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4   REACTOR COOLANT SYSTEM

4.1   DESIGN BASES

The reactor coolant system consists of the reactor vessel, coolant pumps, steam
generators, pressurizer, and interconnecting piping. The functional relation-
ship between major coolant system components is shown in Figure 4-1. The cool-
ant system physical arrangement is shown in Figures 4-2 and 4-3.

The reactor coolant system is designed to meet the following codes:

> Piping and Valves - ASAB31.1-1955 (Pressure Piping) including
> nuclear cases.

> Pump Casing - ASME Boiler and Pressure Vessel Code, Section III,
> Nuclear Vessels.

> Steam Generators - ASME Boiler and Pressure Vessel Code, Section III,
> Nuclear Vessels.

> Pressurizer - ASME Boiler and Pressure Vessel Code, Section III,
> Nuclear Vessels.

> Reactor Vessel - ASME Boiler and Pressure Vessel Code, Section III,
> Nuclear Vessels.

> Welding Qualifications - ASME Boiler and Pressure Vessel Code,
> Section IX.

To assist in the review of the system drawings, a standard set of symbols and
abbreviations have been used and are presented in summary in Figure 9-1.

4.1.1   PERFORMANCE OBJECTIVES

The reactor coolant system is designed to contain and circulate reactor cool-
ant at pressures and flows necessary to transfer the heat generated in the re-
actor core to the secondary fluid in the steam generators. In addition to
serving as a heat transport medium, the coolant also serves as a neutron mod-
erator and reflector, and as a solvent for the soluble poison (boric acid)
utilized in chemical shim reactivity control.

As the coolant energy and radioactive material container, the reactor coolant
system is designed to maintain its integrity under all operating conditions.
While performing this function, the system serves the safeguard objective of
preventing the release to the reactor building of any fission products which
escape the primary barrier, the core cladding.

4.1.2   DESIGN CHARACTERISTICS

4.1.2.1   Design Pressure

The reactor coolant system design, operating, and control set point pressures
are listed in Table 4-1. The design pressure allows for operating transient
pressure changes. The selected design margin considers core thermal lag,

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coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 4-2 through 4-6.

4.1.2.2   Design Temperature

The design temperature for each component is selected above the maximum anticipated coolant temperature in that component under all normal and transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 4-2 through 4-6.

4.1.2.3   Reaction Loads

All components in the reactor coolant system are supported and interconnected so that piping reaction forces result in combined mechanical and thermal stresses in equipment nozzles and structural walls within established code limits. Equipment and pipe supports are designed to absorb piping rupture reaction loads for elimination of secondary accident effects such as pipe motion and equipment foundation shifting.

4.1.2.4   Seismic Loads

Reactor coolant system components are designated as Class I equipment, and are designed to maintain their functional integrity during earthquake. The basic design guide for the seismic analysis is the AEC publication TID-7024, "Nuclear Reactors and Earthquake." Structures and equipment will be designed in accordance with Appendix 5A.

4.1.2.5   Cyclic Loads

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operation. Design cycles are shown in Table 4-7. During unit startup and shutdown, the rates of temperature and pressure changes are limited.

4.1.2.6   Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of reactor coolant system surfaces. The reactor coolant chemistry is discussed in further detail in 9.2.

4.1.3   EXPECTED OPERATING CONDITIONS

Throughout the load range from 15 to 100 per cent power, the reactor coolant system is operated at a constant average temperature. Reactor coolant system pressure is controlled to provide sufficient overpressure to maintain adequate core subcooling.

The minimum operating pressure is established from core thermal analysis. This analysis is based upon the maximum expected inlet and outlet temperatures, the

maximum reactor power, the minimum DNBR required (including instrumentation errors and the reactor control system deadband), and a core flow distribution factor. The maximum operating pressure is established on the basis of ASME Code relief valve characteristics and the margins required for normal pressure variations in the system. Pressure control between the preset maximum and minimum limits is obtained directly by pressurizer spray action to suppress high pressure and pressurizer heater action to compensate for low pressure. Normal operational lifetime transient cycles are discussed in detail in 4.1.4.

4.1.4 SERVICE LIFE

The service life of reactor coolant system pressure components depends upon the end-of-life material radiation damage, Unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures. In the following discussion each of these life-dependent factors will be discussed with regard to the affected components.

4.1.4.1 Material Radiation Damage

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. To assess the potential radiation damage at the end-of-reactor service life, the maximum exposure from fast neutrons (E > 1.0 mev) has been computed to be 3.0×10^{19} n/cm^2 over a 40 year life with an 80 per cent load factor. Reactor vessel irradiation exposure calculations are described in 3.2.2.1.7.

For this neutron exposure, the predicted Nil-Ductility Transition Temperature (NDTT) shift is 250 F based on the curve shown in Figure 4-4.[1] Based on an initial NDTT of 10 F, this shift would result in a predicted NDTT of 260 F.

The "Trend Curve for 550 F Data", as shown in Figure 4-4, represents irradiated material test results and was compiled from the reference documents listed in Table 4-11.

To evaluate the NDTT shift of welds, heat-affected zones and base material for the material used in the vessel, test coupons of these three material types have been included in the reactor vessel surveillance program (4.4.3).

4.1.4.2 Unit Operational Thermal Cycles

To establish the service life of the reactor coolant system components as required by the ASME III, for Class "A" vessels, the unit operating conditions which involve the cyclic application of loads and thermal conditions have been established for the 40 year design life.

The number of thermal and loading cycles to be used for design purposes are listed in Table 4-7 under the title "Design Cycles". The estimated actual cycles based on a review of existing nuclear stations operations are also provided in Table 4-7. Table 4-9 lists those components designed to ASME III - Class "A". The effect of individual transients, and the sum of these transients, are evaluated to determine the fatigue usage factor during the detail

design and stress analysis effort. As specified in ASME III Paragraph 415.2 (d)(6), the cumulative fatigue usage factor will be less than 1.0 for the design cycles listed in Table 4-7.

The transient cycles listed in Table 4-7 are conservative and complete in that they include all significant modes of normal and emergency operation. The estimated frequency basis for the design transient cycles are listed in Table 4-8.

A heatup and cooldown rate of 100 F/hour will be used in the analysis of Transients 1 and 2 in Table 4-7.

The miscellaneous transients (Item 8) listed in Table 4-7 include the initial hydrotests, plus an allowance for future hydrotests in the event that reactor coolant system modifications or repairs may be required. Subsequent to a normal refueling operation only the reactor vessