the R_o relay as if the protective channel
4. has tripped actuating the associated final device. The module lamp confirms that
the module test relay returned to its normal state upon release of the manual
switch.

On-line checks of the system will confirm the normal state of the system, principally by comparative readings of similar analog indications between redundant measurements, and by the status lamps on bistables and logic modules.

The set points of the pressure switches used for ESPS channels 7 and 8 may be
checked by connecting a source of pressure and a pressure gauge to the pressure
4. transmitter connections provided inside the reactor building. This check may be
made regardless of reactor power when access to the building is attained. The
design provides access for this check at all Reactor power levels.

7.1.3.3.5 Manual Trip

A manual trip switch is provided in each Engineered Safeguards Protective
System channel. There are eight manual trip pushbuttons on the control
6. console, one for each protective channel.

7.1.3.3.6 Bypassing

The trip functions of the high and low pressure injection systems are by-
passed whenever the reactor is to be depressurized below the trip point of
the systems. Bypassing must be initiated manually within a fixed pressure
band above the protective system trip point. The high pressure injection
6. system may be bypassed only when the reactor pressure is 1,750 psi or less,
and the low pressure injection system may be bypassed only when the reactor
6. pressure is 900 psi or less. The bypass is automatically removed when the
reactor pressure exceeds the 1,750 and 900 psi values. This is in accordance
with IEEE 279, Section 4.12. The removal set points are above the
trip points in order to obtain a pressure band in which the trips may be by-
passed during a normal cooldown. The bypasses do not prevent actuation of the
4. HP and LP injection systems on high reactor building pressure. Bypassing is
under administrative control. Since the ESPS incorporates triple redundancy
in its analog input subsystems, there are three HP injection bypass switches
and three LP injection bypass switches. Two of the three switches must be
operated to initiate a bypass. Once a bypass has been initiated, the condition
is indicated by the plant annunciator and by lamps associated with the bypass
switches. The switches are backlighted. No provisions are made for manual re-
moval of a bypass before its automatic removal set point is reached.

7.2 REGULATION SYSTEMS

7.2.1 DESIGN BASES

Reactor output is regulated by the use of movable control rod assemblies and soluble boron dissolved in the coolant. Control of relatively fast reactivity effects,including Doppler, xenon, and moderator temperature effects, is accomplished by the control rods. The control response speed is designed to overcome these reactivity effects. Relatively slow reactivity effects, such as fuel burnup, fission product buildup, samarium buildup, and hot-to-cold moderator reactivity deficit, are controlled by soluble boron.

Control rods are normally used for control of xenon transients associated with normal reactor power changes. Chemical shim shall be used in conjunction with control rods to compensate for equilibrium xenon conditions. Reactivity control may be exchanged between rods and soluble boron consistent with limitations on power peaking.

Reactor regulation is a composite function of the Integrated Control System and Rod Drive Control System. Design data for these subsystems are given in the following sections.

7.2.2 ROD DRIVE CONTROL SYSTEM

The rod drive control system (RDC) includes drive controls, power supplies, position indicators, operating panels and indicators, safety devices, and enclosures.

7.2.2.1 Design Basis

The rod drive control system design bases are catagorized into safety considerations, reactivity rate limits, startup considerations, and operational considerations.

7.2.2.1.1 Safety Considerations

a. The control rod assemblies (CRA) are inserted into the core upon receipt of protective system trip signals. Trip command has priority over all other commands.

b. No single failure shall inhibit the protective action of the rod drive control system.

7.2.2.1.2 Reactivity Rate Limits

The speed of the mechanism and group rod worth provide the reactivity change rates required. For design purposes the maximum rate of change of reactivity that can be inserted by any group of rods has been set at 1.1 x 10^{-4} $\Delta K/K/s$. The drive controls, i.e., the drive mechanism and rods combination, have an inherent speed-limiting feature.

4. Speed-limiting is accomplished through the use of 60 Hz synchronous programmer motors. These motors are powered, through transformers, from the same 600 V AC source as the remainder of the CRD system. Thus, the speed of rod motion is locked to the plant's AC power frequency which, in turn, is limited to 64 HZ maximum as controlled by the plant and system frequency control system. At 64 Hz, the speed of rod motion is (64/60) x 30 = 32 in./minute.

7.2.2.1.3 Startup Considerations

The rod drive control system design bases for startup are as follows:

a. Reactor regulation during startup is a manual operation.

b. Control rod "out" motion is inhibited when a high startup rate (short) period) in the source range or intermediate range is detected.

7.2.2.1.4 Operational Considerations

For operation of the reactor, functional criteria related to the rod drive control system are:

a. <u>CRA Positioning</u>

The rod drive control system provides for controlled withdrawal, controlled insertion, and holding of the control rod assemblies (CRAs) to establish and maintain the power level required for a given reactor coolant boron concentration.

b. <u>Position Indication</u>

Continuous rod position indication, as well as full-in and full-out position indication, are provided for each control rod drive.

c. <u>System Monitoring</u>

The rod drive control system design includes provisions for routinely monitoring conditions that are important to safety and reliability.

7.2.2.2 <u>System Design</u>

The rod drive control system provides for withdrawal and insertion of the control rod assemblies to maintain the desired reactor output. This is achieved either through automatic control by the Integrated Control System discussed in Section 7.2.3, or through manual control by the operator. As noted previously, this control compensates for short term reactivity changes. It is achieved through the positioning in the core of sixty-one control rod assemblies and eight axial power-shaping rod assemblies. The sixty-one rods are grouped for control and safety purposes into seven groups. Four groups function as safety rods, and three groups serve as regulating rods. An eighth group serves to regulate axial power peaking due to xenon poisoning. Seven of the eight groups may be assigned from four to twelve control rod assemblies. Eight rod assemblies are used in group eight.

Control rods are arranged into groups at the control rod drive control system patch panel. Typically, twenty-eight rods, including the axial power shaping rods, are assigned to the regulating groups, and forty-one rods are assigned to the safety rod groups. A typical rod grouping arrangement is shown below:

Safety Rods	Regulating Rods	Axial Power Shaping Rods
Group 1 - 8	Group 5 - 12	Group 8 - 8
Group 2 - 12	Group 6 - 4	
Group 3 - 9	Group 7 - 4	
Group 4 - 12		

During startup the safety rod groups are withdrawn first, enabling withdrawal of the regulating control groups. The sequence allows operation of only one regulating rod group at a time except where reactivity insertion rates are low (first and last 25% of stroke), at which time two adjacent groups are operated simultaneously in overlapped fashion. These insertion rates are shown in Figure 7-5.

As fuel is used, dilution of soluble boron in the reactor coolant is necessary. When Group 6 is more than 95% withdrawn, interlocks permit dilution. The reactor controls insert Group 6 to compensate for the reduction in boron concentration by dilution. The dilution is automatically terminated by a pre-set volume measuring device. Interlocks are also provided on Group 6 rod position to terminate dilution at a pre-set insert limit (9.1.2.7).

7.2.2.2.1 System Equipment

The rod drive control system consists of three basic components: (1) control rod drive motor power supplies; (2) system logic; (3) trip breakers. The power supplies consist of four group power supplies, an auxiliary power supply, and two holding power supplies. The group power supplies are of a redundant six-phase half-wave rectifier design. In each half of a group power supply, rectification and switching of power is accomplished through the use of Silicon Controlled Rectifiers (SCRs). This switching sequentially energizes first two, then three, then two of the six CRA motor stator windings in stepping motor fashion, to produce a rotating magnetic field for the control rod assembly motor to position the CRA. Switching is achieved by gating the six SCRs on for the period each winding must be energized. As each of the six windings utilize SCRs to supply power, six gating signals are required.

Gating signals for the group power supplies are generated by a motor driven programmer consisting of a 60-cycle, reversible synchronous motor driving a multichannel photo-optic encoder. The coded light beam excites photo-detectors, generating signals which are amplified to form the Silicon Controlled Rectifier gating signals. The programmer is redundant (except for motor and gears) thus providing separate but synchronized gating signals to the dual power supply units. Command signals to position the control rod drive are introduced at the programmer motor.

Identical power supplies are used for the regulating (control) groups and for the auxiliary power supply. Each half of each group power supply is capable of driving up to 12 drive mechanisms--the maximum number that may be in any one group. The power supplies have dual power inputs, each half fed from separate power sources and each half being capable of carrying the full load.

Unlike the control group power supplies, the holding power supply is used to maintain the safety rods fully withdrawn; consequently, switching is not required. A six-phase d-c power supply is used for this purpose. Two holding power supplies are provided. Each is rated to furnish power to one winding of 48 mechanisms; normal load would be 41 coils for each power supply.

The auxiliary power supply is used to position the safety rod groups and to provide single rod control. The safety rod groups are maneuvered with the auxiliary power supply, and then, when fully positioned, are transferred to the holding busses described above. After positioning the safety rods, the auxiliary power supply is available to the regulating roups, through transfer relays, to serve either as a single rod controller, should repositioning of a single rod be necessary, or, as a spare group controller, should one of the
5. group control power supplies require maintenance. The auxiliary power supply cannot be used to control more than one group at one time.

The system logic encompasses those functions which command control rod motion in the manual or automatic modes of operation, including CRD sequencing, safety and protection features, and the manual trip function. Major components of the logic system are the Operator's Control Panel, CRA position indication panels, automatic sequencer, and relay logic.

Switches are provided at the operators control panel for selection of the desired rod control mode. Control modes are: (1) Automatic mode -- where rod motion is commanded by the Integrated Control System; and (2) Manual mode -- where rod motion is commanded by the operator. Manual control permits operation of a single rod or a group of rods. Alarm lamps on the RDC panel alert the operator
5. to the systems status at all times. The group 8 control rods can only be controlled manually even when the remainder of the system is in automatic control.

The sequence section of the logic system utilizes rod position signals to generate control interlocks which regulate rod group withdrawal and insertion. The sequencer operates in both automatic and manual modes of reactor control, and controls the regulating groups only. Analog position signals are generated by the reed switch matrix on the CRA, and an average group position is generated by an averaging network. This average signal serves as an input to electronic trip units which are activated at approximately 25 and at 75 per cent of group rod withdrawal. Two bistable units are provided for each regulating group. Outputs of these bistables actuate "enable" relays which permit the rod groups to be commanded in automatic or manual mode.

The automatic sequencer circuit can control only rod groups 5, 6 and 7. The safety rod groups, groups 1-4, are controlled manually, one group at a time. In addition, the operator must select the safety group to be controlled and
5. transfer it to the auxiliary power supply before control is possible. There is no way in which the automatic sequencer can affect the operations required to move the safety rods.

In addition to the sequencer, relay logic monitors are provided in the "enable" circuits which prohibit out of sequence conditions.
3. The selection of manual control mode and sequence bypass mode functions permit intentional out-of-sequence conditions. This condition is indicated to the operator.

"Sequence" operation may be bypassed at any time if the manual control mode has been selected. If automatic control is selected, "sequence" operation cannot be bypassed.

5. "Sequence bypass" operation permits selection of any rod group or any single rod for control. It will not permit selection of more than one rod group at any given time. Motion of more than one group at any given time is also not possible when this operation is selected.

Inputs to the system logic from the Reactor Protection System and the Integrated Control System provide interlock control over rod motion. These interlocks cause rod motion command lines and control mode selection to be inhibited.

Under certain conditions the Nuclear Instrumentation generates an "out inhibit" signal. When this signal is received by the CRD System, all out command circuits are disabled, thus preventing withdrawal of all rods in either automatic or
5. manual control.

Automatic insertion of rods can only be commanded by the ICS when the CRD system is in the automatic mode. These commands can only affect rod groups 5, 6 and 7.

In the rod drive control system, two methods of position indication are provided: an absolute position indicator and a relative position indicator. Either position signal is available to the control board indicator through a selector switch. The absolute position transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the motor tube extension. Each switch is hermetically sealed. Switch contacts close when a permanent magnet mounted on the upper end of the lead screw extension comes in close proximity.

As the lead screw (and the control rod assembly) moves, the switches operate sequentially, producing an analog voltage proportional to position. The accuracy of the analog signal is approximately ± 1.1 per cent of full scale (130 in.) and the readout has approximately ± 2.1 per cent of full scale accuracy. Other reed switches included in the same tube with the position indicator matrix provide full-in and full-out limit indications.

The relative position transducer is a small pulse-stepping motor, driven from the power supply for the rod drive motor. This small motor is coupled to a potentiometer with an output signal accuracy of ± 0.7 per cent of full scale position, producing a readout with an accuracy of ± 1.7 per cent of full scale.

Rod drive control system trip breakers are provided to interrupt power to the control rod drive mechanisms. When power is removed, the roller nuts disengage from the lead screw allowing a gravity trip of the CRA.

The group 8 drive mechanisms are modified to prevent rod drop into the core
when power is removed from the stators. In this type of mechanism, the roller
nuts are mechanically restrained to remain engaged with the lead screw at all
5. times. Thus, the mechanical "trip" action has been removed from these ASPR's,
and they remain at the position they occupied immediately before trip was
initiated. When a reactor trip is initiated, power to the group 8 power supply
is interrupted in the same manner as for the other regulating power supplies.
Two series trip methods are provided for removal of power to the CRD mechanisms.
First, a trip is initiated when Reactor Protection System logic interrupts
power to the undervoltage (UV) coil of the main AC feeder breakers. Secondly,
a trip is initiated when the Silicon Control Rectifier gating power and the
DC holding power is interrupted. As parallel power feeds are provided on both
holding and gating power, interruption of both feeds is required for trip
action in either method of trip. Trip circuitry is shown in Figures 7-6 and 7-7.

AC power feed breakers are of the three-pole, stored-energy type and are equipped with instantaneous undervoltage trip coils. Each AC feed breaker is housed in a separate metal clad enclosure. The secondary trip breakers are also of the stored-energy type with two parallel-connected instantaneous undervoltage trip coils consisting of two 2-pole breakers mechanically ganged to interrupt DC busses. All breakers are motor-driven-reset to provide remote reset capability. Each undervoltage trip coil is operated from the Reactor Protection System.

7.2.2.3 System Evaluation

7.2.2.3.1 Safety Considerations

A reactor trip occurs whenever power has been removed from the rod drive motors. The design provides two stored energy breakers which do not require power to interrupt the electrical feeds to rod drive control power supplies and a second set of circuit-interrupting devices in series on the output of the power supplies. All devices have interrupting capacity of sufficient rating to open under any group load configuration. Reactor trip is further assured by providing series trip devices, split buses, and provisions for periodic testing. Trip redundancy is provided by series breakers while availability and testability are provided through dual power sources. Redundant power supplies permit testing of the trip action of each power-interrupting device without loss of plant availability.

Reactivity shutdown margin provided by the safety rods is assured by diversification of their power buses. This feature, as shown in Figure 7-1, utilizes four separate buses, each having a separate trip device, to power the safety rods. A failure in one bus does not reflect into the other buses, therefore, a single failure in the distribution system for the safety rods does not prevent a plant shutdown.

The minimum voltage required to hold a drive in a withdrawn position is 42 volt DC per coil (2 coil "hold" mode). The probability of an external DC source being applied to the control rod drive mechanisms downstream from the reactor trip points such that the CRA's are held in their withdrawn positions after a trip is not considered credible for the following reasons:

1. The secondary trip devices in the Control Rod Drive System remove all DC power from the drives.

2. Control rod drive power cables are terminated at only three points between the control rod drive cabinets and the drive mechanisms.

   Two of these terminations are made outside and inside the reactor
   5. building electrical penetrations inside junction boxes containing only control rod drive power cables. The third termination is made in bulkhead connectors (one per drive) in the area of the reactor. The only other cables terminated in this area are the control rod drive instrumentation cables. The instrumentation cables are terminated in bulkhead connectors of a different size and configuration, therefore mismating of connectors could not be accomplished.

3. No cable splices are permitted between termination points described.

4. DC systems from the batteries at Oconee are not grounded and are equipped with ground detecting circuitry.

In summary, series redundant trip devices having adequate rating, testability and a "split bus" arrangement insure safety of reactor trip circuits.

7.2.2.3.2 Reactivity Rate Limits

The desired rate of change of CRA reactivity insertion and uniform reactivity distribution over the core are provided for by the control rod drive and power
3. supply design, and the selection of rods in a group. The CRA motor, lead screw,
and power supply designs are fixed to provide a uniform rate of speed of 30 in./min.
This speed is determined by the programmer, which is driven by a synchronous motor;
3. the motor rotates at a speed fixed to the 60 hz. AC line frequency. The reactivity
change is then controlled by the rod group size. To insure flexibility in this area, a patch panel has been included in the power supply to enable the interchange of rod worth between rod groups. Any rod may be patched into any group with the exception of Group 8.

Uniform and symmetrical group insertion rate is provided for by synchronous withdrawal of all rods in that group. Such synchronous withdrawal is achieved by the design of the power supply. A group power supply operates synchronously by having its load (4 to 12 CRA motor windings) connected in parallel on the output of the SCR's. As the programmer gates on the SCR's, all rods in a group have the same motor winding energized simultaneously producing synchronous motion of the entire group.

5. Each control rod is provided with a rod position indication monitor (7.2.2.3.4) to
sense asymmetric rod patterns by comparing the individual rod position with its
3. group average position. When the rod moves out-of-step from its group by a preset
amount, the monitor alarms the condition to the operator, computer and the ICS. Depending on the power setting and the control mode, action is initiated by the ICS to insert rods and reduce power.

7.2.2.3.3 Startup Considerations

The rod drive controls receive interlock signals from the ICS and nuclear in-
19. strumentation (NI). These inputs are used to inhibit automatic mode selection
when a large error exists in the ICS reactor control subsystem and to inhibit out motion for high startup rates, respectively.

In addition to the startup considerations, dilution controls, to permit removal of reactor shutdown concentrations of boron in the reactor coolant, are provided. This control bypasses the normal reactor coolant dilution controls, described in Section 7.2.2.2, providing all safety rods are withdrawn from the core and the operator initiates a continuous feed and bleed cycle. An additional interlock on rod Group 5 inhibits the use of this circuit when rod Group 5 is more than 80% withdrawn.

7.2.2.3.4 Operational Considerations

The control rod assembly positioning system provides the ability to move any rod to any position required consistent with reactor safety. As noted in Section 7.2.2.3.2 a uniform speed is provided by the drive system. A fixed rod position when motion is not required is obtained by the power supply ability to energize two adjacent windings of the CRA motor stator. This static energizing of the windings maintain a latched stator and fixed rod position.

Position Indication

As previously described, two separate position indication signals are provided. The absolute position sensing system produces signals proportional to CRA position from the reed switch matrix located on each CRA mechanism. The relative position indication system produces a signal proportional to the number of CRA motor power pulses from a stepping motor and precision potentiometer for each CRA mechanism.

Position indicating readout devices mounted on the operator's console consist of 69 single rod position meters and 4 control group average position meters. Accuracy of all meters is to $\pm 1\%$ of full scale. The operation of a selector switch permits either relative or absolute position information to be displayed on the single rod meters.

The control-group-average meters display the arithmetic average of the relative position signals of all CRA's in a group. A selector switch on the operator's console permits the group meters to display either the positions of all safety rod groups (Groups 1-4) or the positions of all regulating rod groups (Groups 5-7) and the axial power shaping rod groups (Group 8).

Indicator lights are provided on the single-rod meter panel to indicate when each rod is; (1) fully inserted, (2) fully withdrawn, (3) under control and (4) whether a fault is present. Indicators on the operator's console show full insertion, full withdrawal, under-control and fault indication for each of the eight control rod groups.

Failures which could result in unplanned control rod withdrawal are continuously monitored by fault detection circuits. When failures are detected, indicator lights and alarms alert the operator. Fault indicator lights remain on until the fault condition is cleared by the operator. A list of indicated faults is shown below:

(1) Asymmetric rod patterns (indicator and alarm).
(2) Motor rotation faults (indicator only).
3. (3) Sequence faults (indicator and alarm).
(4) Trip faults (indicator and alarm).
(5) Safety rods not withdrawn (indicator only).
(6) Programmer lamp faults (indicator only).

Faults serious enough to warrant immediate action produce automatic correction commands from the fault detection circuits, and manual bypass is not possible. Status indicators on the operator's console provide monitoring of control modes.

A description of each fault detector follows:

4. (1) <u>Asymmetric Rod Monitor</u>

a. Design Basis - To detect and alarm if any rod deviates from its group reference position by more than a maximum of 9 inches true position.

b. Circuit Operation - There are 69 asymmetric rod pattern monitors, one assigned to each control rod. These monitors continuously compare the individual rod absolute position signal with the absolute group reference (average) signal. The absolute value of the difference between the two signals is computed, and if this difference is less than the maximum value set by the circuit calibration, no output results. If, however, the difference is greater than the setpoint, a relay is actuated which alarms the asymmetric condition. Two alarm channels are provided in each monitor which are identical except for the setpoints. One setpoint is calibrated for a 3-inch signal differential (maximum 7-inch true position separation) and initiates an alarm. The other setpoint is at 5-inch signal differential (maximum 9-inch true position separation) and initiates the action described below.

c. Corrective Action - Action taken upon detection of an asymmetric rod fault depends upon the control mode and the power level in effect at the time the fault is detected. Corrective action is the same for any asymmetric condition including "stuck-in", "stuck-out", or dropped control rods.

Detection of a 3-inch signal differential is defined as an "asymmetric rods alarm". Actuation of this alarm causes the fault indicator lamp for that rod to be energized and an alarm signal to be sent to the plant computer and annunciator.

If the condition is not corrected and the separation increases to a 5-inch signal difference, the following actions occur:

(1) "Asymmetric fault" lamp on the operator's console is energized. If operation is in the manual control mode, operator action is required by administrative control.

(2) If operation is in the automatic mode, a "runback fault" signal is sent to the Integrated Control System. The ICS will impose a maximum reactor power level of 60% of rated power if power is initially less than 60%.

When reactor power is greater than 60% of rated power, the Control Rod Drive System generates an "Out Inhibit" signal which disables the "Out" command circuits to all drives and the ICS initiates a runback to 60% reactor power. "Out Inhibit" alarms are sent to the ICS, plant annunciator and plant computer.

Reactor power remains limited to 60% maximum in automatic control until the fault is corrected.

(2) Motor Rotation Fault Detector

a. Design Basis - To detect and prevent unwarranted "out" motion of control rods caused by a failure such that "out" motion results from an "in" command.

b. Circuit Operation - Each of the five programmers is equipped with a rotation fault detector. These circuits consist of a rotation direction sensor and a command versus rotation comparator. The rotation sensor is mechanically coupled to the programmer and produces an output signal which reflects the direction of rotation of the programmer. The comparator compares the rotation with the actual command. If actual rotation is in the "out" direction but command is for "in" motion, or, if "out" rotation occurs when there is no command, the comparator actuates relays which interrupt all command lines to the programmer; at the same time a "d-c brake" voltage is applied to the programmer motor to prevent coasting.

c. Corrective Action - If a programmer motor rotation fault is detected when operation is in the automatic control mode, the automatic mode disengages and an alarm lamp alerts the operator to the malfunction. Control reverts to the manual mode and remains in manual until the fault is cleared and the system is reset by the operator.

(3) Sequence Monitor

a. Design Basis - To detect any motion of the regulating rod groups outside of the predetermined automatic sequence patterns, and to prevent further automatic motion when such conditions occur.

b. Circuit Operation - The sequence monitor continuously compares the group average (reference) signals for each regulating rod group with the allowable sequence patterns. Bistable amplifiers and digital logic are used for this purpose. In addition, the rod group "enable" circuits are monitored to determine if a group is enabled for motion out-of-turn. The safety rod groups' out limit signals serve as a permissive to automatic sequencing: the sequence monitor prevents automatic control until the safety rods are fully withdrawn.

c. Corrective Action - When an out-of-sequence condition is detected and operation is in the automatic control mode, the automatic mode disengages and an alarm lamp alerts the operator to the malfunction. Control reverts to manual and remains in manual until the fault is corrected and the system is reset by the operator.

(4) Rod Position Sensor Faults

All rod position sensor faults lead to false asymmetric, stuck, or dropped rod symptoms which are acted upon by the Asymmetric Rod Monitor described in Item (1) above.

(5) Trip Fault Detector

a. Design Basis - To sense faults which may affect operation of the trip circuits, such as one trip breaker in the tripped position during normal operation.

b. Circuit Operation - The circuit contains elements which sense the state of each trip device as well as the state of each of the four trip channels. If the state of a device fails to agree with the state of its associated trip channel, a trip fault will be alarmed. Other independent circuits sense and confirm that a trip, if commanded, has actually occurred.

c. Corrective Action - Alarms are provided.

(6) Safety Rods Not Withdrawn

a. Design Basis - To prevent, on plant startup, withdrawal of the regulating rods until the safety rods are fully withdrawn.

b. Circuit Operation - The circuit continuously monitors the group "out" limit for the four safety rod groups. When the four groups are all
4. fully withdrawn, signals are sent to the sequencer and the sequence monitor which permit automatic control.

c. Corrective Action - Alarms are provided.

(7) Programmer Lamp Fault Detector

a. Design Basis - To detect failures in the programmer lamp circuits.

b. Circuit Operation - Each set of programmer lamps is provided with a circuit which monitors the current flow to the lamps. When abnormal conditions exist, an alarm is actuated.

c. Corrective Action - Alarms are provided.

7.2.3 INTEGRATED CONTROL SYSTEM

7.2.3.1 Design Basis

The Integrated Control System provides the proper coordination of the reactor, steam generator feedwater control, and turbine under all operating conditions. Proper coordination consists of producing the best load response to the unit load demand while recognizing the capabilities and limitations of the reactor, steam generator feedwater system, and turbine. When all single portion of the plant is at an operating limit or control selection is on manual, the Integrated Control System design uses the limited or manual section as a load reference.

The Integrated Control System maintains constant average reactor coolant temperature between 15 and 100 per cent rated power and constant steam pressure at all loads. Optimum unit performance is maintained by limiting steam pressure variations; by limiting the unbalance between the steam generator, turbine, and the reactor; and by limiting the total unit load demand upon loss of capability of the steam generator feed system, the reactor, or the turbine generator. The control system provides limiting actions to assure proper relationships between the generated load, turbine valves, feedwater flow, and reactor power.

The response of the reactor coolant system to increasing and decreasing power transients is limited by the Integrated Control System as indicated in Table
4. 7-4.

4. Table 7-4

ICS Transient Limits

Transient	Power Range (% Full Power)	Ramp Input Limit (% Power/min)	Step Input Limit (% Power)
Power Increase	20 - 90	10	10
	15 - 20	5	0
	90 - 100	5	0
Power Decrease	100 - 20	10	10
	20 - 15	5	0

The turbine bypass system permits a load drop of 40 per cent or a turbine trip from 40 per cent load without safety valve operation. The turbine bypass system and safety valves permit a 100 per cent load drop without reactor trip.

7.2.3.2 System Design

7.2.3.2.1 General Description

The Integrated Control System includes four independent subsystems as shown in Figure 7-8. The four subsystems are: the Unit Load Demand; the Integrated Master; the Steam Generator; and the Reactor. The system philosophy is that control of the plant is achieved through feed-forward control from the Unit Load Demand. The Unit Load Demand produces demands for parallel control of the turbine, reactor and steam generators feedwater system through respective subsystems.

The Steam Generator Control is capable of automatic or manual feedwater control from startup to full output. The Integrated Master Control is capable of automatic or manual turbine valve control from minimum turbine load to full output, and of manual control below minimum turbine load. The Reactor Control is designed for automatic or manual operation above 15 per cent output, and for manual operation below 15 per cent.

The basic function of the Integrated Control System is matching megawatt generation to unit load demand. The Integrated Control System does this by coordinating the steam flow to the turbine with the rate of steam generation. To accomplish this efficiently, the following basic reactor/steam-generator requirements are satisfied:

a. The ratios of feedwater flow and Btu input to the steam generator are balanced as required to obtain desired steam conditions.

b. Btu input and feedwater flow are controlled:

1. To compensate for changes in fluid and energy inventory requirements at each load.

2. To compensate for temporary deviations in feedwater temperature resulting from load change, feedwater heating system upsets or final steam pressure changes.

7.2.3.2.2 Unit Load Demand

The Unit Load Demand (ULD) is designed to accomplish three objectives related to the operation of the plant. First, the ULD conditions the load demand signal from the system dispatcher to make it compatible with the power level of the plant and its ability to change load. Second, the ULD permits the operator to separate the plant from the dispatch system and manually establish the power output. Third, the ULD initiates load limiting and runback functions to restrict operation within prescribed limits. Figure 7-9 illustrates the functions incorporated in the subsystem.

The Unit Load Demand obtains a load demand signal from the system dispatcher, the plant computer, or the operator. The load demand is restrained by a maximum load limiter, a minimum load limiter, a rate limiter, and a runback limiter.

Rate limiting is designed as a function of load, so transients are limited as
4. shown in Table 7-4. A frequency loop is added to match the speed droop char-
acteristic of the turbine speed controls.

The limiter acts to runback and/or limit the load demand under any of the following conditions:

a. Loss of one or more reactor coolant pumps.

b. Total feedwater flow lags total feedwater demand, or reactor power lags reactor power demand, by more than 5 per cent.

c. Loss of one feedwater pump.

d. Asymmetric rod withdrawal patterns exists in reactor.

e. The generator separates from the 230 kV bus.

The output of the limiters is a megawatt demand signal which is applied to the turbine control, steam generator feedwater control, and reactor control in parallel.

The controlling subsystems of the ICS (turbine control, steam generator feedwater control,and reactor control) normally operate in the automatic mode in response to a demand signal from the ULD. The subsystems control function is kept within pre-established bounds under other than normal automatic operation by a "load tracking" feature built into the ICS. The system will switch to the load tracking mode if either of the following conditions exists:

a. One or more of the subsystems are in manual.

b. Errors greater than pre-set limits develop between the demand and the variable.

In this mode, the load demand is made to follow the manual or limited control subsystem. Load tracking continues until the limiting condition is brought back to within the pre-established deadband or the subsystem is returned to automatic operation.

7.2.3.2.3 Integrated Master

The Integrated Master has been designed to receive the megawatt demand signal from the Unit Load Demand Subsystem and convert this signal into a demand for the feedwater, turbine, and reactor control. A functional diagram of the Integrated Master Control is shown in Figure 7-10. The megawatt demand is compared with the generator megawatt output, and the resulting megawatt error signal is used to change the steam pressure set point. The turbine valves then change position to control steam pressure. As the megawatt error reduces to zero, the steam pressure set point is returned to the steady state value. By limiting the effect of megawatt error on the steam pressure set point, the system can be adjusted to permit controlled variations in steam pressure to achieve the desired rate of turbine response to megawatt demand.

Unit load demand is utilized as the feed-forward demand to the steam generator, reactor, and turbine while operating in the integrated control mode. This demand is compensated for deviations in the steam header pressure.

The turbine bypass system operates from the header pressure error or individual steam generator pressures as an overpressure relief for the turbine header. The turbine bypass system permits a load drop of 40 per cent, or a turbine trip from 40 per cent load without safety valve operation.

7.2.3.2.4 Steam Generator Control

Control of the steam generator is based on matching feedwater flow to megawatt demand with bias provided by the error between steam pressure set point and steam pressure. The pressure error increases the feedwater flow demand if the pressure is low. It decreases the feedwater flow demand if the pressure is high. Figure 7-11 illustrates the steam generator feedwater controls.

The basic control actions for parallel steam generator operation are:

a. Megawatt demand converted to feedwater demand.

b. Steam pressure compared to set pressure, and the pressure error converted to feedwater demand.

c. Total feedwater demand computed from sum of a and b.

d. Total feedwater flow demand split into feedwater flow demand for each steam generator.

e. Feedwater demand compared to feedwater flow for each steam generator. The resulting error signals position the feedwater flow controls to match feedwater flow to feedwater demand for each steam generator.

For operation below 15 per cent load, the steam generator control acts to maintain a preset minimum downcomer water level. The conversion to level control is automatic and is introduced into the feedwater control train through an auctioneer. At electrical loads below 15 per cent, the turbine bypass valves will operate to control steam pressure rise.

The steam generator control also provides ratio, limit, and runback actions as shown in Figure 7-11, which include:

a. Steam Generator Load Ratio Control

Under normal conditions the steam generators will each produce one-half of the total load. Steam generator load ratio control is provided to balance reactor inlet coolant temperatures during operation with more reactor coolant pumps in one loop than in the other.

b. Water Level Limits

A maximum water level limit prevents overpumping of feedwater and assures superheated steam under all operating conditions.

A minimum water limit is provided for 15 per cent low load control in the downcomer section.

c. Reactor Coolant Flow Limiters

These limiters restrict feedwater demand to match reactor coolant pumping capability. For example, if one reactor coolant pump is not operating, the maximum feedwater demand to the steam generator in the loop with the inoperative pump is limited to one-half normal.

d. Reactor Outlet and Feedwater Low Temperature Limits

These limiters reduce feedwater demand when the reactor outlet temperature or the feedwater temperature is low.

e. Feedwater Cross Limits

A feedwater demand signal is limited to maintain the feedwater demand always within 5 per cent of the reactor power.

Feedwater demand is limited to within 5 per cent of the reactor power demand both in the increase and decrease feedwater demand directions.

f. Steam Generator Pressure Limit

Individual steam generator pressure limits respective feedwater demands whenever pressure increases in the steam generators.

g. Auxiliary Feedwater

Upon loss of all reactor coolant pumps, and/or both feedwater pumps, the ICS positions valves to direct flow to the auxiliary feedwater header.

h. Feedwater Valve Control

Valve position demand for each steam generator is applied to both the startup and the main feedwater valves, through control stations. These valves are sequenced into operation so that the startup valve opens first (from zero to 15% load) followed by the main feedwater valve.

i. Feedwater Pump Control

Feedwater pump speed is controlled to maintain a constant differential pressure drop across feedwater valves.

7.2.3.2.5 Reactor Control

The reactor control is designed to maintain a constant average reactor coolant temperature over the load range from 15 to 100 per cent of rated power. The steam system operates on constant pressure at all loads. The average reactor coolant temperature decreases over the range from 15 per cent to zero load. Figure 7-12 shows the reactor coolant and steam temperatures and the steam pressure over the entire load range.

The reactor control consists of analog computing equipment with inputs of megawatt demand, core power, and reactor coolant average temperature. The output of the controller is an error signal that causes the control rod drive to be positioned until the error signal is within a deadband. A block diagram of the reactor control is shown on Figure 7-13.

First, reactor power level demand (N_d) is computed as a function of the megawatt demand (MW_d) and the reactor coolant system average temperature deviation ($\overline{\Delta T}$) from the set point, according to the following equation:

$$N_d = K_1 MW_d + K_2 (\overline{\Delta T} + \frac{1}{\tau} \int \overline{\Delta T}\, dt)$$

Megawatt demand is introduced as a part of the demand signal through a proportional unit having an adjustable gain factor (K_1). The temperature deviation is introduced as a part of the demand signal after proportional plus reset (integral) action is applied. For the temperature deviation, K_2 is the adjustable gain and τ is the adjustable integration factor.

The reactor power level demand (N_d) is then compared with the reactor power level signal (N_i), which is derived from the nuclear instrumentation. The resultant error signal ($N_d - N_i$) is the reactor power level error signal (E_N).

When the reactor power level error signal (E_N) exceeds the deadband settings, the control rod drive receives a command that withdraws or inserts rods depending upon the polarity of the power error signal.

The following additional features are provided with the reactor power controller:

a. A high limit on reactor power level demand (N_d).

b. An adjustable low limit on reactor power level demand (N_d).

c. A megawatt demand limit imposed by lack of feedwater flow capability from the steam generator controls.

The reactor controls incorporate automatic or manual rod control above 15 per cent of rated power and manual rod control below 15 per cent of rated power.

7.2.3.3 System Evaluation

7.2.3.3.1 System Failure Considerations

Redundant sensors for major system parameters are available to the Integrated Control System. The operator can select any of the redundant sensors from the control room. Manual reactivity control is available at all power levels. Loss of electrical power to automatic control stations reverts the control system to manual, ultimately placing ICS in "load tracking."

Maloperation or failure of the ICS or any of its subsystems places no automatic limitations on reactor operation because the ICS reverts to the manual mode, ultimately placing the ICS in "load tracking", i.e., following the actual generated load.
4.

Failure of the ICS does not diminish the safety of the reactor. None of the functions provided by the ICS are required for reactor protection or for actuation of the ESPS. The reactor protection criteria, used in the analysis of the accidents presented in Section 14, can be met irrespective of ICS action.

7.2.3.3.2 System Limits

Maximum and minimum limits on the reactor power level demand signal (N_d) prevent the automatic reactor controls from initiating undesired power excursions.

Maximum and minimum levels on the megawatt demand signal (MW_d) prevent the reactor controls from initiating undesired power excursions.

Cross limiting between the steam generators and the reactor prevents reactor power excursions that may result in a reactor trip from reactor coolant pressure or temperature.

7.2.3.3.3 Modes of Control

The Integrated Control System is designed to revert to a "Load Tracking" mode of control to tie the unit to the subsystem on manual or to the subsystem being limited. In the startup control mode the reactor is prevented from automatic rod withdrawal below 15 per cent power.

In startup control mode, the controls are arranged so that the steam system follows reactor power rather than turbine system power demand. The controls will limit steam bypass to the condenser when condenser vacuum is inadequate.

7.2.3.3.4 Loss-of-Load Considerations

The nuclear unit is designed to accept 10 per cent step load rejection without safety valve action or turbine bypass valve action. The combined actions of the control system and the turbine bypass valve permit a 40 per cent load reduction or a turbine trip from 40 per cent load without safety valve action. The controls will limit steam dump to the condenser when condenser vacuum is inadequate, in which case the safety valves may operate. The combined actions of the control system, the turbine bypass valve, and the safety valves permit separation from the external transmission system without reactor trip.

The features that permit continued operation under load rejection conditions include:

a. Integrated Control System

During normal operation the Integrated Control System controls the unit load in response to load demand from the system dispatch center or from the operator. During normal load changes and small frequency changes, turbine control is through the speed changer to maintain constant steam pressure.

During large load and frequency upsets, the turbine governor takes control to regulate frequency. For these upset conditions, frequency error at the input to the integrated control system becomes more important in providing load matching.

b. 100 Per Cent Relief Capacity in the Steam System

This provision acts to reduce the effect of large load drops on the reactor system.

Consider, for example, a sudden load rejection greater than 10 per cent. When the turbine generator starts accelerating, the governor valves and the intercept valves begin to close to maintain set frequency. At the same time the megawatt demand signal is reduced, which reduces the governor speed changer setting, feedwater flow demand, and reactor power level demand. As the governor valves close, the steam pressure rises and acts through the control system to reinforce the feedwater flow demand reduction already initiated by the reduced megawatt demand signal. In addition, when the load rejection is of sufficient magnitude, the turbine bypass valves open to reject excess steam to the condenser, and the safety valves open to exhaust steam to the atmosphere. The rise in steam pressure and the reduction in feedwater flow cause the average reactor coolant temperature to rise which reinforces the reactor power level demand reduction, already established by reduced megawatt demand, to restore reactor coolant temperature to set value.

As the turbine generator returns to set frequency, the turbine controls revert to steam pressure control rather than frequency control. This feature holds steam pressure within relatively narrow limits and prevents further large steam pressure changes.

7.3 INSTRUMENTATION

7.3.1 NUCLEAR INSTRUMENTATION

The nuclear instrumentation system is shown in Figure 7-14. The system meets the intent of the Proposed IEEE "Criteria for Nuclear Power Plant Protection Systems", dated August, 1968, (IEEE No. 279), for those elements associated with the Reactor Protective Systems.

7.3.1.1 Design Basis

The nuclear instrumentation (NI) system is designed to supply the reactor operator with neutron information over the full operating range of the reactor and to supply reactor power information to the RPS and to the Integrated Control System.

The system sensors and instrument strings are redundant in each range of measurement. Measurement ranges are designed to overlap to provide complete and continuous information over the full operating range of the reactor.

7.3.1.2 System Design

The nuclear instrumentation has nine channels of neutron information divided into three ranges of sensitivity: source range, intermediate range, and power range. The three ranges combine to give a continuous measurement of reactor power from source level to approximately 125 per cent of rated power or ten decades of information. A minimum of one decade of overlapping information is provided between successive higher ranges of instrumentation. The relationship between instrument ranges is shown in Figure 7-15.

The source range instrumentation has two redundant count rate channels originating in two high sensitivity proportional counters. These channels are used over a counting range of 0.1 to 10^6 counts/sec as displayed on the operator's control console in terms of log counting rate. The channels also measure the rate of change of the neutron level as displayed for the operator in terms of startup rate from -0.5 to +5 decades/min. An interlock is provided, i.e., a control rod withdraw "inhibit" on a high startup rate of +2 decades/min., in either channel.

The intermediate range instrumentation has two log N channels originating in
two electrically identical gamma-compensated ion chambers. Each channel pro-
vides eight decades of flux level information in terms of the log of ion
3. chamber current and startup rate. The ion chamber output range is from 10^{-11}
to 10^{-3} amperes. The startup rate range is from -0.5 to +5 decades/min. A
high startup rate of +3 decades/min. in either channel will initiate a control
rod withdraw inhibit.

The power range channels have five linear level channels originating in 5 composite uncompensated ion chambers. The channel output is directly proportional to reactor power and covers the range from 0 to 125 per cent of rated power. The gain of each channel is adjustable providing a means for calibrating the output against a reactor heat balance.

Power range channels NI-5, 6, 7, and 8 supply reactor power level information continuously to the RPS. Dual indicators on the control console provide the operator with both total reactor power information, ϕ, and reactor power imbalance information, $\Delta\phi$, from each of the four channels. The method of obtaining ϕ and $\Delta\phi$ is described in Section 7.3.1.2.1.

The fifth power range channel, NI-9, provides reactor power information to the ICS and to a recorder located on the control console above the dual indicators. The channel is in no way associated with the RPS. While channel NI-9 is the normal source of reactor power information to the ICS and to the power range recorder, the operator may elect to use a selected channel supplying the RPS as the source. Isolation amplifiers are used to provide isolation from the RPS. Isolation amplifiers are used to buffer all signals leaving the system cabinets, preventing the reflection of faults on external signal lines back into the system.

7.3.1.2.1 Neutron Detectors

The detectors used in the source range channels are BF_3 proportional counters. The detector high voltage is automatically turned off when the flux level is approximately one decade above the useful operating range. Conversely, the high voltage is turned on automatically when the flux level returns to within approximately one decade of the maximum useful range. The conditions under which the high voltage will be automatically turned off are that the flux level be above 10^{-9}A in both intermediate ranges or 10 per cent power in power range channels NI-5 or NI-6 and NI-7 or NI-8.

The intermediate range compensated ion chambers are of the electrically adjustable gamma-compensating type. Each detector has a separate adjustable high voltage power supply and an adjustable compensating voltage supply.

Uncompensated ion chambers are used in the power range channels. Each power range detector consists of two 72-inch sections with a single high voltage connection and two separate signal connections. The outputs of the two sections are summed and amplified by the linear amplifiers in the associated power range channel to obtain a signal proportional to total reactor power, ϕ. A signal proportional to the difference in percent full power between the top and bottom halves of the core, the reactor power imbalance or $\Delta\phi$, is derived from the difference in currents from the top and bottom sections of the detector. The difference signal is displayed on the control console to permit the operator to maintain proper axial power distribution. The manual test and calibration facilities provide a means for reading the output of the individual sections of the detector. Each detector has a combined sensitive volume extending approximately from the bottom to the top of the reactor core.

The physical locations of the neutron detectors are shown in Figure 7-16. The power range detectors for channels NI-5, 6, 7 and 8 are positioned adjacent to each of the four quadrants of the core. The power range detector for channel NI-9 is adjacent to the power range detector for channel NI-5. The normal source range detectors are located on opposite sides of the core 180 degrees apart. The intermediate range detectors are also located on opposite sides of the core 180 degrees apart.

Table 7-5 provides pertinent characteristics of the out-of-core neutron detectors. The flux ranges illustrated in Figure 7-15 are seen to be compatible with these characteristics. Nearly identical Westinghouse out-of-core detectors are presented in use at power reactors as follows:

Tube Type	Reactors	Utility
8. PC	Haddam Neck San Onofre	Connecticut Yankee Power Southern California Edison
CIC	Beznau-I R.E. Ginna	NOK Rochester Gas & Electric
UCIC	Haddam Neck	Connecticut Yankee Power

Table 7-5
CHARACTERISTICS OF OUT-OF-CORE
NEUTRON DETECTOR ASSEMBLIES

Characteristic	Source	Intermediate	Power
Type Tube	PC(WL-23682)	CIC(WL-23635)	UCIC(WL-23636)
4. Sensitivity			
Thermal Neutron Flux	31 CPS/nv	4.4×10^{-14} A/nv	3.4×10^{-13} A/nv
Gamma Flux	NA	2.3×10^{-13} A/R/hr (compensated)	3×10^{-10} A/R/hr
Maximum Ratings			
External Pressure	180 psia	180 psia	180 psia
8. Temperature	300 F	300 F	300 F
Thermal Neutron Flux			
Operating	1.7×10^{4} nv	2.5×10^{10} nv	2.5×10^{10} nv
Non-Operating	1×10^{10} nv	2.5×10^{11} nv	2.5×10^{11} nv
Gamma Flux	1×10^{5} R/hr	5×10^{5} R/hr	5×10^{5} R/hr
Integrated Exposure Before 10% Reduction in Sensitivity			
Neutron	10^{19} nvt	10^{19} nvt	10^{19} nvt
Gamma	3×10^{9} R	3×10^{9} R	3×10^{9} R

7.3.1.2.2 Test and Calibration

Test and calibration facilities are built into the system to permit an accurate calibration of the system and the detection of system failures in accordance with the requirements of Reactor Protective System design and IEEE No. 279.

7.3.1.3 System Evaluation

The nuclear instrumentation will monitor the reactor over a minimum 10-decade range from source to 125 percent of rated power. The full power neutron flux level at the power range detectors will be approximately 3.2×10^{9} nv. The
4. 1. detectors employed will provide a linear response up to approximately 2.5×10^{10} nv before they are saturated.

The intermediate range channels overlap the source range and the power range channels as shown in Figure 7-15, providing the continuity of information needed during startup.

The steady-state radial flux distribution within the reactor core will be
measured by the incore neutron detectors (7.3.3). Both out-of-core and incore
detectors will be used to obtain the axial power distribution. The sum of the
outputs from the two sections of each (out-of-core) power range detector will
4. be calibrated to a heat balance. The sum will be recalibrated whenever it is
determined that the sum disagrees with the heat balance by 2% or more. The
signals from the two sections of the detector may be individually read and com-
pared independent of the sum of the outputs. The operator, therefore, may cor-
relate the difference signal against the core power distribution obtained from
the incore system.

7.3.1.3.1 Primary Power

The nuclear instrumentation draws its primary power from vital buses and uninterruptable buses described in 8.2.2.

7.3.1.3.2 Reliability and Component Failure

The requirements established for the reactor protection system apply to the nuclear instrumentation. All channel functions are independent of every other channel, and where signals are used for safety and/or control, electrical isolation is employed to meet the criteria of 7.1.1.

7.3.1.3.3 Relationship to Reactor Protective System

The relation of the nuclear instrumentation to the RPS is described in 7.1. Power range channels NI-5, 6, 7, and 8 are associated with the Reactor Protective System. One of these may also provide information for the Integrated Control System.

The periodic test requirements of the Reactor Protective System are not dictated by the accuracy of the power range channels. The accuracy of the linear amplifiers is better than $\pm$ 0.2 percent including drift.

7.3.2 NON-NUCLEAR PROCESS INSTRUMENTATION

7.3.2.1 <u>Design Bases</u>

The non-nuclear process instrumentation provides the required input signals of process variables for the reactor protective, regulating, and auxiliary systems. It performs the required process control functions in response to those systems and provides instrumentation for startup, operation, and shutdown of the reactor system under normal and emergency conditions.

7.3.2.2 System Design

The non-nuclear instrumentation provides measurements used to indicate, record, alarm, interlock, and control process variables such as pressure, temperature, level, and flow in the reactor coolant, steam supply, and auxiliary reactor systems as shown in Figure 7-17 and system drawings in Sections 6, 9, and 11. Process variables required on a continuous basis for the startup, operation, and shutdown of the unit are indicated, recorded, and controlled at the control rooms. Alternate essential indicators and controls are provided at other locations to maintain the reactor in a hot shutdown condition if the control rooms have to be evacuated. Other instrumentation is provided at auxiliary panels with alarm at the control rooms.

Response time and accuracy of measurements are adequate for reactor protective and regulating systems and other control functions to be performed.

Instrumentation in the protective systems is provided to operate as required under the environmental conditions specified in Paragraph 7.1.1.7.

7.3.2.2.1 Non-Nuclear Process Instrumentation in Protective Systems

Four independent measurement channels are provided for each process parameter for input to the Reactor Protective System.

Three independent measurement channels are provided for each process parameter for input to the Engineered Safeguards Protective System.

a. Reactor Outlet Temperature

Reactor outlet temperature inputs to the Reactor Protective System are provided by two (2) fast-response resistance elements and associated transmitters in each loop.

b. Reactor Coolant Flow

Reactor coolant flow inputs to the Reactor Protective System are provided by eight (8) high-accuracy differential pressure transmitters which measure flow through calibrated flow tubes. Operation of each reactor coolant pump breaker is also monitored as an indication of flow.

An additional differential pressure transmitter for each loop is utilized as the normal flow measurement providing input to the Integrated Control System.

c. Reactor Coolant Pressure

Reactor Protective System inputs of reactor coolant pressure are provided by two (2) pressure transmitters in each loop. An additional pressure transmitter, independent of the Reactor Protective System, will be provided for pressurizer pressure control.

Engineered Safeguards Protective System inputs of reactor coolant pressure in each loop are provided by redundant pressure transmitters. One pressure signal is utilized for recording, low pressure alarm, and interlock to decay heat removal return flow valves.

d. Reactor Building Pressure

Reactor building pressure inputs to the Engineered Safeguards Protective System are provided by:

1. Three (3) pressure transmitters which are located outside the reactor building. These provide inputs for initiation of reactor building isolation, high pressure injection, low pressure injection, and reactor building cooling.
2. Three (3) groups of two pressure switches each are located outside the reactor building. These provide input signals of

high reactor building pressure for initiation of reactor building spray by safeguards actuation.

The following tables provide pertinent information concerning the NNI sensors supplying inputs to the RPS and Engineered Safeguards System, respectively:

Table 7-6

NNI Inputs to RPS

Characteristic	Reactor Outlet Pressure (NR)[1]	Reactor Outlet Temperature (NR)	Reactor Coolant Flow
Component Item Number	RC3A-PT1 RC3A-PT2 RC3B-PT1 RC3B-PT2	RC4A-TE1 RC4A-TE4 RC4B-TE1 RC4B-TE4	RC14A-dPT1 RC14A-dPT2 RC14A-dPT3 RC14A-dPT4 RC14B-dPT1 RC14B-dPT2 RC14B-dPT3 RC14B-dPT4
Reactor Protective Channel	A,B,C,D	A,B,C,D	A,B,C,D[2]
37 Sensor Type	Rosemount Press. Transmitter	Rosemont RTD	Bailey Differential Pressure
Accuracy [4]	$\pm 0.5\%$ of span	$\pm 0.17°F$	$\pm 0.25\%$ of span
Repeatability [4]	$\pm 0.1\%$ of span	NA	$\pm 0.05\%$ of span
Expected Failure Mode	to low pressure	to hi temperature	to low flow
Type Readout	all indicating	all indicating	all indicating
Power Required	external	external	external
Sensors Connected to Common Taps	RC3A-PT1 & RC3A-PT3[3] RC3A-PT1 & RC3A-PT4 RC3B-PT1 & RC3B-PT3	All sensors have separate taps.	All sensors for same loop are connected to common taps.

NOTES: (1) NR = Narrow Range

(2) Each channel has an input from each loop.

(3) Pressure taps for each RPS channel are independent. A RPS channel and an ESPS channel may have common pressure sensing taps.

(4) Manufacturer's guarantee

Table 7-7

NNI Inputs to Engineered Safeguards

Characteristic	Reactor Outlet Pressure (WR)[1]	Reactor Building Pressure (WR)	Reactor Building Pressure (NR)
6. Component Item Number	RC3A-PT3 RC3A-PT4 RC3B-PT3	BS4-PS1 & 2 BS4-PS3 & 4 BS4-PS5 & 6	BS4-PT1 BS4-PT2 BS4-PT3
ESPS Channel	A,B,C	A,B,C	A,B,C
Sensor Type	Motorola Pressure	Static -0-Ring Pressure Switch	Motorola Pressure
4. Accuracy (2)	±0.5% of span	NA	±0.5% of span
Repeatability (2)	±0.1% of span	±3% of span	±0.1% of span
Expected Failure Mode	to low pressure	no contact action	to low pressure
Type Readout	all indicating	NA	all indicating
Power Required	external	none	external
6. Sensors Connected to Common Taps	See Note (3) on Table 7-6	BS4-PS1,BS4-PS2 & BS4-PT1 BS4-PS3,BS4-PS4 & BS4-PT2 BS4-PS5,BS4-PS6 & BS4-PT3	All separate building penetrations

NOTE: (1) WR = Wide Range

(2) Manufacturers guarantee

7.3.2.2.2 Non-Nuclear Process Instrumentation in Regulating Systems

Selective redundant measurements and input signals are provided for the process variables required for critical control functions.

The following inputs to the Integrated Control System are provided:

a. Reactor Outlet Temperature

Selected loop or unit average outlet temperature input is provided in each loop by two (2) fast response resistance elements and associated transmitters.

b. Reactor Controlling Average Temperature

Instrumentation separate from the Reactor Protective System supplies input to the Integrated Control System.

Loop or unit average temperature signals are selected for indication and input as controlling average temperature. Automatic selection determined by loop flows is provided for input of the appropriate signal.

Reactor inlet temperature signals required for loop, and unit average or differential temperatures are provided in each loop by two (2) fast response resistance elements and associated transmitters.

c. Reactor Inlet Differential Temperature

Reactor inlet differential temperature is indicated and provided for input to the Integrated Control System.

d. Reactor Coolant Flow

Reactor coolant flow signals are provided for each loop and summed for total flow. Total flow is recorded and "low" total flow is alarmed. Flow measurement in each loop is provided by a transmitter, independent of the Reactor Protective System, which measures flow through the flow tube.

Selective redundant measurement by one of the other four loop flow transmitters is also provided from the Reactor Protective System.

Loop "low" flow signals provide the logic for automatic selection of reactor controlling average temperature.

Contacts from reactor coolant pump motor breakers provide fast indication to the ICS that a pump has tripped.

~~7-35~~ OVER FOR 7-36 →

e. Feedwater Temperature

Feedwater temperature input is provided by two (2) fast response resistance elements and associated transmitters. The selected input also provides signals for indication and feedwater temperature compensation.

f. Feedwater Flow

Feedwater flow input is provided from a full range flow calculator for each loop. The calculator automatically selects and computes startup or main feedwater flow signals to provide the required full range flow input signal. The main feedwater flow measurement in each loop is provided by redundant differential pressure transmitters that measure flow through a flow nozzle. Startup feedwater flow measurement in each loop is provided by a differential pressure transmitter that measures flow through a flow nozzle.

g. Feedwater Control Valves Differential Pressure

Pressure drop measurement across the valves is provided for input by redundant differential pressure transmitters. The selected input signal is also indicated.

h. Steam Generator Level

Selected "startup" level and "operate" level inputs are provided from each steam generator. Redundant measurements of each level are provided by differential pressure transmitters. Temperature compensation to augment the predetermined compensation for normal operating temperature is provided by two (2) fast response resistance elements and associated transmitters which measure steam generator lower downcomer temperature.

The selected "operate" level input is recorded and "high" level alarmed. The selected "startup" level input is indicated and "low" level alarmed.

A full range level measurement is provided for indication of each steam generator level but does not provide protective or regulating systems input.

i. Steam Generator Outlet Pressure

Selected outlet pressure input is provided from each steam generator. Measurement is made by pressure transmitters in both outlet lines of each steam generator. The selected input is also indicated.

j. Turbine Header Pressure

Turbine header pressure measurement is provided for input by a pressure transmitter in each header line from the steam generators. The selected pressure signal is also recorded, and high and low pressures alarmed. Additional redundant transmitters in each header line provide indication only.

7.3.2.2.3 Other Non-Nuclear Process Instrumentation

The following instrumentation is provided for measurement and control of process variables necessary for proper reactor operation:

a. Pressurizer Temperature

Pressurizer temperature is measured by two (2) fast response resistance elements and their associated transmitters. The selected output signal is indicated and supplies input for pressurizer level temperature compensation.

b. Pressurizer Level Control

Pressurizer level is measured by three (3) differential pressure transmitters. One signal is selected for temperature compensation and output for recording, level control, alarms and interlock to deenergize the pressurizer electric heaters on low level. The level controller output positions the makeup control valve in the High Pressure Injection to maintain a minimum preset level. Pressurizer level is lowered by reactor coolant letdown by manual control at the control room.

c. Reactor Coolant Pressure Control

The reactor coolant pressure signal for control is provided by a fifth channel measurement. Selective redundant measurement by one of the other four pressure transmitters is also provided from the Reactor Protective System.

The selected signal is used as an input to pressure switches which provide signals for automatic control of:

1. Pressurizer electric heaters.

2. Pressurizer spray control valve.

3. Pressurizer electromatic relief valve.

The heaters are grouped in banks which are energized below preset pressures.

The selected signal also provides input to a pressure controller which automatically modulates the output of one bank of heaters to maintain a preset pressure. The spray and relief valves are opened above preset pressures.

The selected signal is recorded and high and low pressures alarmed.

d. Startup Reactor Coolant Pump Control

Interlock signals of reactor coolant inlet temperature are provided to each pump switching logic to prevent operation of more than three pumps during startup until a preset temperature is reached.

e. Feed and Bleed Control

The feed and bleed control instrumentation in the High Pressure Injection System provides control and interlocks to permit adjustment of the reactor coolant boron concentration.

7.3.2.3 System Evaluation

The quantity and types of process instrumentation have been selected to provide assurance of safe and orderly operation of all systems and processes over the full operating range of the plant. Some of the criteria for design are:

a. Separate instrumentation has been provided for the protective systems and vital control circuits.

b. Time of response and accuracy of measurements are adequate for protective and control functions to be performed.

c. Where wide process variable ranges are required and precise control is involved, both wide range and narrow range instrumentation are provided.

d. All electrical and electronic instrumentation required for operation is supplied from redundant vital and uninterruptable instrumentation buses.

7.3.3 INCORE MONITORING SYSTEM

7.3.3.1 Design Basis

The incore monitoring system provides neutron flux detectors to monitor core performance. Incore, self-powered neutron detectors measure the neutron flux in the core to provide a history of power distributions during power operation. Data obtained provides power distribution information and fuel burnup data to assist in fuel decisions. The plant computer provides normal system readout and a backup readout system is provided for selected detectors.

7.3.3.2 System Design

7.3.3.2.1 System Description

The incore monitoring system for Oconee 1 consists of assemblies of self-powered neutron detectors and temperature detectors located at 52 preselected positions within the core. The incore monitoring locations are shown on Figure 7-18. In this arrangement, an incore detector assembly consisting of seven local flux detectors, one background detector, and one thermocouple is installed in the instrumentation tube of each of 52 fuel assemblies. The local detectors are positioned at seven different axial elevations to indicate the axial flux gradient. The outputs of the local flux detectors are referenced to the background detector output so that the differential signal is a true measure of neutron flux. (The temperature detectors located just above the top of the active fuel in the fuel assemblies measure core outlet temperature and will be used during the testing program.)

For Oconee Units 2 and 3, fifty-two assemblies are provided. The assemblies are similar except that they are not provided with calibration tubes or thermocouples. Readout for the incore detectors is performed by the plant computer. Multi-point recorder readouts of selected detectors are provided independent of the computer.

When the reactor is depressurized, the incore detector assemblies can be inserted or withdrawn through guide tubes which originate at a shielded area in the reactor building as shown in Figure 7-19. These guide tubes enter the bottom head of the reactor vessel where internal guides extend up to the instrumentation tubes of 52 selected fuel assemblies. The instrumentation tube serves as the guide for the incore detector assembly. During refueling operations, the incore detector assemblies are withdrawn approximately 13 feet to allow free transfer of the fuel assemblies. After the fuel assemblies are placed in their new locations, the incore detector assemblies are returned to their fully inserted positions.

7.3.3.2.2 Calibration Techniques

The nature of the detectors permits the manufacture of nearly identical detectors which produces a high relative accuracy between individual detectors. The detector signals are compensated continuously for burnup of the neutron sensitive material.

Calibration of detectors is not required. The in-core self-powered detectors are controlled to precise levels of initial sensitivity by quality control during the manufacturing stage. The sensitivity of the detector changes over its lifetime due to such factors as detector burnup, control rod position, fuel burnup, etc. The results of experimental programs to determine the magnitude of these factors have been incorporated into calculations and will be used to correct the output of the in-core detectors for these factors. Operation of detectors in both power and test reactors has demonstrated that this compensation program, when coupled with the initial sensitivity, provides detector readout accuracies sufficient to eliminate the need for a calibration system.

However, a calibration system has been installed in Oconee 1 to provide on-line confirmation of the experimentally derived compensation calculation methods. The calibration system consists of incore self-powered neutron detectors positioned by hand in selected detector assembly calibration tubes as shown in Figure 7-18.

7.3.3.3 System Evaluation

7.3.3.3.1 Operating Experience

Self-powered in-core neutron detectors have been operated since 1962. Such detectors have been assembled and irradiated in a Babcock & Wilcox development program that began in 1964.

The B&W Development Program included these tests:

a. Parametric studies of the self-powered detector.

b. Detector ability to withstand PWR environment.

c. Multiple detector assembly irradiation tests.

d. Background effects.

e. Readout system tests.

f. Mechanical withdrawal-insertion tests.

g. Mechanical high pressure seal tests.

h. Relationship of flux measurement to power distribution experiments.

Conclusions drawn from the results of the test programs are as follows:

a. The detector sensitivity, resistivity, and temperature effects are satisfactory for use.

b. A multiple detector assembly can provide axial flux data in a single channel and can withstand reactor environment.

c. Background effects will not prevent satisfactory operation in a PWR environment.

d. Plant computer systems are successful as read-out system for in-core monitors.

For incore monitoring system development program results and conclusions, refer to B&W Topical Report BAW-10001; "In-Core Instrumentation Test Program."

7.3.3.3.2 Detection and Control of Xenon Oscillations

Under normal operating conditions, the in-core detectors supply information to the operator in the control room.

Each individual detector measures the neutron flux at its locality and is used to determine the local power density. The individual power densities are then averaged and a peak-to-average power ratio calculated. This information can be used to indicate possible power oscillations.

The application of this system for detection of xenon oscillation and its minimum sensitivity is continuing to be examined through the analysis of experimental data. However, previous performance data are available to demonstrate performance capability. A series of Physics Verification Program Reports developed under AEC Contract No. AT(30-1)-3647 and B&W Contract No. 41-2007 have previously been submitted to the Commission for review. Much of the data compiled was taken by self-powered detectors and shows the performance capabilities of the detectors. Upon initial installation, the self-powered detector has the capability to measure the relative flux with an accuracy of 5 per cent of the flux when used in conjunction with an adjacent background detector. The sensitivity of the detector will decrease with exposure to neutron flux due to transmutation of the emitter in the detector. However, by use of integrated current inventories, it is felt that the additional inaccuracies shall be no more than 1 per cent per year for the average flux conditions.

The use of the in-core monitoring system to detect xenon oscillations is described in B&W Topical Report BAW-10010; "Stability Margins for Xenon Oscillations."

7.4 OPERATING CONTROL STATIONS

Following proven power station design philosophy, all control stations, switches, controllers, and indicators necessary to start up, operate, and shut down Units 1 and 2 are located in one control room. Controls for Unit 3 are located in a separate control room. Control functions necessary to maintain safe conditions after a loss-of-coolant accident are initiated from the centrally located control rooms. Controls for certain auxiliary systems are located at remote control stations when the system controlled does not involve unit control or emergency functions.

7.4.1 GENERAL LAYOUT

The control room for Units 1 and 2 is designed so that one man can supervise operation of both units during normal steady state conditions. During other than normal operating conditions, other operators are available to assist the control operator. Figure 7-20 shows the control room layout for Units 1 and 2. Unit 3 has similar accessibility to the various controls. The control boards are subdivided to show the location of control stations and to display information pertaining to various sub-systems.

7.4.2 INFORMATION DISPLAY AND CONTROL FUNCTIONS

Consideration is given in the control board layout to the fact that certain systems normally require more attention from the operator. The integrated control system is therefore located nearest the center line of the boards (Section 1 on Figure 7-20).

On Section 2 of the control board, one indicator will be provided for each control rod. Fault detectors in the rod drive control system are used to alert the operator should an abnormal condition exist for any individual control rod. Displayed in this same area are limit lights for each control rod group and all nuclear instrumentation information required to start up and operate the reactor. Control rods are manipulated from the Section 2 bench position. Computer readout facilities for alarm monitoring and sequence monitoring are located here to aid the operator.

A process computer is used on each unit to provide fuel management measurements and calculations. These computers also provide for alarm monitoring, performance monitoring, data logging, sequence monitoring, and facilitate control of some functions during start-up and shut-down of the turbine-generator. Monitoring and display functions of the computer which audit nuclear steam supply system parameters of major interest are duplicated elsewhere in the control rooms. This type of computer application has been successfully applied to units presently in operation on the Duke system.

Variables associated with operation of the secondary side of the station are displayed and controlled from Section 1 and 3 of the control board. These variables include steam pressure and temperature, feedwater flow and temperature, electrical load, and other signals involved in the integrated control system. Section 3 of the control board also contains indication and controls of the reactor coolant system parameters.

The engineered safeguards system is controlled and monitored from Section 8 of the vertical boards. Indicating lights are provided as a means of verifying the proper operation of the engineered safeguards system. Control switches located on this panel allow manual operation of equipment that is not controlled elsewhere in the control room or test of individual units.

Control and display equipment for station auxiliary systems are located on Section 6 of the control board.

Reactor coolant pump controls located on Section 5 of the control boards consists of the pump controls and auxiliary instrumentation required for pump operation. Also mounted on this section are the auxiliary electrical system controls required for manual switching between the various power sources described in 8.2.2.

Controls and indications for all normal ventilation systems are located on Section 7 of the control boards.

In order to maintain the desired accessibility for control of the station, miscellaneous recorders not required for station control are located on the vertical recorder boards where they are visible to the operator. Radiation monitoring information is also indicated there.

7.4.3 SUMMARY OF ALARMS

Visible and audible alarm units are incorporated into the control boards to warn the operator if limiting conditions are approached by any system. Audible Reactor Building evacuation alarms are initiated from the radiation monitoring system and from the source range nuclear instrumentation. Audible alarms are sounded in appropriate areas throughout the station if high radiation conditions are present in that area. Alarms for the nuclear systems are indicated in process diagrams in sections 6, 7, and 9.

7.4.4 COMMUNICATION

7.4.4.1 Control Room to Inside Station Communication

The telephones within the station are connected to a PAX which provides for communication and paging. The 200 line PAX equipment provides 3-digit dialing, dial tone, ring-back tone, busy tone, and telephone signal. The power supply is normally 115V AC providing 48V DC at 5 ampere output. Upon loss of AC power, automatic switch-over to standby batteries is accomplished in less than 50 milliseconds. The batteries will supply power for one hour of operation.

The public address system is accessible to plant telephone by dial. In the event of PAX failure, the PA system is operable through eleven handsets installed at strategic locations in the station.

A sound powered telephone system is provided and will consist of network of conductor pairs converted to jacks through the plant. Sound powered hand-
4. sets will be plugged into the jacks to form talking paths. There are nine (9) separate talking paths available for each unit. The system is completely independent from any other telephone system and involves no external power supply.

7.4.4.2 Control Room to Outside Station

The commercial and microwave telephone system provides direct communication to points outside the station area. The commercial telephone provides dial access through the PAX to the public telephone system. One part of the microwave system is integrated into the PAX and includes wireless access to the switchboard in the General Office at Charlotte, the Spartanburg Dispatcher, and Lee Steam Station. Another part of the microwave system is independent of the PAX and includes non-dial call ability to Charlotte Dispatcher, Spartanburg Dispatcher, the substation at Central, and Lee Steam Station. For an emergency situation involving loss of AC power, the microwave transmitter has its own battery for eight hours operation and propane-engine generator with fuel for at least one week.

The control room is also equipped with transmitter-receivers which operate on 47.98 megahertz and 47.84 megahertz and have ability to call the substation at Central as well as mobile receivers.

7.4.4.3 Exclusion Area Control

An emergency vehicle and a boat are provided, each of which contains a transmitter-receiver for communications with the Unit 1-2 control room and an amplifier speaker to be used for warning in the exclusion area.

7.4.5 OCCUPANCY

Safe occupancy of the control room during abnormal conditions is provided for in the design of the auxiliary building. Adequate shielding is used to maintain tolerable radiation levels in the control rooms for maximum hypothetical accident conditions. Each control room ventilation system is provided with radiation detectors and appropriate alarms. Provisions are made for the control room air to be recirculated through absolute and charcoal filters. Emergency lighting is provided.

The potential magnitude of a fire in either control room is limited by the following factors:

a. The control room construction and furnishings are of noncombustible materials.

b. Control cables and switchboard wiring meet the flame test as described in Insulated Power Cable Engineers Association Publication S-61-402 and National Electrical Manufacturers Association Publication WC 5-1961.

c. Qualified trained personnel, adequate extinguishers, and accessibility to all control room areas are provided.

A fire, if started, would be of such a small magnitude that it could be extinguished by the operator using a hand fire extinguisher. The resulting smoke and vapors would be removed by the ventilation system.

Essential auxiliary equipment is controlled by either stored energy, closing-type, air circuit breakers which are accessible and can be manually closed in the event d-c control power is lost, or by a-c motor starters which have individual control transformers.

If temporary evacuation of the control room is required while operating at any power, the operator will trip the control rods and start the Keowee hydro units prior to evacuating the control room. This action can also be accomplished from the cable room located one elevation below the control room. After evacuation, the operator can establish and maintain a hot shutdown condition from the Emergency Shutdown Panel located outside the control room. The following instrumentation and controls are available on the Emergency Shutdown Panel:

1. Pressurizer Level Indicator
2. Pressurizer Heater Control
3. RC Pressure Indicator
4. RC Outlet Temperature Indicator
5. Turbine Steam Supply Header Pressure Indicator
6. Turbine Bypass Valve Loop "A" Station
7. Turbine Bypass Valve Loop "B" Station
8. Startup Feedwater Valve Loop "A" Station
9. Startup Feedwater Valve Loop "B" Station
10. Steam Generator "A" Startup Level
11. Steam Generator "B" Startup Level
12. Letdown Storage Tank Level Indicator
13. HP Injection Pump "B" Control Switch

4.

If HP Injection Pump "A" is in operation, it can be tripped from the 4.16 KV switchgear located on elevation 796' + 6". The operator has control of HP Injection Pump "B" at the Emergency Shutdown Panel. Makeup to the Letdown Storage Tank can be obtained, if desired from one of the following sources:

1. RC Bleed Holdup Tank
4. 2. Concentrated Boric Acid Storage Tank
3. Boric Acid Mix Tank

The necessary pumps can be controlled from the Waste Disposal Control Panel.

7.4.6 AUXILIARY CONTROL STATIONS

Auxiliary control stations are provided where their use simplifies control of auxiliary systems equipment such as waste evaporator, sample valve selectors, chemical addition, etc. The control functions initiated from local control stations do not directly involve either the engineered safeguards system or the reactor control system. Sufficient indicators and alarms are provided so that the Oconee control room operator is made aware of abnormal conditions involving remote control stations.

7.4.7 SAFETY FEATURES

Control room layouts provide the necessary controls to start, operate and shut down the units with sufficient information display and alarm monitoring to assure safe and reliable operation under normal and accident conditions. Special emphasis is given to maintaining control during accident conditions. The layout of the engineered safeguards section of the control board is designed to minimize the time required for the operator to evaluate ths system performance under accident conditions.

7.5 INDENTIFICATION OF PROTECTIVE EQUIPMENT

All safety related sensors, transmitters, transducers, cabinets, etc. located outside the control room will be physically identified by placement of a permanent, conspicuous tag on or adjacent to the device. A typical tag will measure 4 3/4" x 2 3/4" and will bear the wording "Safety Related."
18. The basic background color will be red with the wording printed on a 1 x 3 inch gray, yellow, blue or orange field. This field color will designate the redundant channel with which the device is identified. The following color scheme will be used:

4. Gray	Swgr 1TC Ld Ctr 1X8 MCC 1XS1 ESG channel 1, 3, 5 & 7 DC Pnlbd 1DIA Vital Pwr Pnlbd 1KVIA RPS Ch A
Yellow	Swgr 1TD Ld Ctr 1X9 MCC 1XS2 ESG channel 2, 4, 6, & 8

	DC Pnlbd 1DTB Vital Pwr Pnlbd 1KVIB RPS Ch B
Blue	Swgr 1TE MCC 1XS3 DC Pnlbd 1DIC Vital Pwr Pnlbd 1KVIC RPS Ch C ESG Channel Even-Odd
Orange	DC Pnlbd 1DID Vital Pwr Pnlbd 1KVID RPS Ch D

4.

B&W TOPICAL REPORTS

3.

BAW-10001	"In-Core Instrumentation Test Program"
BAW-10010	"Stability Margins for Xenon Oscillations"

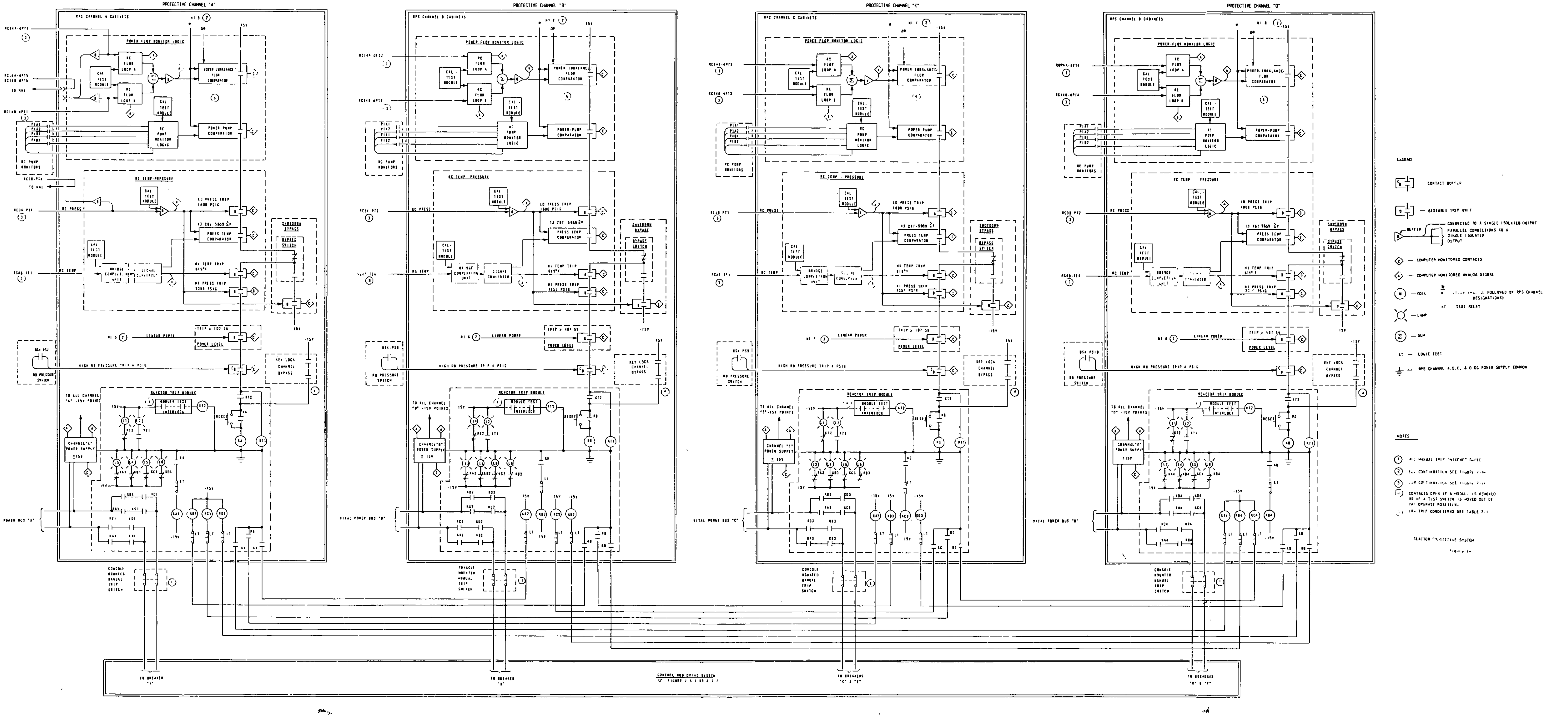
Rev. 3. 3/16/70
Rev. 4. 4/20/70

Table 7.8
Summary of Seismic Considerations Applied to the Reactor Protective System and Engineered Safety Features

EQUIPMENT	LOCATION	SEISMIC DESIGN BASES	TESTS AND ANALYSES
Reactor Protective System Cabinets & Engineering Safety System Cabinets	Auxiliary Building Control Room (Elevation 822')	Maximum floor acceleration[1] at the base of equipment is 0.231g with the critical frequencies[2] between 1 and 20 cycles per second.	Each module or device was tested at frequencies between 7 and 40 cycles per second with accelerations up to and including 1g with no problems.
Reactor Protective System Sensors	Reactor Building (Various Elevations)	Maximum floor acceleration at the base of equipment is 0.23g with the critical frequencies between 0.6 and 20 cycles per second.	Representative sensors are tested between 1 and 20 cycles per second horizontally and between 10 and 40 cycles per second vertically with accelerations up to and including 1g.
Engineering Safety System Sensors	Reactor Building (Various Elevations)	Maximum floor acceleration at the base of equipment is 0.23g with the critical frequencies between 0.6 and 20 cycles per second.	One type of sensor was tested and no resonances were found between 0-100 cycles per second. The maximum deviation noted was 0.25% of the range when the sensor was vibrated at 100 cycles per second and 10g's for two hours. The other type of sensor used was tested satisfactorily at 0.25g over the frequency range of 4 to 20 cycles per second.

NOTES:

1. Maximum floor acceleration shall be defined as the peak acceleration[3] of that floor as a result of the maximum hypothetical earthquake at the base of the supporting structure.
2. Critical frequency shall be defined as that range of frequencies on the respective response spectrum in which there exists an acceleration greater than the maximum floor acceleration.
3. Peak floor acceleration has been determined by an acceleration time history of the supporting structure.



REACTOR PROTECTIVE SYSTEM

OCONEE NUCLEAR STATION

Figure 7 - 1

Rev. 1 9/15/69

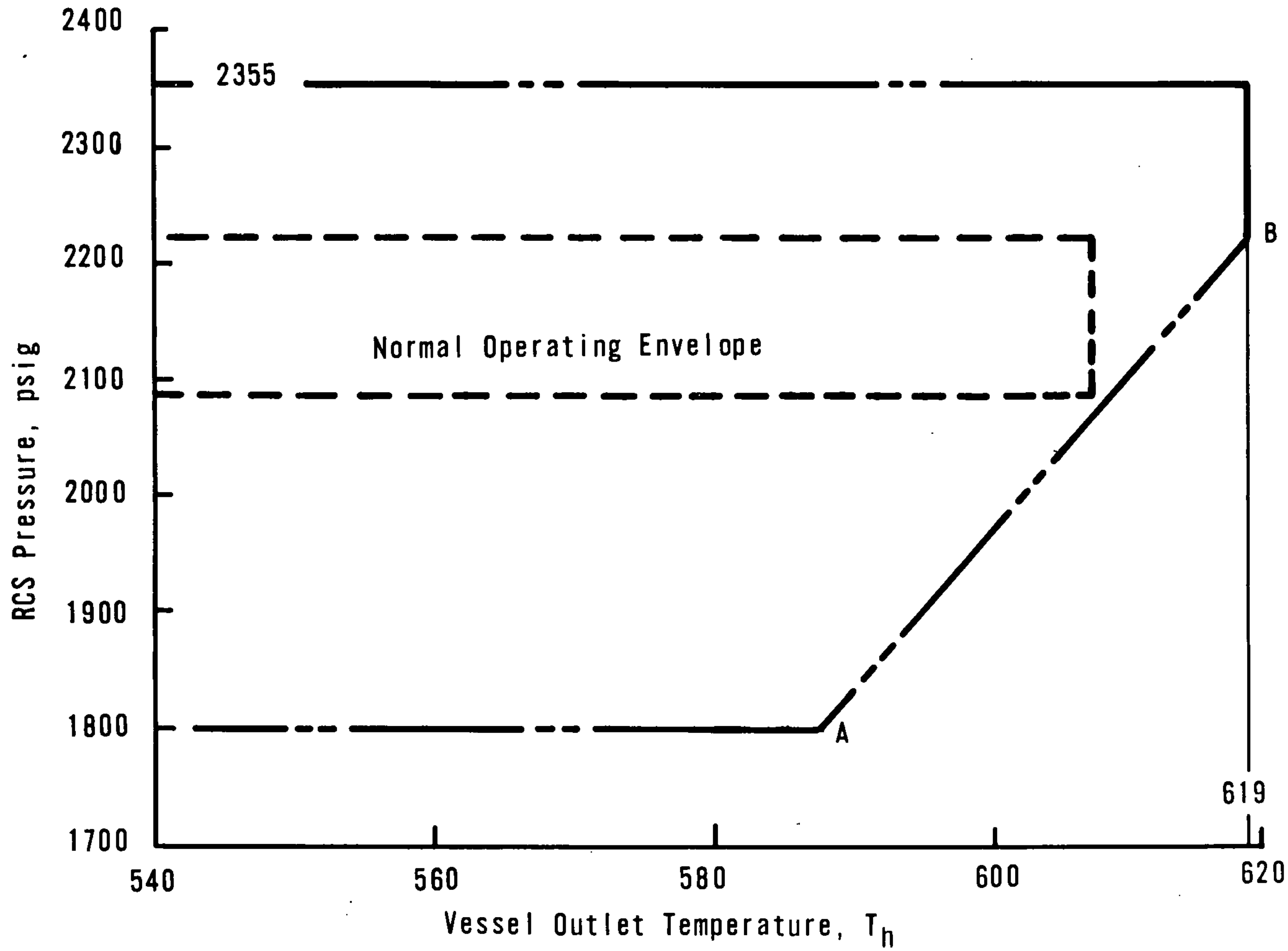
Rev. 3 3/16/70

Rev. 9 8/11/70

Rev. 10 8/28/70

Rev. 15 12/30/70

Rev. 16 7/30/71

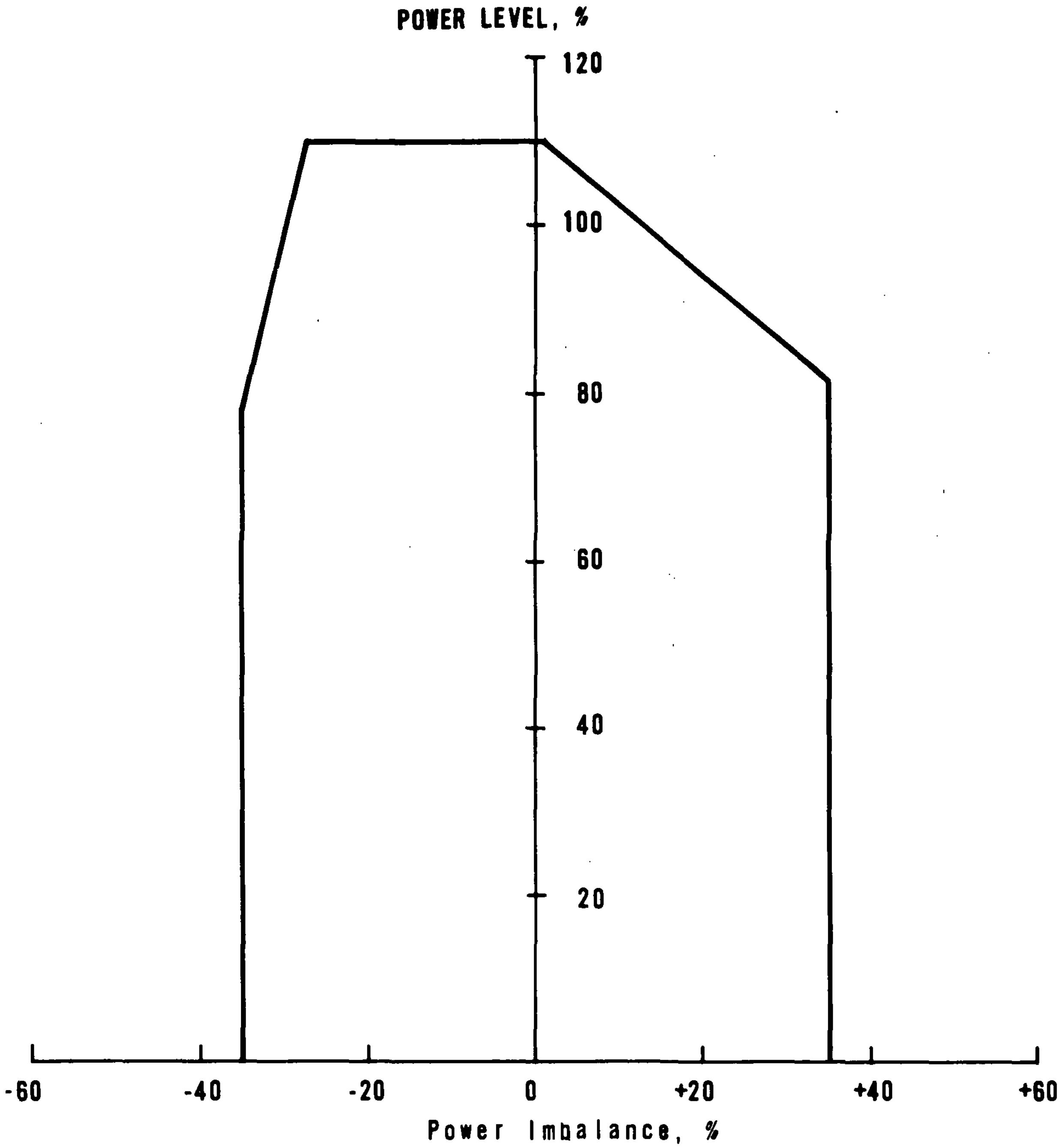


PRESSURE-TEMPERATURE BOUNDARIES

OCONEE NUCLEAR STATION

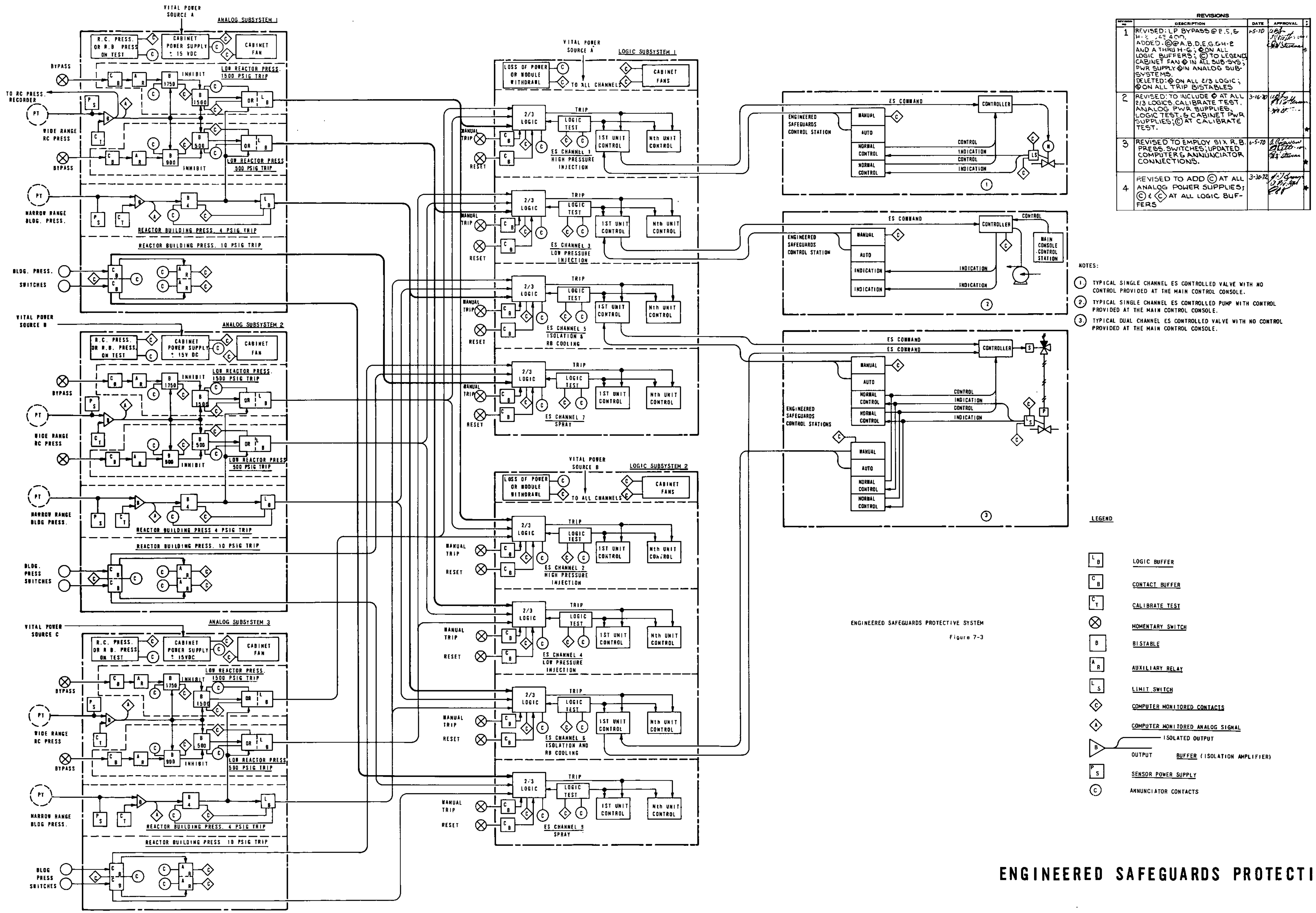
Figure 7 - 2

Rev. 9 8/11/70



POWER-IMBALANCE BOUNDARIES

(Reference Supplement 9 Revisions for Oconee 3)



NOTES:

1. TYPICAL SINGLE CHANNEL ES CONTROLLED VALVE WITH NO CONTROL PROVIDED AT THE MAIN CONTROL CONSOLE.
2. TYPICAL SINGLE CHANNEL ES CONTROLLED PUMP WITH CONTROL PROVIDED AT THE MAIN CONTROL CONSOLE.
3. TYPICAL DUAL CHANNEL ES CONTROLLED VALVE WITH NO CONTROL PROVIDED AT THE MAIN CONTROL CONSOLE.

ENGINEERED SAFEGUARDS PROTECTIVE SYSTEM

Figure 7-3

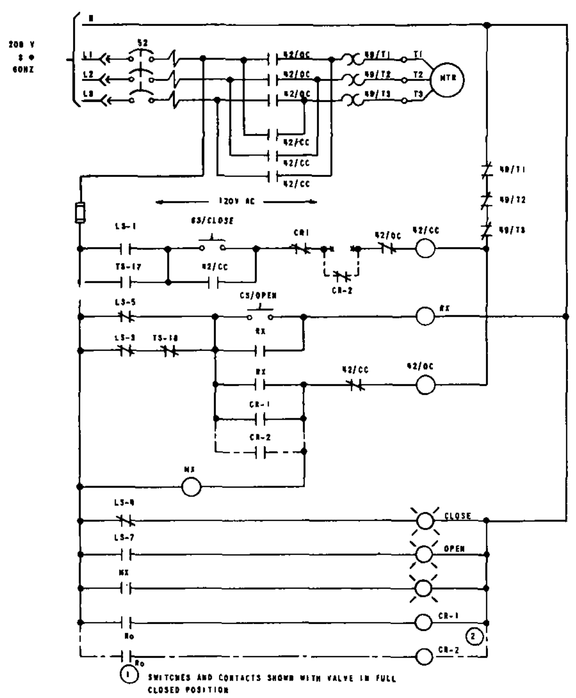
ENGINEERED SAFEGUARDS PROTECTIVE SYSTEM

OCONEE NUCLEAR STATION

Figure 7 - 3

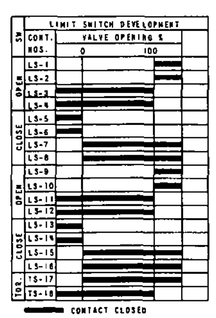
Rev. 19 5/5/72

TYPICAL CONTROL CIRCUIT FOR MOTOR OPERATED VALVES



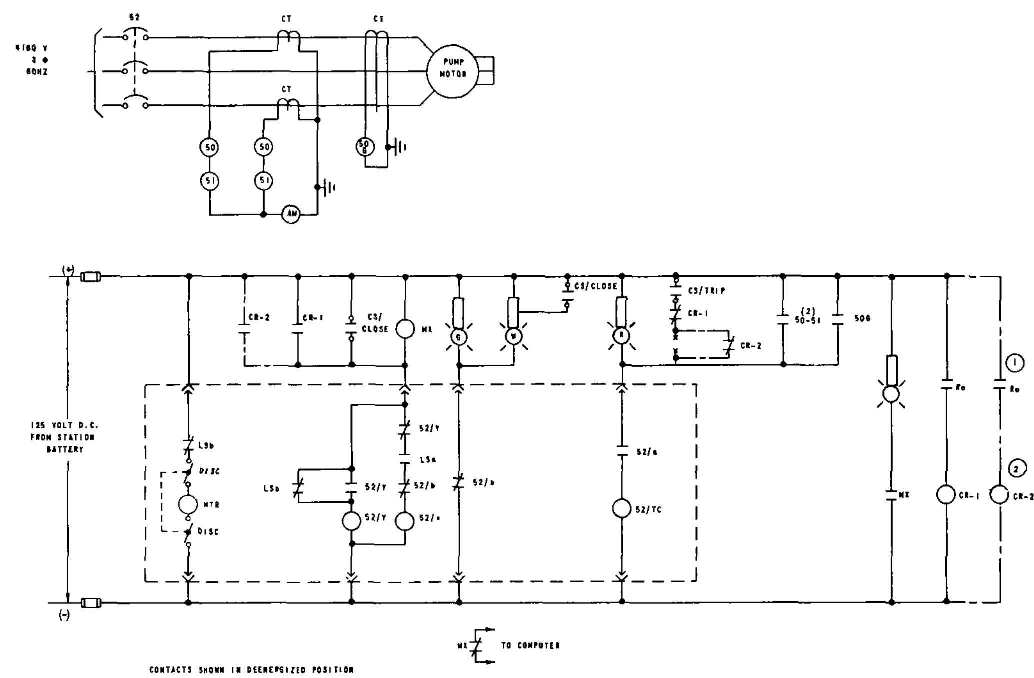
(1) SWITCHES AND CONTACTS SHOWN WITH VALVE IN FULL CLOSED POSITION

52 - AC CIRCUIT BREAKER
49 - THERMAL RELAY
42 - CONTACTOR
CC - CLOSING COIL
OC - OPENING COIL
TS - TORQUE SWITCH
LS - LIMIT SWITCH
CR - CONTROL RELAY
CS - CONTROL SWITCH
MTR - MOTOR
MX - MONITORING RELAY
RX - AUXILIARY RELAY



CONTACT CLOSED

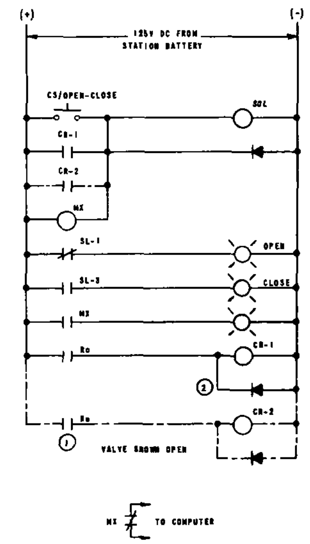
TYPICAL CONTROL CIRCUIT FOR ENGINEERED SAFEGUARD PUMP MOTORS

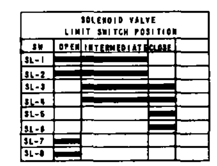


CONTACTS SHOWN IN DEENERGIZED POSITION

TC - TRIP COIL
50 - INSTANTANEOUS OVERCURRENT RELAY
51 - TIME OVERCURRENT RELAY
52 - AC CIRCUIT BREAKER
CS - CONTROL SWITCH
50G - GROUND FAULT RELAY
CR - CONTROL RELAY
MTR - MOTOR
DISC- DISCONNECT
MX - MONITORING RELAY

TYPICAL CONTROL CIRCUIT FOR SOLENOID OPERATED VALVES

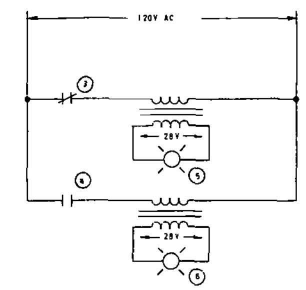




CONTACT CLOSED

CS - CONTROL SWITCH
RX - AUXILIARY RELAY
SOL - SOLENOID
SL - SOLENOID LIMIT SWITCH
CR - CONTROL RELAY
MX - MONITORING RELAY

TYPICAL T&B MODULE INDICATOR LIGHT

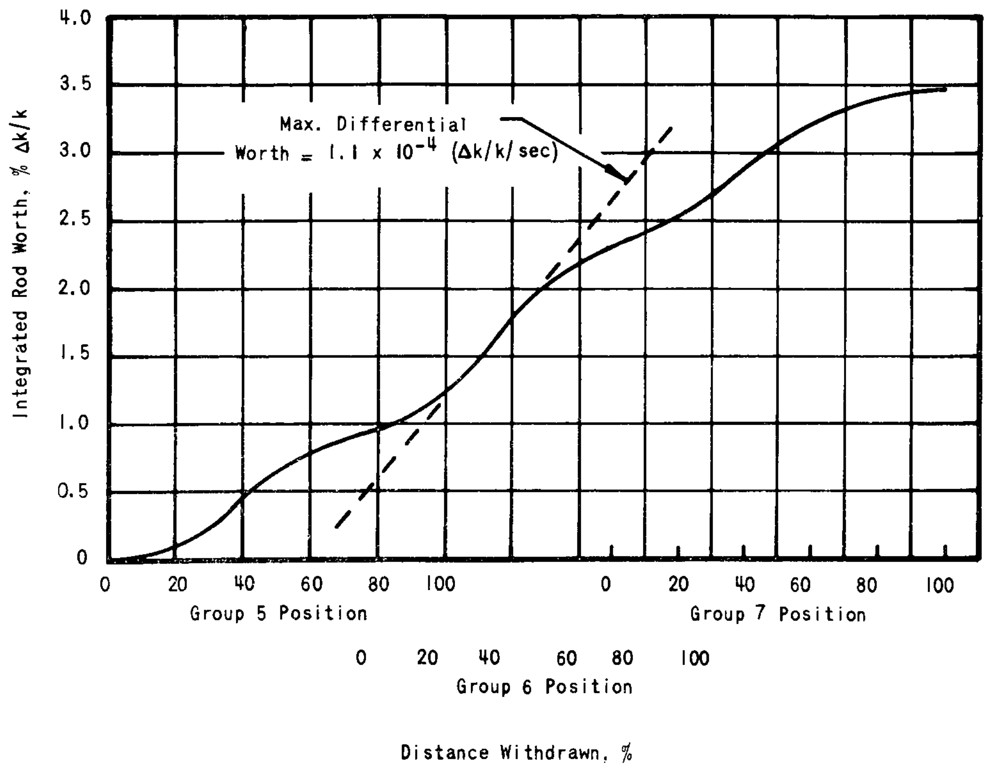


NOTES:

(1.) Ro CONTACTS ARE PART OF THE ENGINEERED SAFEGUARDS PROTECTIVE SYSTEM SEE FIGURE 7-3
(2.) CR-2 USED ONLY WHEN TWO CHANNEL CONTROL IS REQUIRED
(3.) LS-9 FOR MOTOR OPERATED VALVES, SL-7 FOR SOLENOID VALVES, AND 52/b FOR PUMP SWITCHGEAR.
(4.) LS-6 FOR MOTOR OPERATED VALVES, SL-5 FOR SOLENOID VALVES, AND 52/a FOR PUMP SWITCHGEAR.
(5.) "OPEN" FOR VALVES AND "STOP" FOR PUMPS.
(6.) "CLOSE" FOR VALVES AND "RUN" FOR PUMPS

TYPICAL CONTROL CIRCUITS FOR ENGINEERED SAFEGUARDS SYSTEM EQUIPMENT

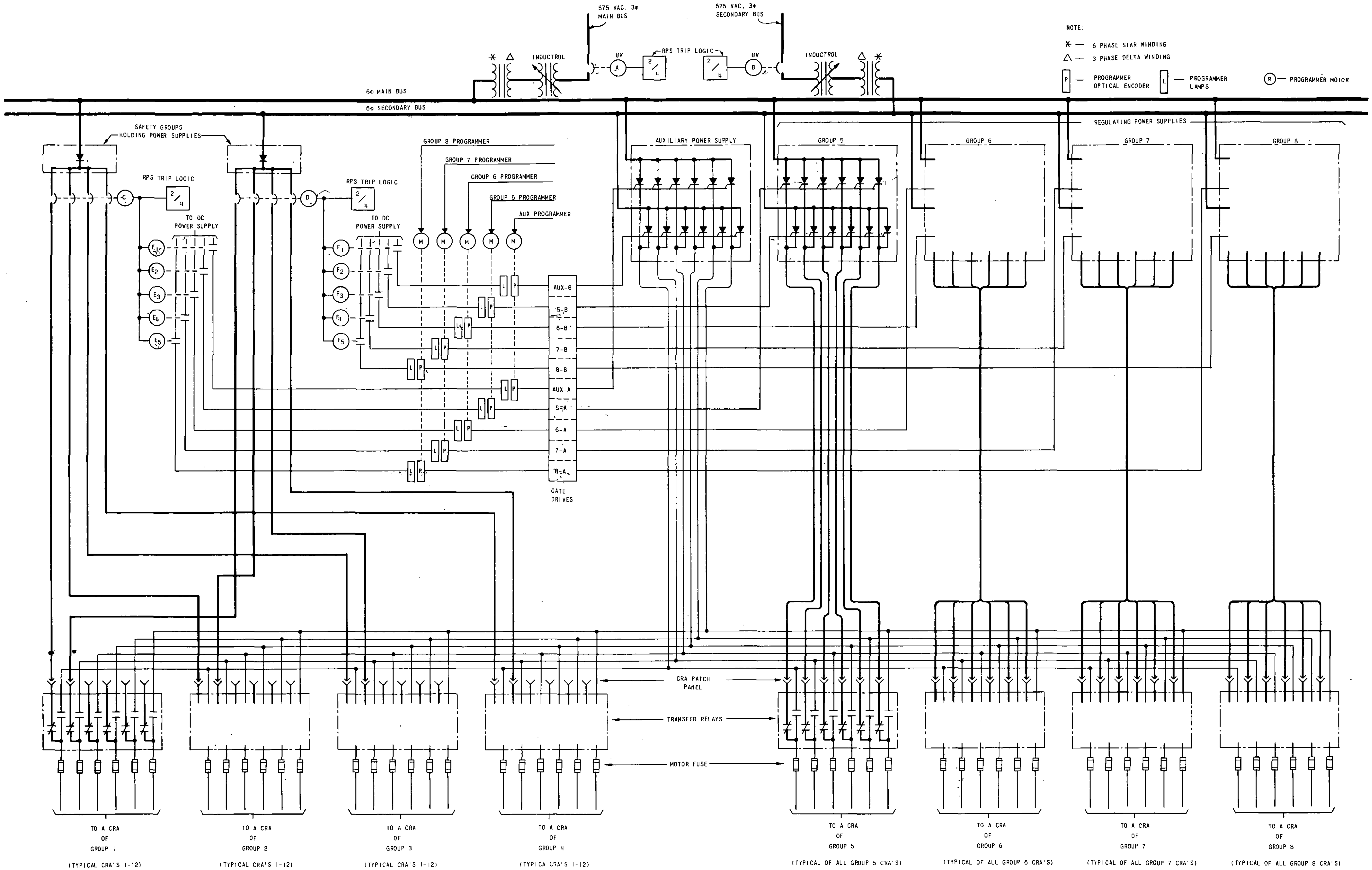
OCONEE NUCLEAR STATION
Figure 7 - 4



AUTOMATIC CONTROL ROD GROUPS – –
TYPICAL WORTH CURVE VERSUS
DISTANCE WITHDRAWN

OCONEE NUCLEAR STATION

Figure 7 - 5

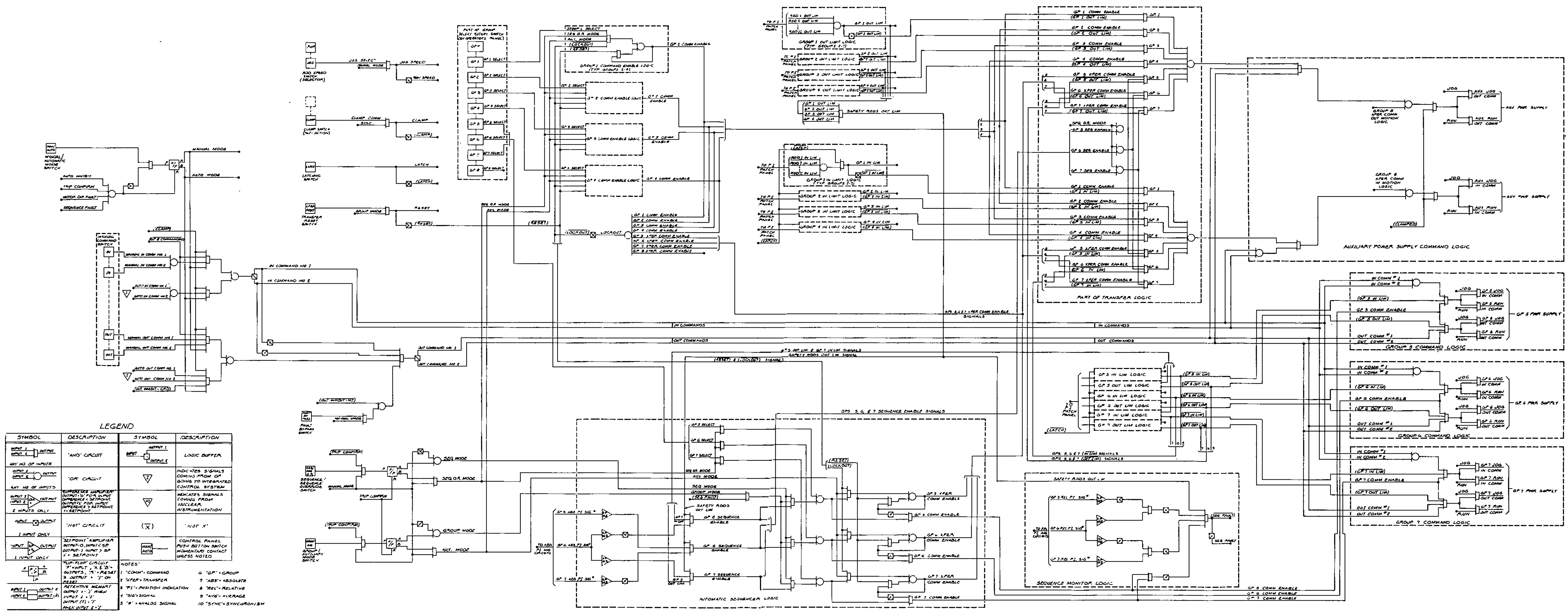


ROD DRIVE CONTROLS

OCONEE NUCLEAR STATION

Figure 7 - 6

Rev. 4 4/20/70

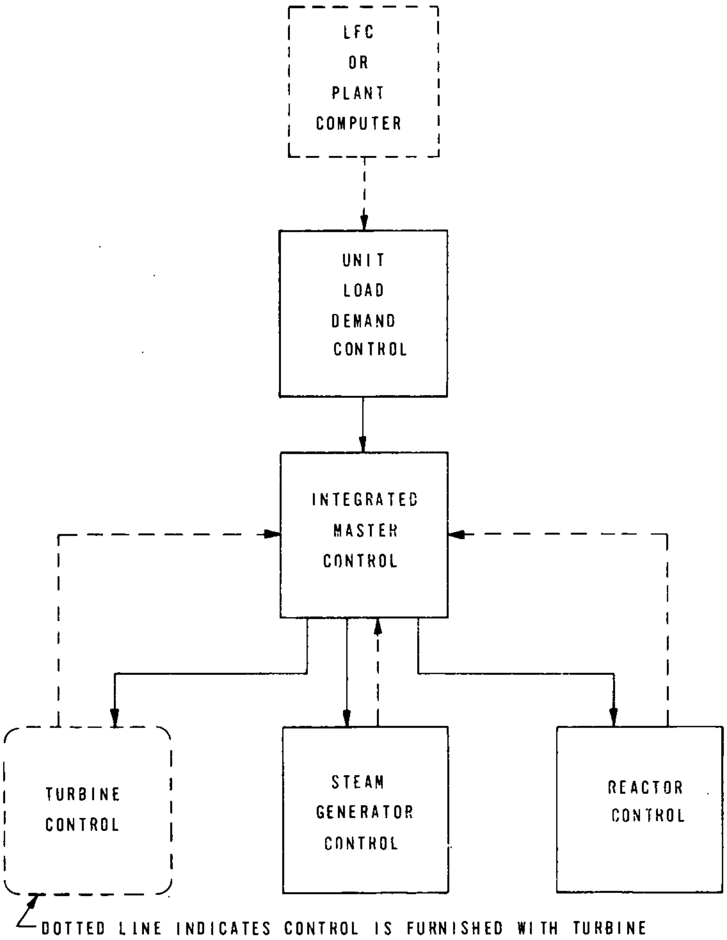


CONTROL ROD DRIVE LOGIC DIAGRAM

OCONEE NUCLEAR STATION

Figure 7 - 6A

(New) Rev. 5 5/25/70



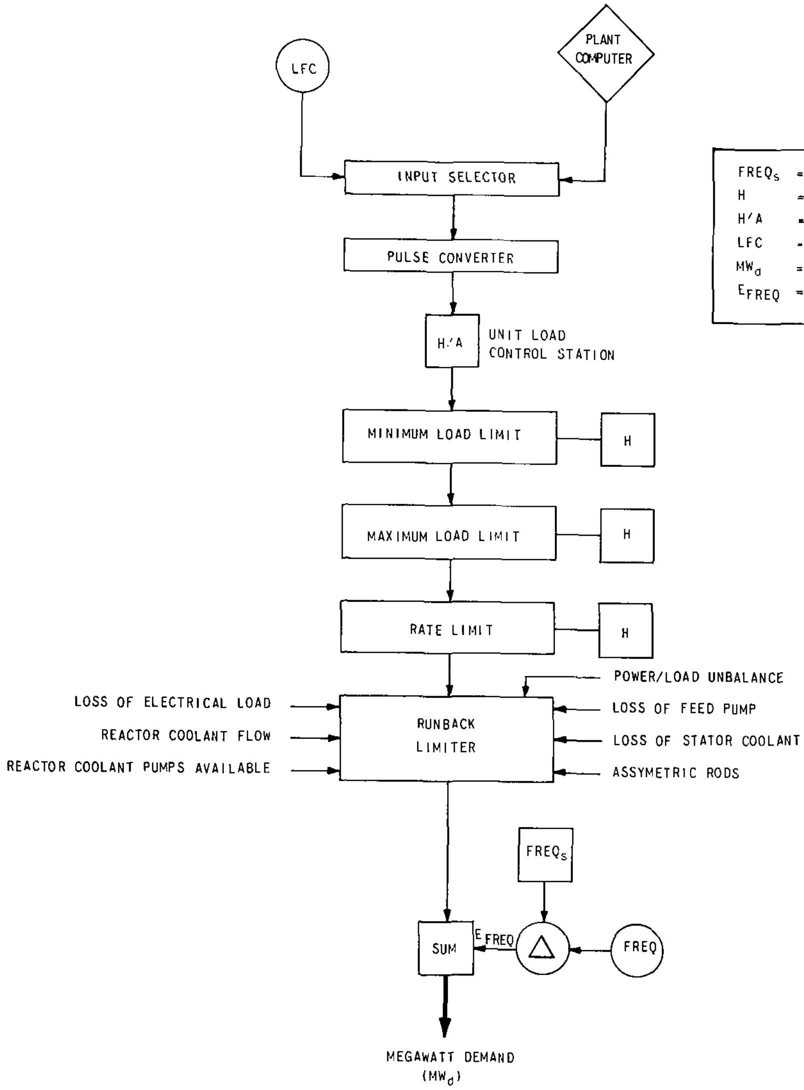
NOTES:

1. TURBINE CONTROL IS FROM THE INTEGRATED MASTER
2. LFC IS CUSTOMER'S LOAD FREQUENCY CONTROL.

INTEGRATED CONTROL SYSTEM

OCONEE NUCLEAR STATION

Figure 7 - 8



LEGEND

$FREQ_S$	=	FREQUENCY SET POINT
H	=	MANUAL SET VALUE
H/A	=	MANUAL/AUTOMATIC CONTROL STATION
LFC	=	CUSTOMER'S LOAD FREQUENCY CONTROL
MW_d	=	MEGAWATT DEMAND
E_{FREQ}	=	FREQUENCY ERROR

UNIT LOAD DEMAND-INTEGRATED CONTROL SYSTEM

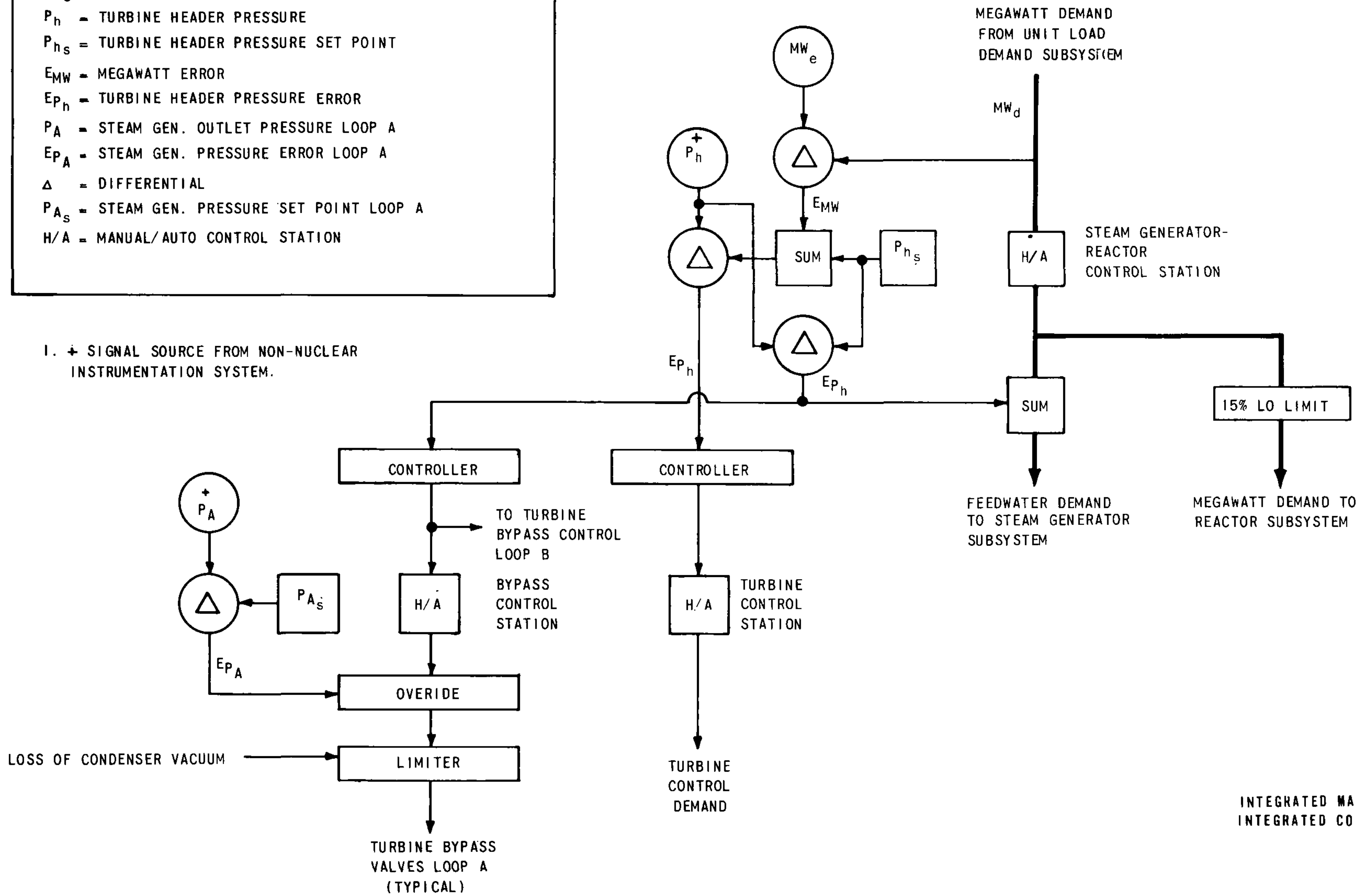
OCONEE NUCLEAR STATION

Figure 7 - 9

LEGEND

MW_d = MEGAWATT DEMAND
MW_e = ACTUAL MEGAWATT GENERATED
P_h = TURBINE HEADER PRESSURE
P_{h_s} = TURBINE HEADER PRESSURE SET POINT
E_{MW} = MEGAWATT ERROR
E_{P_h} = TURBINE HEADER PRESSURE ERROR
P_A = STEAM GEN. OUTLET PRESSURE LOOP A
E_{P_A} = STEAM GEN. PRESSURE ERROR LOOP A
Δ = DIFFERENTIAL
P_{A_s} = STEAM GEN. PRESSURE SET POINT LOOP A
H/A = MANUAL/AUTO CONTROL STATION

1. + SIGNAL SOURCE FROM NON-NUCLEAR INSTRUMENTATION SYSTEM.



INTEGRATED MA
INTEGRATED CO

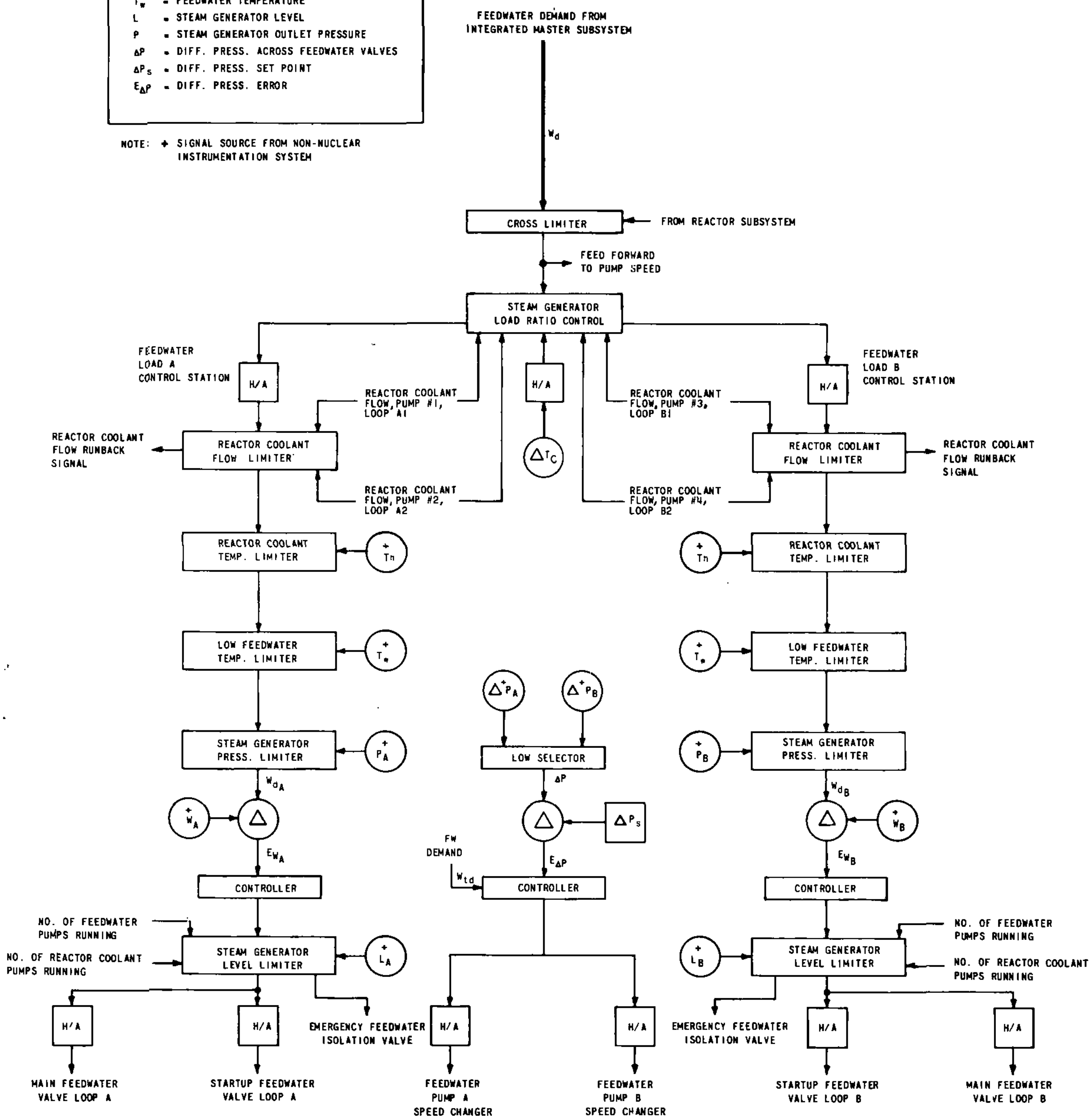
INTEGRATED MASTER-INTEGRATED CONTROL SYSTEM

OCONEE NUCLEAR STATION
Figure 7 - 10

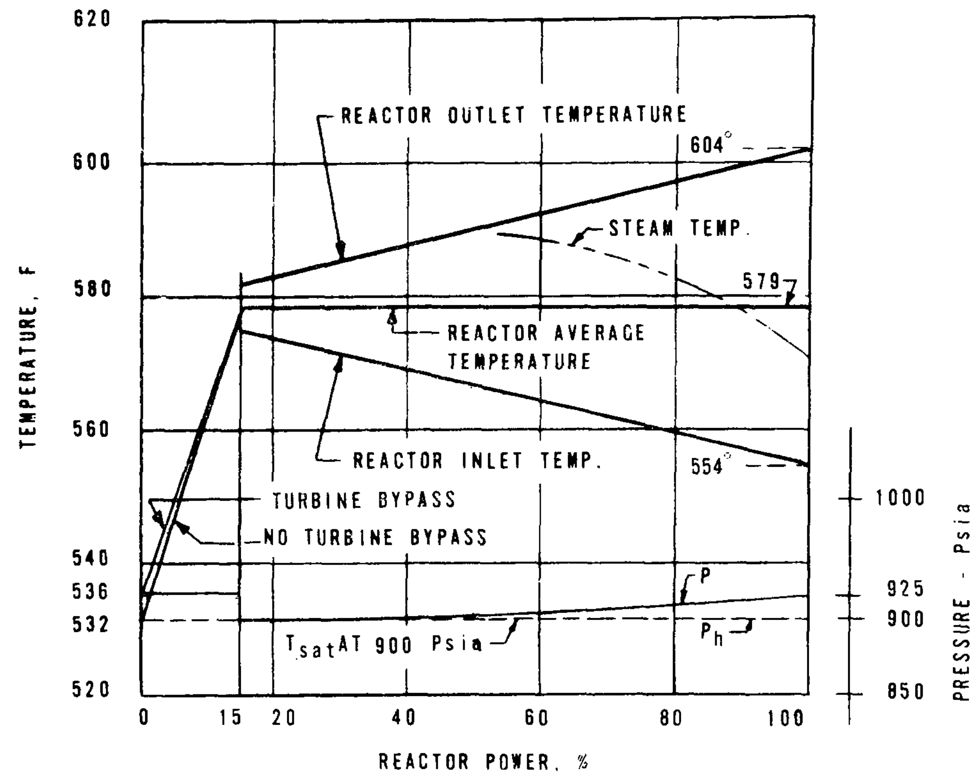
LEGEND

- W = FEEDWATER FLOW
- $\bar{T}c$ = AVERAGE REACTOR INLET TEMP. (COLD LEG)
- Δ = DIFFERENTIAL
- H/A = MANUAL/AUTOMATIC CONTROL STATION
- W_d = FEEDWATER DEMAND
- Tn = REACTOR OUTLET TEMPERATURE (HOT LEG)
- T_w = FEEDWATER TEMPERATURE
- L = STEAM GENERATOR LEVEL
- P = STEAM GENERATOR OUTLET PRESSURE
- ΔP = DIFF. PRESS. ACROSS FEEDWATER VALVES
- ΔP_s = DIFF. PRESS. SET POINT
- $E_{\Delta P}$ = DIFF. PRESS. ERROR

NOTE: + SIGNAL SOURCE FROM NON-NUCLEAR INSTRUMENTATION SYSTEM



STEAM GENERATOR CONTROL - INTEGRATED CONTROL SYSTEM



P STEAM GENERATOR OUTLET PRESSURE

P_h TURBINE HEADER PRESSURE

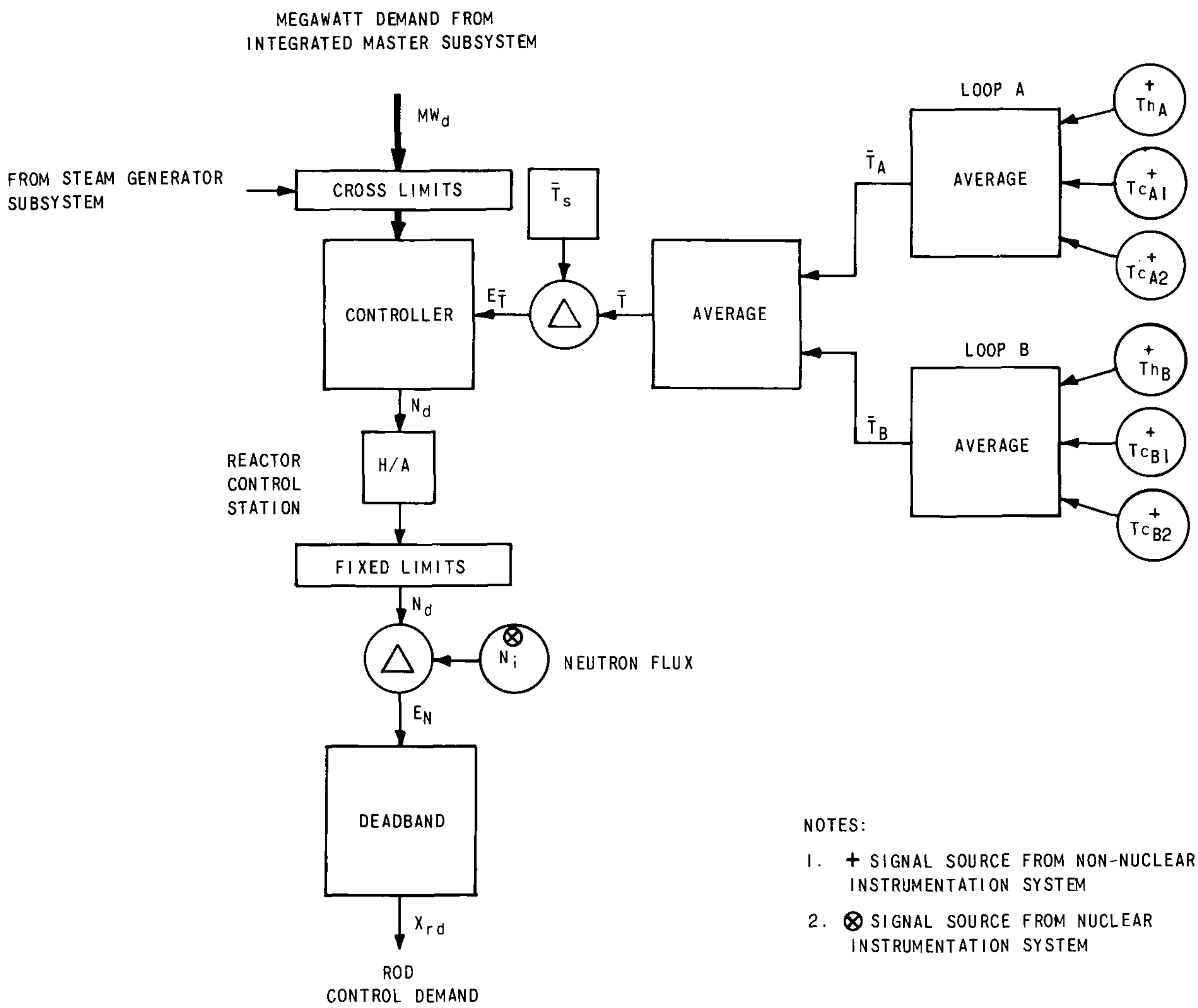
REACTOR AND STEAM TEMPERATURES VERSUS REACTOR POWER

OCONEE NUCLEAR STATION

Figure 7 - 12

LEGEND

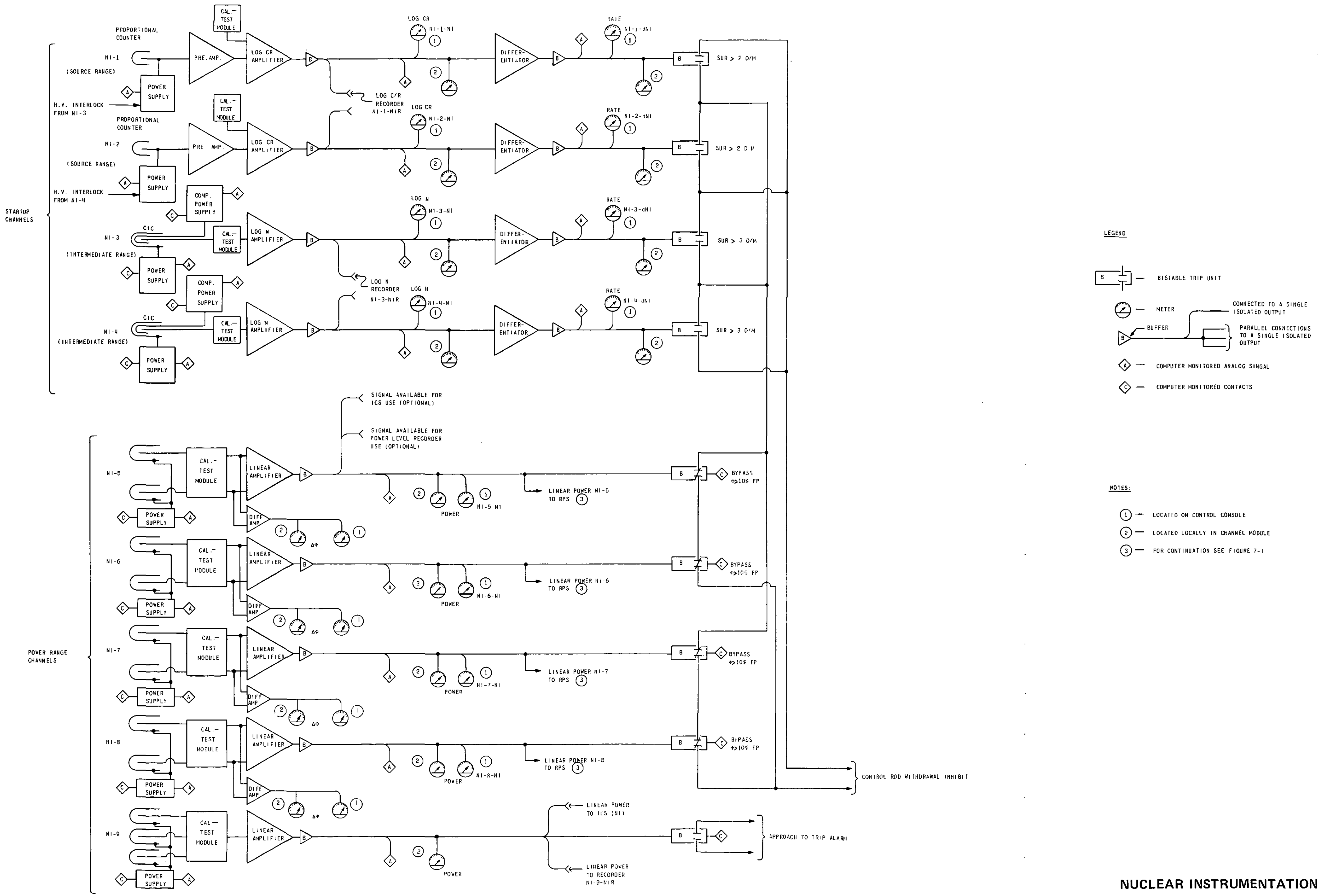
$\bar{T}_s$	-	AVERAGE TEMPERATURE SET POINT
Tc	-	REACTOR COOLANT SYSTEM COLD LEG TEMPERATURE
Th	-	REACTOR COOLANT SYSTEM HOT LEG TEMPERATURE
$\bar{T}$	=	REACTOR AVERAGE COOLANT TEMPERATURE
$E_{\bar{T}}$	-	DEVIATION OF AVERAGE TEMPERATURE FROM SETPOINT
Δ	-	DIFFERENTIAL
H/A	=	MANUAL/AUTOMATIC STATION
N_d	-	REACTOR POWER LEVEL DEMAND
N_i	-	REACTOR POWER LEVEL
E_N	=	REACTOR POWER LEVEL ERROR
X_{rd}	-	ROD CONTROL DEMAND



NOTES:

1. + SIGNAL SOURCE FROM NON-NUCLEAR INSTRUMENTATION SYSTEM
2. ⊗ SIGNAL SOURCE FROM NUCLEAR INSTRUMENTATION SYSTEM

REACTOR CONTROL-INTEGRATED CONTROL SYSTEM

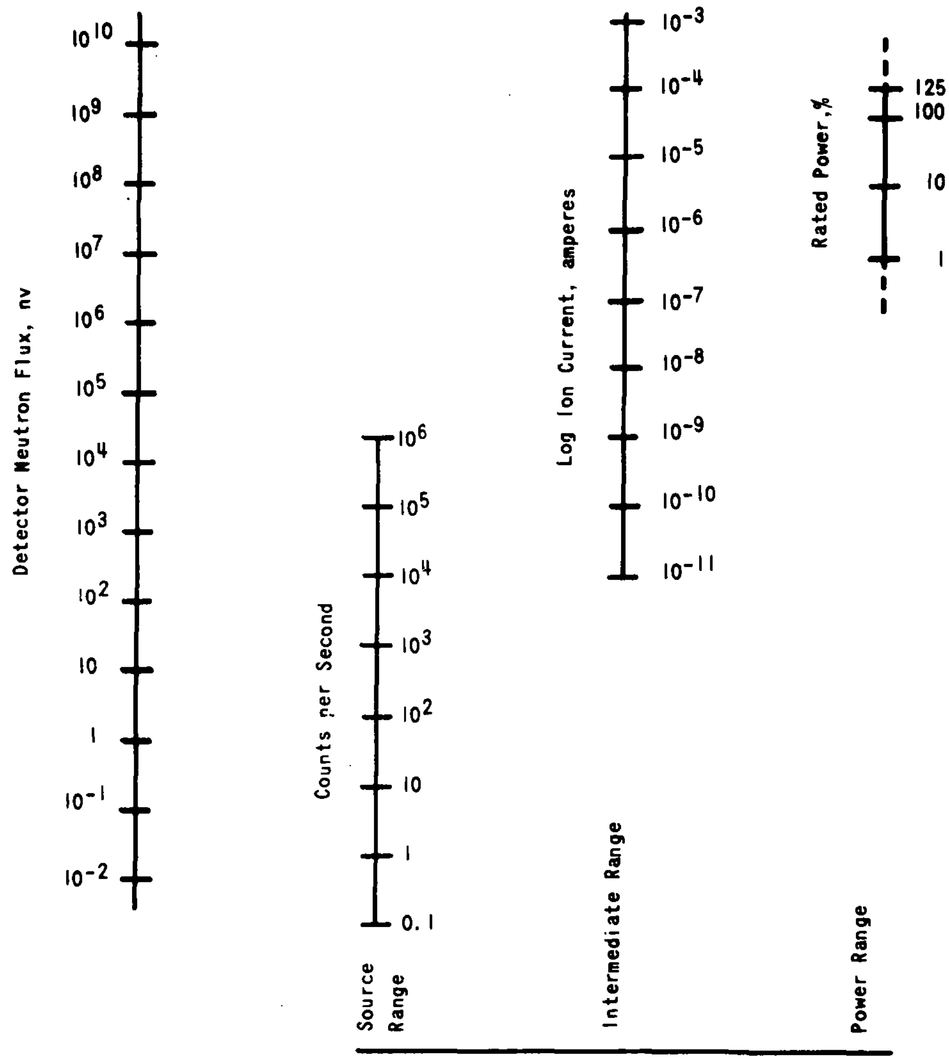


NUCLEAR INSTRUMENTATION SYSTEM

OCONEE NUCLEAR STATION

Figure 7 - 14

Rev. 3 3/16/70

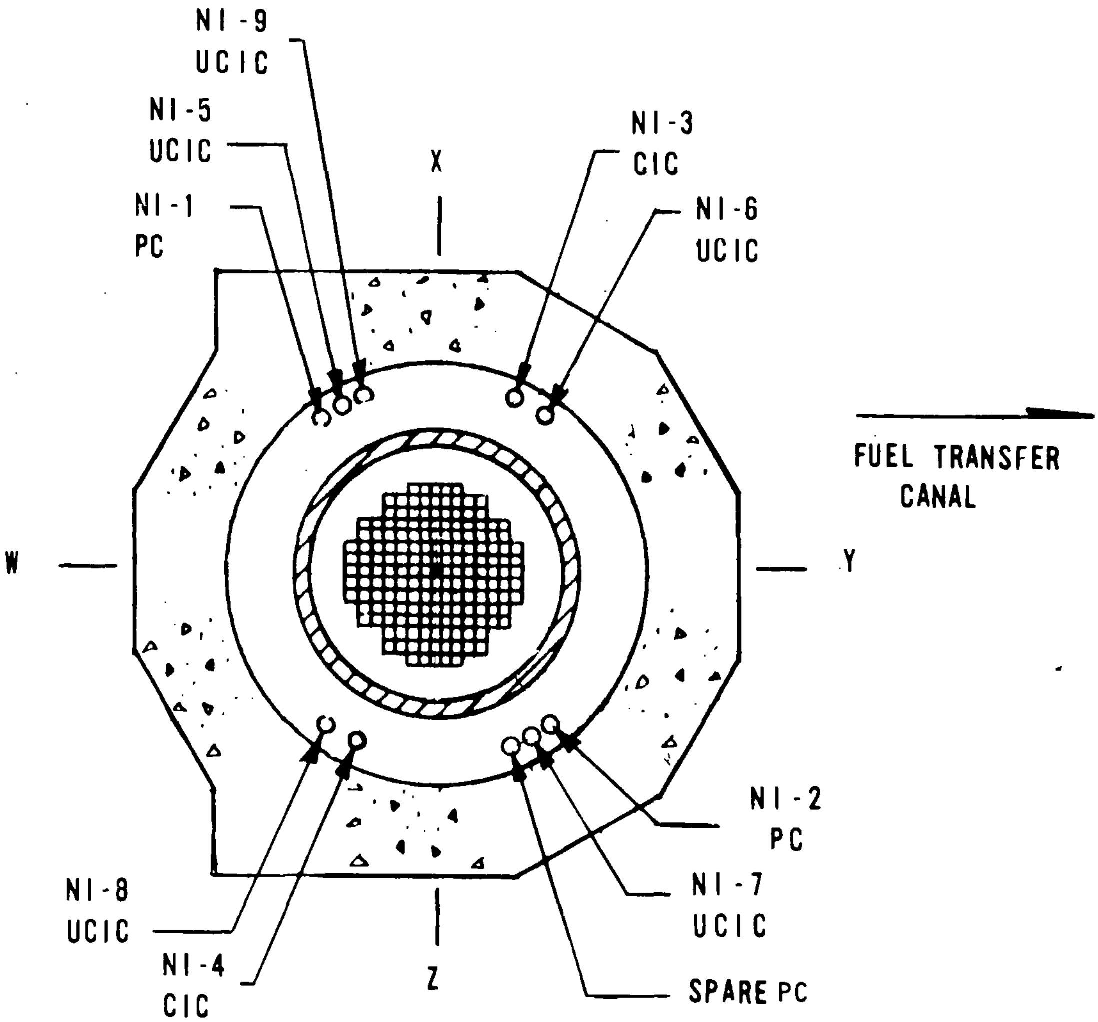


NUCLEAR INSTRUMENTATION FLUX RANGES

OCONEE NUCLEAR STATION

Figure 7 - 15

Rev. 37 06/03/76



LEGEND

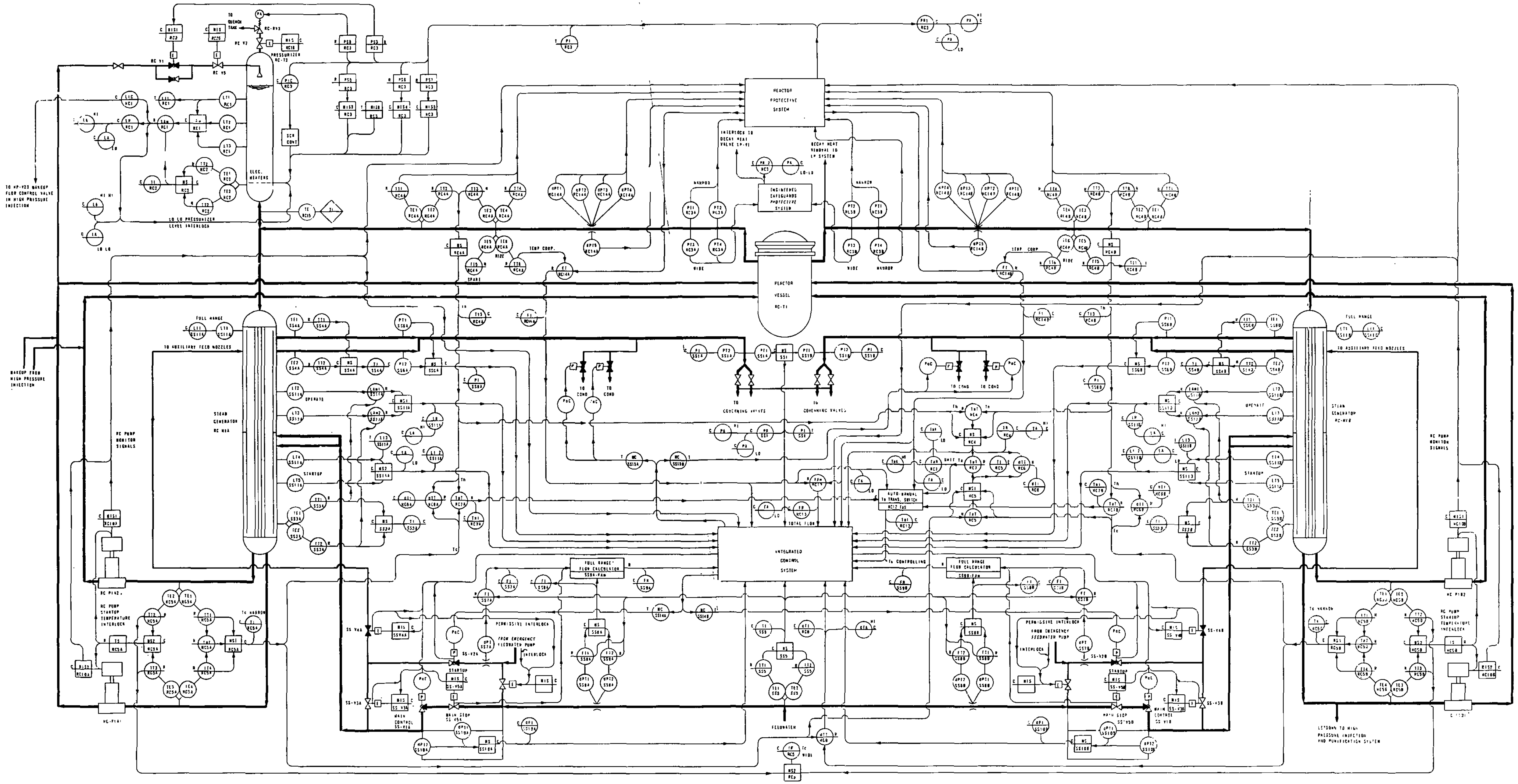
PC - PROPORTIONAL COUNTER - SOURCE RANGE DETECTOR

CIC - COMPENSATED ION CHAMBER - INTERMEDIATE RANGE DETECTOR

UCIC - UNCOMPENSATED ION CHAMBER - POWER RANGE DETECTOR

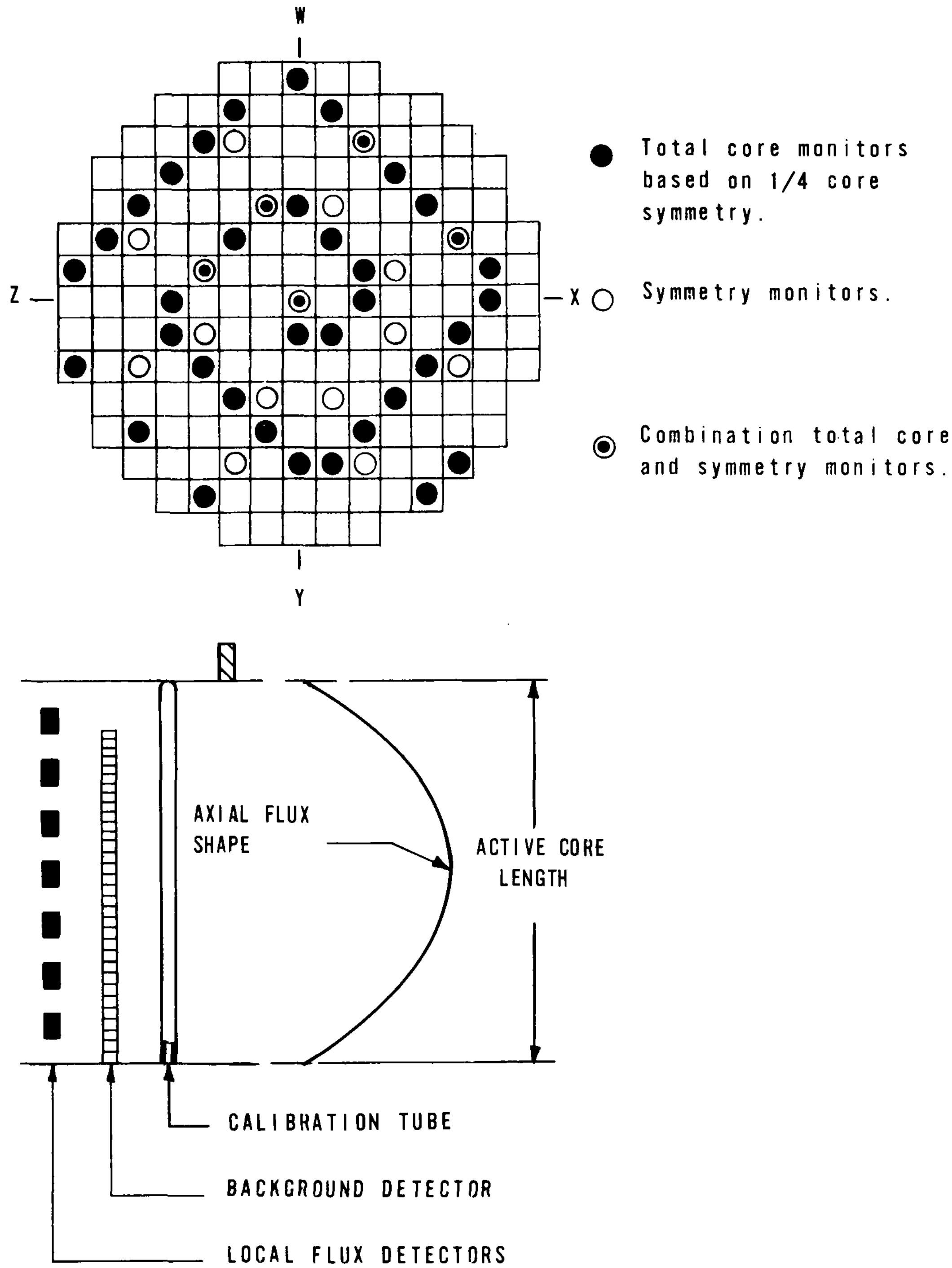
NUCLEAR INSTRUMENTATION
DETECTOR LOCATIONS

OCONEE NUCLEAR STATION
Figure 7 - 16
Rev. 37 06/03/76



REACTOR COOLANT AND STEAM SUPPLY SYSTEMS
NON-NUCLEAR INSTRUMENTATION SCHEMATIC

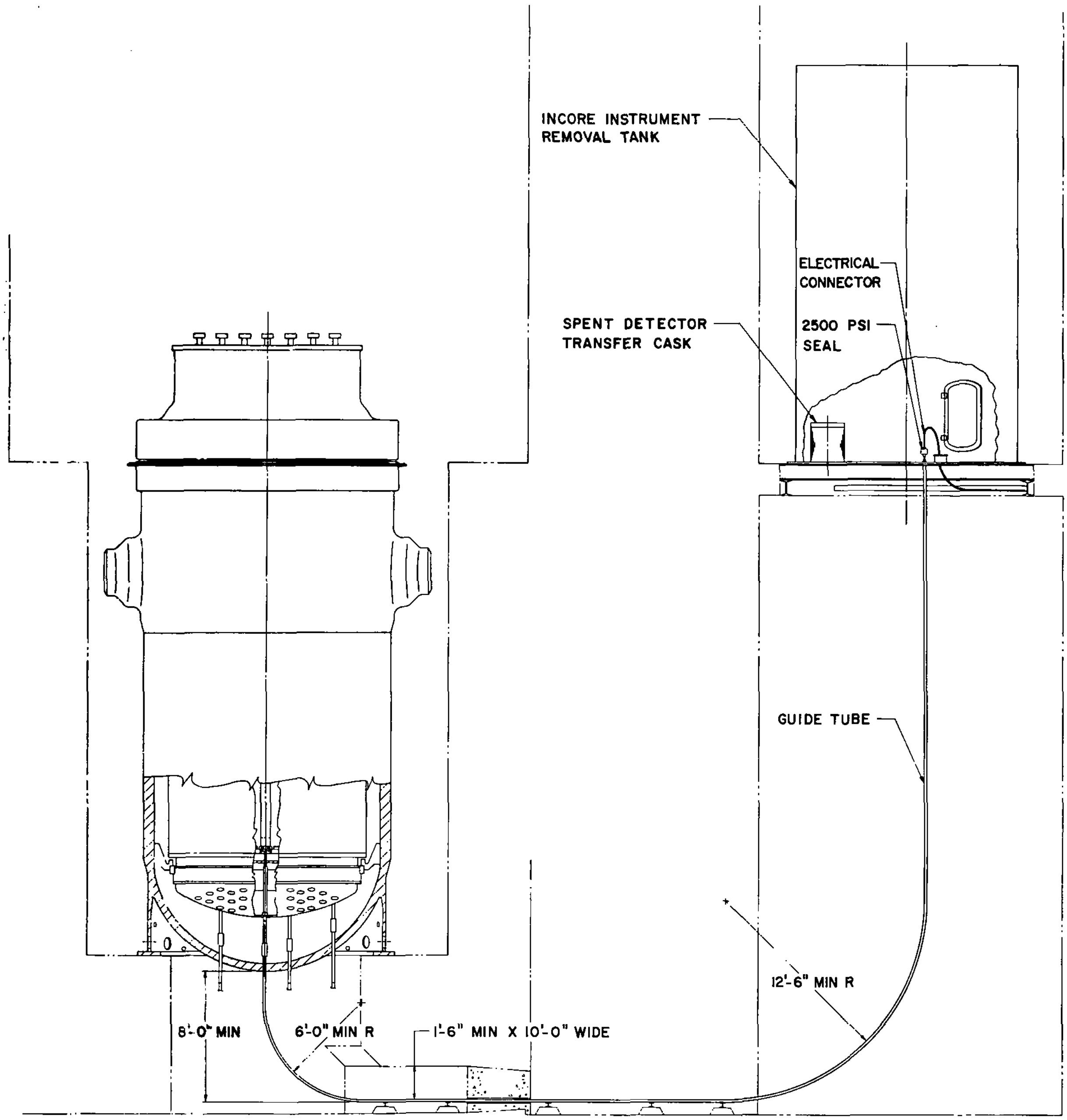
OCONEE NUCLEAR STATION
Figure 7 - 17



INCORE DETECTOR LOCATIONS

OCONEE NUCLEAR STATION

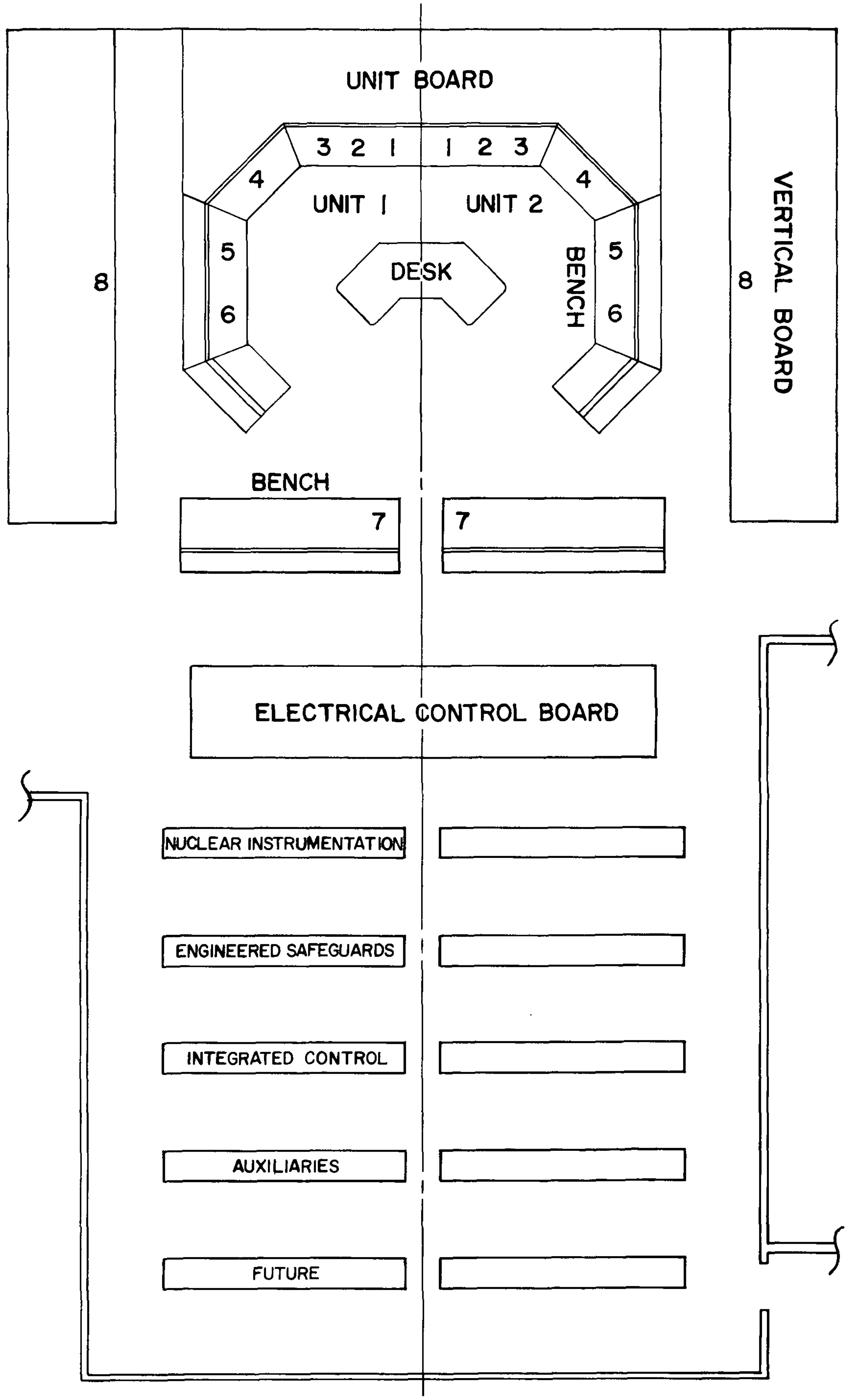
Figure 7 - 18



INCORE MONITORING CHANNEL

OCONEE NUCLEAR STATION

Figure 7 - 19



CONTROL ROOM LAYOUT

OCONEE NUCLEAR STATION
Figure 7 - 20

LIST OF EFFECTIVE PAGES
FSAR APPENDIX 7A

Instrumentation Testing

APPENDIX 7A

OCONEE NUCLEAR STATION

FINAL SAFETY ANALYSIS REPORT

ENVIRONMENTAL AND SEISMIC QUALIFICATION TESTING FOR NUCLEAR INSTRUMENTATION/ REACTOR PROTECTION SYSTEM AND ENGINEERED SAFEGUARDS PROTECTIVE SYSTEM EQUIPMENT

Submitted with FSAR Revision No. 7

July 9, 1970

APPENDIX 7A

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

LIST OF TABLES

7A.1 ENVIRONMENTAL QUALIFICATION TESTING

7A.1.1 INTRODUCTION

The objective of the environmental tests conducted was to confirm that the Nuclear Instrumentation/Reactor Protection System (NI/RPS) and Engineered Safeguard Protective System (ES) equipment met or surpassed the design specification.

7A.1.2 TEST RESULTS

7A.1.2.1 SYSTEM MODULES

Table 7A-1 lists the individual system modules, the test performed and the results of whether the module met or surpassed the specification requirement.

7A.1.2.2 SYSTEM POWER SUPPLIES

System Environment Test

The power supply in the system cabinet was subjected to the following conditions:

- Power Supply Load - 125%
- Line Voltage - Tabulated below
- Control Room Temperature - Tabulated below
- System Cabinet Fan - Off

Line VAC	Ambient Temperature	No Fan - Effect
120	75 F	None
120	110 F	None
140	75 F	4°F margin after 2 hours
140	110 F	0°F margin after 20 mins.

Test results indicated the System Power Supply is the limiting system component when subjected to a combination of temperature, line voltage variation and 125% output load. The temperature of a power transitor rated for 308F was sensed to evaluate the temperature margin. As the environmental test approached this limit the test was terminated. Therefore, margin is defined as the difference between terminating temperature and the rated limit of 308F at the elapsed time.

7A.1.3 CONCLUSION

Each component tested demonstrated its capability to meet design specification and operate for the length of time following a design basis accident.

Table 7A-1

Environmental Test Results

System	Module	Radiation	Temperature	Humidity	Meets or Surpasses Specifications
NI/RPS	Count Rate Amplifier	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Rate of Change	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS & ES	Buffer Amplifier	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Linear Amplifier	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Log Amplifier	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Square Root Extractor	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Linear Bridge	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Signal Convertor Pressure	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Signal Convertor Temperature	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	System Power Supply	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Detector Power Supply	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Auxiliary Power Supply	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Contact Monitor	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS & ES	Bistable	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS & ES	Auxiliary Relay	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Reactor Trip Assembly	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Source Range Test	NA	$\geq$ 110 F	$\geq$ 90% @ 24 hr.	Yes
9. NI/RPS	Intermediate Range Test	NA	$\geq$ 110 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Power Range Test	NA	$\geq$ 110 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Temperature Test	NA	$\geq$ 110 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Pressure Test	NA	$\geq$ 110 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Flow Test	NA	$\geq$ 110 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Contact Monitor Test	NA	$\geq$ 110 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Preamplifier	$\gamma \geq 2x10^4$R accumulated	$\geq$ 160 F	$\geq$ 90% @ 24 hr.	Yes
ES	Logic Buffer	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
ES	Contact Buffer	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
ES	Trip Logic	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
ES	Logic Test	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
ES	Unit Control	NA	$\geq$ 140 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS & ES	Flow Failure Detector	NA	$\geq$ 140 F	NA	Yes
NI/RPS	Cable Input Assemblies	NA	$\geq$ 160 F	$\geq$ 90% @ 24 hr.	Yes
NI/RPS	Cable Input Assembly	NA	$\geq$ 160 F	$\geq$ 90% @ 24 hr.	Yes
9. NI/RPS	Bailey BY dp Transmitter	$\gamma \geq 2x10^4$R accumulated	286	100%	Yes
NI/RPS	Motorola RC56H Pressure Transmitter	$\gamma \geq 2x10^4$R accumulated	286	100%	Yes
NI/RPS	Source Range Detector Assembly	NA	$\geq$ 210 F	$\geq$ 100%	Yes
NI/RPS	Intermediate Range Detector Assembly	NA	$\geq$ 210 F	$\geq$ 100%	Yes
NI/RPS	Power Range Detector Assembly	NA	$\geq$ 210 F	$\geq$ 100%	Yes

NA - Test not applicable to this module.

7A.2 SEISMIC QUALIFICATION TESTING

7A.2.1 GENERAL TEST OBJECTIVE

The objective of the seismic tests of the Nuclear Instrumentation/Reactor Protective System and the Engineered Safeguards Protective System was to demonstrate that these systems will perform normally during and after either a maximum hypothetical earthquake (MHE) or a design earthquake. Separate tests have been conducted for the protective system electronic logic modules and equipment cabinets, for the nuclear instrumentation neutron detectors, and for the pressure transmitters.

7A.2.2 PROTECTIVE SYSTEM EQUIPMENT CABINETS

7A.2.2.1 CABINET TEST OBJECTIVE

The Nuclear Instrumentation/Reactor Protective System and the Engineered Safeguards Protective System electronic equipment cabinets were tested to establish the acceleration levels required for qualification testing of the system electronic modules.

7A.2.2.2 CABINET TEST DESCRIPTION

An eccentric mass vibration exciter was attached to the complete equipment cabinet assembly with the normal number of modules installed. Several accelerometers were attached to the cabinet assembly and these accelerometers monitored to determine the response of the cabinets to the input excitation. Data was taken over a range of exciter input frequencies and the cabinet resonant frequencies and damping factors were determined.

7A.2.2.3 CABINET TEST RESULTS

The electronic cabinet assembly test results were as follows:

Table 7A-2

Cabinet Seismic Test Results

Plane of Motion	Resonant Frequency (H_z)	Damping Coefficient (% of Critical)
Front to Back	11.5	4.4
Side to Side	9.9	31

The building floor response spectra were selected based on cabinet damping and cabinet location in the auxiliary building. The response spectra were then adjusted to account for the possible amplification of the floor motion by the equipment cabinets. These cabinet response spectra were produced by multiplying the building floor response spectra by the ratio of the maximum predicted acceleration within the actual cabinet to the predicted maximum acceleration for a stiff (high resonant frequency) cabinet.

7A.2.2.4 CABINET TEST CONCLUSIONS

Testing of the electronic logic modules has shown that the modules have no resonances below 36 Hertz. Entering the adjusted cabinet response spectra at this frequency yields required module test levels of .51g in the front to back direction and .69g in the side to side direction. The vibration test of the modules must produce accelerations within the modules in excess of these levels.

7A.2.3 PROTECTIVE SYSTEM LOGIC MODULES

7A.2.3.1 MODULE TEST CRITERIA

7A.2.3.1.1

Each module or unit was subjected to a sinusoidal force component individually applied to each of the three planes normal to its usual mounting position. The force was applied over the frequency range of 0 to 40 hertz. The magnitude of the force component was a constant 1 g from 7 to 40 hertz. Below 7 hertz the force component was variable whose magnitude was less than 1 g and was a function of the test facility constraints.

7A.2.3.1.2

The following equipment performance was insured both during and after the testing in each plane of motion.

1. Specified allowable reference error of modules and channels must not change by more than a factor of two during the seismic testing interval.
2. Accuracy of modules and channels must return to within a factor of 1.5 of the specified allowable reference error at termination of test. In addition, all errors must be detectable and correctable through normal test and calibration procedures.
3. All equipment must retain the capability to perform its specified functions both during and after the test.

7A.2.3.1.3

Side to side horizontal motion of the module in its usual mounting position was defined as X plane motion, front to back horizontal motion was defined as Z plane motion, and vertical motion was defined as T plane motion.

7A.2.3.2 MODULE TEST DESCRIPTION

The criteria for this testing requires that all units be dynamically tested in a mode closely duplicating normal channel operation. Test fixtures for all modules duplicated channel or sub-channel mounting and electrical interconnections. All inputs to test fixtures were designed to duplicate or approximate normal application inputs. For those modules which display increased noise sensitivity for certain input ranges, appropriate inputs were chosen to insure operation in the most critical region. All units were pre-calibrated to normal system specifications and functionally checked immediately prior to initiation of test. All pertinent data was recorded during the pre-test checkout of equipment.

During the actual test interval (period of excitation) all outputs and selected critical points were continuously monitored with appropriate test instruments. Where necessary, module and channel tripping functions were evaluated during the test period. All pertinent data was recorded and documented during this period of excitation.

Following completion of the excitation period, a complete post-test checkout of the units was performed. All pertinent data was recorded and documented in the same format as the pre-test information.

The following components were tested:

Nuclear Instrumentation/Reactor Protective System

- Preamplifier
- Count Rate Amplifier
- Log Amplifier
- Linear Amplifier
- Buffer Amplifier
- Rate of Change
- Linear Bridge
- Signal Converter
- Contact Monitor
- Square Root Extractor
- Bistable
- Power Range Test
- Intermediate Range Test
- Source Range Test
- Temperature Test
- Pressure Test
- Flow Test
- Contact Monitor Test
- System Power Supply
- Detector Power Supply
- Auxiliary Power Supply
- Auxiliary Relay
- Fan Failure Detector
- Reactor Trip
- Cable Input Assemblies
- Cable Input Assembly

Engineered Safeguards Protective System

Logic Buffer
Contact Buffer
Trip Logic
Logic Test
Unit Control

Peripheral Equipment

Type "KQ" Pressure Xmtr.
Pressure Switch
Switch Module
Type "BY" Pressure Xmtr.

7A.2.3.3 MODULE TEST RESULTS

7A.2.3.3.1 Explanation of Tabulated Data

Tabulated data for the entire testing sequence appears in Tables 7A-3a through 7A-5b. Tables 7A-3a through 7A-4d are divided into seven columns which indicate the effects of the test, if any, on each module. Tables 7A-3a through 7A-3d give the results during the actual test interval, while Tables 7A-4a through 7A-4d give the post-test results. Tables 7A-5a and 7A-5b give a composite evaluation of the effects on typical channels. These tables are divided into three columns which indicate the effects associated with each channel.

The following is a list, along with a brief explanation, of the various column headings found in the tables.

1. ΔV_o (D.C.) - The maximum D.C. voltage error observed at the output or selected critical point of a given unit. This error is referenced to the value obtained in the pre-test check.
2. ΔV_o (Noise) - This is the maximum magnitude of increase in the noise level of an output or selected critical point of a given unit. This value is also referenced to conditions existing in the pre-test check.
3. % Allowable D.C. Error - A ratio, expressed as a percentage, as follows:

$$\% = \frac{\Delta V_o \text{ (D.C.)}}{BA} 100$$

where,

> B = Factor of two for test interval or a factor of 1.5 for the post-test check
>
> A = Specified allowable reference error of the module, unit, or channel

4. % Allowable Noise - A ratio, expressed as a percentage, as follows:

$$\% = \frac{\Delta V_o \text{ (Noise)}}{BA} (100);$$

where,

> B = Factor of two for test interval or a factor of 1.5 for the post-test check
>
> A = Specified allowable reference noise of the module, unit, or channel

5. Number of Contact Failures - Number of false contact movements, within a relay or switch, as a result of the test.
6. Number of Function Failures - Number of times a module or unit fails to perform its required function as a result of the test.

7A.2.3.3.2 Determination of Composite (X, Y, Z) Plane Error

Error resulting from seismic testing is proportional to the magnitude and direction of the applied force vector. Since this proportionality exists, it is desirable to reflect this relationship in any computation of total error as a result of composite plane motion. This relationship has been taken into account in obtaining total error from the individual plane error. The relations used are as follows:

1. $\Delta V_o \quad \alpha\vec{F}$ where $\vec{F}$ = the vector force
2. ΔV_{o_t} (Total) $\alpha\vec{F}_t$ (Total) where $\vec{F}_t$ (Total) - the vector summation of forces
3. $/\vec{F}_t/ \; = \sqrt{F_X^2 + F_Y^2 + F_Z^2}$
4. $/\Delta V_{o_t}^2/ \; = \sqrt{\Delta V_{o_X}^2 + \Delta V_{o_Y}^2 + \Delta V_{o_Z}^2}$

Table 7A-3a

Module Test Results - X Plane

Name of Unit	ΔV_o (D.C.)	ΔV_o (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise	Number of Contact Failures	Number of Function Failures
Preamplifier	0 mv	0 mv	0	0	-	None
Count Rate Amplifier	1 mv	0 mv	.5	0	None	None
Log Amplifier	1 mv	0 mv	.5	0	-	None
Linear Amplifier	2 mv	0 mv	10	0	-	None
Buffer Amplifier	1 mv	0 mv	5	0	-	None
Rate of Change	1 mv	0 mv	5	0	-	None
Linear Bridge	0 mv	0 mv	0	0	None	None
Signal Converter	6 mv	0 mv	30	0	None	None
Contact Monitor	1 mv	0 mv	1	0	-	None
Square Root Extractor	1 mv	0 mv	5	0	-	None
Bistable	0 mv	-	0	-	None	None
Power Range Test	0 mv	0 mv	0	0	None	None
Intermediate Range Test	0 mv	0 mv	0	0	None	None
Source Range Test	3 mv	0 mv	30	0	None	None
Temperature Test	0 mv	0 mv	0	0	None	None
Pressure Test	-	-	-	-	None	None
Flow Test	3 mv	0 mv	30	0	None	None
Contact Monitor Test	-	-	-	-	None	None
System Power Supply	0 mv	0 mv	0	0	None	None
Detector Power Supply	0 mv	40 mv	0	20	None	None
Auxiliary Power Supply	0 mv	60 mv	0	30	None	None
Auxiliary Relay	-	-	-	-	None	None
Fan Failure Detector	-	-	-	-	None	None
Switch Module	-	-	-	-	None	None
Reactor Trip	-	-	-	-	None	None
Logic Buffer	-	-	-	-	None	None
Contact Buffer	-	-	-	-	None	None
Trip Logic	-	-	-	-	None	None
Safeguards Logic Test	-	-	-	-	None	None
Unit Control	-	-	-	-	None	None
Type "KQ" XMTR.	2 mv	0 mv	2.2	0	-	None
Pressure Switch	-	-	-	-	None	None
Type "BY" XMTR.*	250 mv	0 mv	125	0	-	None

*See next page.

Table 7A-3a (Cont'd)

The asterisk beside the "BY" Pressure Transmitter indicates that unit failed the acceptance criteria, but was qualified to 20 Hz in the horizontal mode (X and Z Planes) of test. The unit is, however, qualified to 40 Hz in the vertical mode (Y Plane) of test.

Since excitations of the magnitude applied during the test at frequencies above 20 Hz are not predicted for the Maximum Hypothetical Earthquake or the Design Earthquake, this equipment is considered satisfactorily qualified.

Revised Data for Type "BY" Pressure Transmitter

Z Plane (0 to 20 Hz) ΔV (D.C.)	ΔV (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise
80 mv	0	40%	0%
X Plane (0 to 20 Hz)			
15 mv	0	7.4%	0%
X,Y,Z Composite (0 to 20 Hz)			
87 mv	30 mv	43.5%	60%

Table 7A-3b

Module Test Results - Y Plane

Name of Unit	ΔV_o (D.C.)	ΔV_o (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise	Number of Contact Failures	Number of Function Failures
Preamplifier	0 mv	0 mv	0	0	-	None
Count Rate Amplifier	3 mv	0 mv	1.5	0	None	None
Log Amplifier	0 mv	0 mv	0	0	-	None
Linear Amplifier	0 mv	6 mv	0	3	-	None
Buffer Amplifier	1 mv	0 mv	5	0	-	None
Rate of Change	1 mv	0 mv	5	0	-	None
Linear Bridge	0 mv	0 mv	0	0	None	None
Signal Converter	5 mv	0 mv	25	0	None	None
Contact Monitor	2 mv	0 mv	2	0	-	None
Square Root Extractor	1.5 mv	0 mv	7.5	0	-	None
Bistable	0 mv	-	0	-	None	None
Power Range Test	0 mv	0 mv	0	0	None	None
Intermediate Range Test	0 mv	0 mv	0	0	None	None
Source Range Test	1 mv	0 mv	10	0	None	None
Temperature Test	2 mv	0 mv	33	0	None	None
Pressure Test	-	-	-	-	None	None
Flow Test	2 mv	0 mv	20	0	None	None
Contact Monitor Test	-	-	-	-	None	None
System Power Supply	20 mv	100 mv	6.7	50	None	None
Detector Power Supply	0 mv	0 mv	0	0	None	None
Auxiliary Power Supply	0 mv	0 mv	0	0	None	None
Auxiliary Relay	-	-	-	-	None	None
Fan Failure Detector	-	-	-	-	None	None
Switch Module	-	-	-	-	None	None
Reactor Trip	-	-	-	-	None	None
Logic Buffer	-	-	-	-	None	None
Contact Buffer	-	-	-	-	None	None
Trip Logic	-	-	-	-	None	None
Safeguards Logic Test	-	-	-	-	None	None
Unit Control	-	-	-	-	None	None
Type "KQ" XMTR.	6 mv	0 mv	6.7	0	-	None
Pressure Switch	-	-	-	-	None	None
Type "BY" XMTR.	60 mv	30 mv	30	75	-	None

Table 7A-3c

Module Test Results - Z Plane

Name of Unit	ΔV_o (D.C.)	ΔV_o (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise	Number of Contact Failures	Number of Function Failures
Preamplifier	0 mv	0 mv	0	0	-	None
Count Rate Amplifier	5 mv	0 mv	2.5	0	None	None
Log Amplifier	4 mv	0 mv	2	0	-	None
Linear Amplifier	2 mv	0 mv	10	0	-	None
Buffer Amplifier	1 mv	0 mv	5	0	-	None
Rate of Change	0 mv	0 mv	0	0	-	None
Linear Bridge	0 mv	0 mv	0	0	None	None
Signal Converter	6 mv	0 mv	30	0	None	None
Contact Monitor	2 mv	0 mv	2	0	-	None
Square Root Extractor	0 mv	0 mv	0	0	-	None
Bistable	5 mv	-	14.7	-	None	None
Power Range Test	0 mv	0 mv	0	0	None	None
Intermediate Range Test	0 mv	0 mv	0	0	None	None
Source Range Test	2 mv	0 mv	20	0	None	None
Temperature Test	0 mv	0 mv	0	0	None	None
Pressure Test	-	-	-	-	None	None
Flow Test	0 mv	0 mv	0	0	None	None
Contact Monitor Test	-	-	-	-	None	None
System Power Supply	150 mv	0 mv	50	0	None	None
Detector Power Supply	100 mv	80 mv	5.5	40	None	None
Auxiliary Power Supply	0 mv	29 mv	0	14.5	None	None
Auxiliary Relay	-	-	-	-	None	None
Fan Failure Detector	-	-	-	-	None	None
Switch Module	-	-	-	-	None	None
Reactor Trip	-	-	-	-	None	None
Logic Buffer	-	-	-	-	None	None
Contact Buffer	-	-	-	-	None	None
Trip Logic	-	-	-	-	None	None
Safeguards Logic Test	-	-	-	-	None	None
Unit Control	-	-	-	-	None	None
Type "KQ" XMTR.	13 mv	15 mv	14.5	75	-	None
Pressure Switch	-	-	-	-	None	None
Type "BY" XMTR.*	160 mv	20 mv	80	50	-	None

*See next page.

Table 7A-3c (Cont'd)

The asterisk beside the "BY" Pressure Transmitter indicates that unit failed the acceptance criteria, but was qualified to 20 Hz in the horizontal mode (X and Z Planes) of test. The unit is, however, qualified to 40 Hz in the vertical mode (Y Plane) of test.

Since excitations of the magnitude applied during the test at frequencies above 20 Hz are not predicted for the Maximum Hypothetical Earthquake or the Design Earthquake, this equipment is considered satisfactorily qualified.

Revised Data for Type "BY" Pressure Transmitter

Z Plane (0 to 20 Hz) ΔV (D.C.)	ΔV (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise
80 mv	0	40%	0%
X Plane (0 to 20 Hz)			
15 mv	0	7.4%	0%
X,Y,Z Composite (0 to 20 Hz)			
87 mv	30 mv	43.5%	60%

Table 7A-3d

Module Test Results - Resultant Error from X, Y, and Z Planes

Name of Unit	ΔV_o (D.C.)	ΔV_o (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise	Number of Contact Failures	Number of Function Failures
Preamplifier	0 mv	0 mv	0	0	-	None
Count Rate Amplifier	6 mv	0 mv	6	0	None	None
Log Amplifier	4 mv	0 mv	2	0	-	None
Linear Amplifier	3 mv	6 mv	15	3	-	None
Buffer Amplifier	2 mv	0 mv	10	0	-	None
Rate of Change	1 mv	0 mv	5	0	-	None
Linear Bridge	0 mv	0 mv	0	0	None	None
Signal Converter	10 mv	0 mv	50	0	None	None
Contact Monitor	3 mv	0 mv	3	0	-	None
Square Root Extractor	2 mv	0 mv	10	0	-	None
Bistable	5 mv	-	14.7	-	None	None
Power Range Test	0 mv	0 mv	0	0	None	None
Intermediate Range Test	0 mv	0 mv	0	0	None	None
Source Range Test	4 mv	0 mv	40	0	None	None
Temperature Test	2 mv	0 mv	33	0	None	None
Pressure Test	-	-	-	-	None	None
Flow Test	4 mv	0 mv	40	0	None	None
Contact Monitor Test	-	-	-	-	None	None
System Power Supply	151 mv	100 mv	50.4	50	None	None
Detector Power Supply	100 mv	89 mv	5.5	44.5	None	None
Auxiliary Power Supply	0 mv	67 mv	0	33.5	None	None
Auxiliary Relay	-	-	-	-	None	None
Fan Failure Detector	-	-	-	-	None	None
Switch Module	-	-	-	-	None	None
Reactor Trip	-	-	-	-	None	None
Logic Buffer	-	-	-	-	None	None
Contact Buffer	-	-	-	-	None	None
Trip Logic	-	-	-	-	None	None
Safeguards Logic Test	-	-	-	-	None	None
Unit Control	-	-	-	-	None	None
Type "KQ" XMTR.	14 mv	15 mv	15.6	75	-	None
Pressure Switch	-	-	-	-	None	None
Type "BY" XMTR.*	303 mv	36 mv	151	90	-	None

*See next page.

Table 7A-3d (Cont'd)

The asterisk beside the "BY" Pressure Transmitter indicates that unit failed the acceptance criteria, but was qualified to 20 Hz in the horizontal mode (X and Z Planes) of test. The unit is, however, qualified to 40 Hz in the vertical mode (Y Plane) of test.

Since excitations of the magnitude applied during the test at frequencies above 20 Hz are not predicted for the Maximum Hypothetical Earthquake or the Design Earthquake, this equipment is considered satisfactorily qualified.

Revised Data for Type "BY" Pressure Transmitter

Z Plane (0 to 20 Hz) ΔV (D.C.)	ΔV (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise
80 mv	0	40%	0%
X Plane (0 to 20 Hz)			
15 mv	0	7.4%	0%
X,Y,Z Composite (0 to 20 Hz)			
87 mv	30 mv	43.5%	60%

Table 7A-4a

Module Post Test Results - X Plane

Name of Unit	ΔV_o (D.C.)	ΔV_o (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise	Number of Contact Failures	Number of Function Failures
Preamplifier	0 mv	0 mv	0	0	-	None
Count Rate Amplifier	1 mv	0 mv	.6	0	None	None
Log Amplifier	2 mv	0 mv	1.3	0	-	None
Linear Amplifier	6 mv	15 mv	40	10	-	None
Buffer Amplifier	1 mv	0 mv	6.7	0	-	None
Rate of Change	1 mv	0 mv	6.7	0	-	None
Linear Bridge	0 mv	0 mv	0	0	None	None
Signal Converter	7 mv	0 mv	46	0	None	None
Contact Monitor	1 mv	0 mv	1.3	0	-	None
Square Root Extractor	1 mv	0 mv	6.7	0	-	None
Bistable	1 mv	-	4	-	None	None
Power Range Test	0 mv	8 mv	0	53	None	None
Intermediate Range Test	0 mv	0 mv	0	0	None	None
Source Range Test	0 mv	0 mv	0	0	None	None
Temperature Test	0 mv	0 mv	0	0	None	None
Pressure Test	0 mv	0 mv	0	0	None	None
Flow Test	1 mv	0 mv	13	0	None	None
Contact Monitor Test	1 mv	0 mv	13	0	None	None
System Power Supply	0 mv	0 mv	0	0	None	None
Detector Power Supply	100 mv	0 mv	7.4	0	None	None
Auxiliary Power Supply	0 mv	0 mv	0	0	None	None
Auxiliary Relay	-	-	-	-	None	None
Fan Failure Detector	-	-	-	-	None	None
Switch Module	-	-	-	-	None	None
Reactor Trip	-	-	-	-	None	None
Logic Buffer	-	-	-	-	None	None
Contact Buffer	-	-	-	-	None	None
Trip Logic	-	-	-	-	None	None
Safeguards Logic Test	-	-	-	-	None	None
Unit Control	-	-	-	-	None	None
Type "KQ" XMTR.	10 mv	0 mv	15	0	-	None
Pressure Switch	-	-	-	-	None	None
Type "BY" XMTR.	15 mv	0 mv	10	0	-	None

Table 7A-4b

Module Post Test Results - Y Plane

Name of Unit	ΔV_o (D.C.)	ΔV_o (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise	Number of Function Failures	Number of Function Failures
Preamplifier	0 mv	0 mv	0	0	-	None
Count Rate Amplifier	2 mv	0 mv	1.3	0	None	None
Log Amplifier	0 mv	0 mv	0	0	-	None
Linear Amplifier	3 mv	0 mv	20	0	-	None
Buffer Amplifier	0 mv	0 mv	0	0	-	None
Rate of Change	2 mv	0 mv	13	0	-	None
Linear Bridge	1 mv	0 mv	67	0	None	None
Signal Converter	2 mv	0 mv	13	0	None	None
Contact Monitor	1 mv	0 mv	1.3	0	-	None
Square Root Extractor	0 mv	0 mv	0	0	-	None
Bistable	1 mv	-	4	-	None	None
Power Range Test	0 mv	0 mv	0	0	None	None
Intermediate Range Test	0 mv	0 mv	0	0	None	None
Source Range Test	0 mv	0 mv	0	0	None	None
Temperature Test	0 mv	0 mv	0	0	None	None
Pressure Test	0 mv	0 mv	0	0	None	None
Flow Test	0 mv	0 mv	0	0	None	None
Contact Monitor Test	0 mv	0 mv	0	0	None	None
System Power Supply	10 mv	0 mv	4.5	0	None	None
Detector Power Supply	0 mv	0 mv	0	0	None	None
Auxiliary Power Supply	0 mv	0 mv	0	0	None	None
Auxiliary Relay	-	-	-	-	None	None
Fan Failure Detector	-	-	-	-	None	None
Switch Module	-	-	-	-	None	None
Reactor Trip	-	-	-	-	None	None
Logic Buffer	-	-	-	-	None	None
Contact Buffer	-	-	-	-	None	None
Trip Logic	-	-	-	-	None	None
Safeguards Logic Test	-	-	-	-	None	None
Unit Control	-	-	-	-	None	None
Type "KQ" XMTR.	12 mv	0 mv	17	0	-	None
Pressure Switch	-	-	-	-	None	None
Type "BY" XMTR.	17 mv	0 mv	11	0	-	None

Table 7A-4c

Module Post Test Results - Z Plane

Name of Unit	ΔV_o (D.C.)	ΔV_o (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise	Number of Contact Failures	Number of Function Failures
Preamplifier	0 mv	0 mv	0	0	-	None
Count Rate Amplifier	2 mv	0 mv	1.3	0	None	None
Log Amplifier	2 mv	0 mv	1.3	0	-	None
Linear Amplifier	1 mv	0 mv	6.7	0	-	None
Buffer Amplifier	2 mv	0 mv	13	0	-	None
Rate of Change	1 mv	0 mv	6.7	0	-	None
Linear Bridge	0 mv	0 mv	0	0	None	None
Signal Converter	11 mv	0 mv	73	0	None	None
Contact Monitor	2 mv	0 mv	2.7	0	-	None
Square Root Extractor	0 mv	0 mv	0	0	-	None
Bistable	1 mv	-	4	-	None	None
Power Range Test	1 mv	0 mv	22	0	None	None
Intermediate Range Test	0 mv	0 mv	0	0	None	None
Source Range Test	0 mv	0 mv	0	0	None	None
Temperature Test	0 mv	0 mv	0	0	None	None
Pressure Test	0 mv	0 mv	0	0	None	None
Flow Test	1 mv	0 mv	13	0	None	None
Contact Monitor Test	1 mv	0 mv	13	0	None	None
System Power Supply	10 mv	0 mv	4.4	0	None	None
Detector Power Supply	0 mv	0 mv	0	0	None	None
Auxiliary Power Supply	0 mv	0 mv	0	0	None	None
Auxiliary Relay	-	-	-	-	None	None
Fan Failure Detector	-	-	-	-	None	None
Switch Module	-	-	-	-	None	None
Reactor Trip	-	-	-	-	None	None
Logic Buffer	-	-	-	-	None	None
Contact Buffer	-	-	-	-	None	None
Trip Logic	-	-	-	-	None	None
Safeguards Logic Test	-	-	-	-	None	None
Unit Control	-	-	-	-	None	None
Type "KQ" XMTR.	19 mv	15 mv	28	100	-	None
Pressure Switch	-	-	-	-	None	None
Type "BY" XMTR.	25 mv	0 mv	17	0	-	None

Table 7A-4d

Module Post Test Results - Resultant Error from X, Y, and Z Planes

Name of Unit	ΔV_o (D.C.)	ΔV_o (Noise)	Percent Allowable D.C. Error	Percent Allowable Noise	Number of Contact Failures	Number of Function Failures
Preamplifier	0 mv	0 mv	0	0	-	None
Count Rate Amplifier	3 mv	0 mv	2	0	None	None
Log Amplifier	3 mv	0 mv	2	0	-	None
Linear Amplifier	7 mv	15 mv	47	10	-	None
Buffer Amplifier	2 mv	0 mv	13	0	-	None
Rate of Change	2 mv	0 mv	13	0	-	None
Linear Bridge	1 mv	0 mv	67	0	None	None
Signal Converter	13 mv	0 mv	87	0	None	None
Contact Monitor	2 mv	0 mv	2.7	0	-	None
Square Root Extractor	1 mv	0 mv	6.7	0	-	None
Bistable	2 mv	-	8	-	None	None
Power Range Test	1 mv	8 mv	22	53	None	None
Intermediate Range Test	0 mv	0 mv	0	0	None	None
Source Range Test	0 mv	0 mv	0	0	None	None
Temperature Test	0 mv	0 mv	0	0	None	None
Pressure Test	0 mv	0 mv	0	0	None	None
Flow Test	1 mv	0 mv	13	0	None	None
Contact Monitor Test	1 mv	0 mv	13	0	None	None
System Power Supply	14 mv	0 mv	6.2	0	None	None
Detector Power Supply	100 mv	0 mv	7.4	0	None	None
Auxiliary Power Supply	0 mv	0 mv	0	0	None	None
Auxiliary Relay	-	-	-	-	None	None
Fan Failure Detector	-	-	-	-	None	None
Switch Module	-	-	-	-	None	None
Reactor Trip	-	-	-	-	None	None
Logic Buffer	-	-	-	-	None	None
Contact Buffer	-	-	-	-	None	None
Trip Logic	-	-	-	-	None	None
Safeguards Logic Test	-	-	-	-	None	None
Unit Control	-	-	-	-	None	None
Type "KQ" XMTR.	25 mv	15 mv	37	100	-	None
Pressure Switch	-	-	-	-	None	None
Type "BY" XMTR.	34 mv	0 mv	23	0	-	None

Table 7A-5a

Module Test Results -
Typical Channel Resultant Error

Channel	Percent Allowable D.C. Error	Percent Allowable Noise
Source Range	5.2	0
Intermediate Range	3.8	0
Power Range	13.3	3
Coolant Flow	14.4	0
Pump Power Monitor	5.9	0
Reactor Pressure	10	0
Outlet Temperature	27.4	0

NOTE: The above errors do not include input device errors.

Table 7A-5b

Module Post Test Results -
Typical Channel Resultant Error

Channel	Percent Allowable D.C. Error	Percent Allowable Noise
Source Range	3.1	0
Intermediate Range	3.5	0
Power Range	22	10
Coolant Flow	9	0
Pump Power Monitor	4.6	0
Reactor Pressure	8.9	0
Outlet Temperature	35	0

NOTE: The above errors do not include input device errors.

As shown in the above relations, the total error for any unit or module is the root mean square of the errors resulting from each single plane motion.

When computing typical channel errors as a result of composite force application, the relations given above still hold. Therefore, the total channel error is given by the algebraic sum of the individual unit errors.

7A.2.3.3.3 Extension of Data to Parameters not Monitored

The chosen parameters -- D.C. accuracy and noise levels -- are totaly representative of all parameters affecting module performance. D.C. accuracy is a measure of virtually all parameters inherent to module, unit, or channel performance. D.C. accuracy reflects stability, gain, linearity, repeatability, and in general the dynamic performance of the device in question. The remaining parameters of response and noise rejection are adequately indicated by relative magnitude and form of the output noise. It is, therefore, concluded that all performance parameters are reflected in the tabulated data presented in this report.

7A.2.3.3.4 Acceptance Criteria for Tabulated Data

Allowable D.C. error and noise are expressed as percentage of maximum tolerable limits. A module or unit is considered qualified under the Seismic Criteria if the percent allowable figure is 100% or less.

7A.2.3.4 MODULE TEST CONCLUSIONS

The tabulated data presented in Section 7A.2.3.3 shows that all Nuclear Instrumentation/Reactor Protective System modules and all Engineered Safeguards Protective System modules and Peripheral Equipment are qualified under the seismic criteria set forth in Section 7A.2.3.1. It may also be seen that any channel combinations of the tested modules also meet the stated criteria.

A separate test of representative electronic modules was conducted which showed that the modules have dampings between 6 and 15% of critical damping and resonant frequencies in excess of 36 Hertz. Since the excitation of the modules was sinusoidal, magnification of the input acceleration occurred within the modules. The magnitude of this magnification is one-half the reciprocal of the percent damping. Accordingly, the effective module test levels were greater than the input accelerations by a factor of at least 3.3, as listed in Table 7A-6.

Table 7A-6

Input Test Acceleration

Minutes into Sweep	Hz	Vertical Shaker: Input Accel. (g's)	Vertical Shaker: Effective Accel. (g's)	Horizontal Shaker: Input Accel. (g's)	Horizontal Shaker: Effective Accel. (g's)
0	1	0.07	0.23	0.07	0.23
	2	0.17	0.57	0.07	0.23
	3	0.37	1.23	0.20	0.67
1	4	0.40	1.33	0.33	1.10
	5	0.73	2.43	0.66	2.20
2	6	0.78	2.60	0.67	2.23
	7	1.00	3.33	0.97	3.23
	8	1.00	3.33	1.00	3.33
3	8.8	1.00	3.33	1.00	3.33
4	11.4	1.00	3.33	1.00	3.33
5	14.0	1.00	3.33	1.00	3.33
6	16.6	1.00	3.33	1.00	3.33
7	19.2	1.00	3.33	1.00	3.33
8	21.8	1.00	3.33	1.00	3.33
9	24.4	1.00	3.33	1.00	3.33
10	27.0	1.00	3.33	1.00	3.33
11	29.6	1.00	3.33	1.00	3.33
12	32.2	1.00	3.33	1.00	3.33
13	34.8	1.00	3.33	1.00	3.33
14	37.4	1.00	3.33	1.00	3.33
15	40.0	1.00	3.33	1.00	3.33

The above schedule was maintained within + - 2 Hz at any time interval, with acceleration levels maintained at + - 10 percent.

The accelerations applied to the modules during the test were in excess of the required test levels listed in Section 7A.2.2.4 for all frequencies above 3 Hz. Although the levels below 3 Hz were less than required because of test apparatus limitations, the test was sufficient to qualify the electronic modules since it has been shown that no resonances occur below 3 Hz and because the test levels at greater than 3 Hz are well in excess of the required test levels. Thus, the protective system electronic logic modules and equipment cabinets have met the test objectives of Section 7A.2.1.

7A.2.4 NUCLEAR INSTRUMENTATION NEUTRON DETECTORS

7A.2.4.1 NEUTRON DETECTOR TEST OBJECTIVE

The detector was subjected to sinusoidal excitation in the horizontal plane with respect to the detector's normally installed position. The detector was vibrated in the horizontal plane at a constant acceleration of 1 g over a frequency range of 5 to 20 Hertz. Below 5 Hertz the force level was variable whose magnitude was less than 1 g and was a function of the test facility constraints. The frequency sweep duration was 30 seconds minimum.

7A.2.4.2 NEUTRON DETECTOR TEST DESCRIPTION

The detector was also subjected to sinusoidal excitation in the vertical plane. The detector was vibrated at a constant acceleration of 1 g over a frequency range of 10 to 40 Hertz. The frequency sweep duration was 30 seconds minimum.

The power supply and electrometers were converted to the detector to monitor currents generated during the test. In addition, detector resistance, capacitance, and nuetron sensitivity were measured before and after the test. These parameters must remain within acceptance specification limits and must not change by more than ± 15% of the initial value as a result of the test.

7A.2.4.3 NEUTRON DETECTOR TEST RESULTS AND CONCLUSION

The Intermediate Range Neutron Detector Assembly and Power Range Neutron Detector Assembly have been tested and have satisfactorily met the test objectives. The Source Range Neutron Detector Assembly is presently undergoing testing.

7A.2.5 SEISMIC TESTING OF PRESSURE TRANSMITTERS

7A.2.5.1 TRANSMITTER TEST OBJECTIVE

The pressure transmitter was subjected to a sinusoidal force component individually applied to each of the three orthogonal axes of the transmitter. Since there were no resonances observed below 100 Hertz, the transmitter was vibrated for two hours at 10 g's at 100 Hertz along each of the three mutually perpendicular axes. At no time during the testing could the output of the transmitter deviate by more than 0.5% of range.

7A.2.5.2 TRANSMITTER TEST RESULTS

The pressure transmitter satisfactorily met the test objective. The maximum deviation of the transmitter output was 0.25% of range.

7A.2.6 CONCLUSION

Since all components of the Nuclear Instrumentation/Reactor Protective System and the Engineered Safeguards Protective System have met their respective seismic test objectives, these systems have been satisfactorily qualified in accordance with the general seismic test objectives of Section 7A.2.1.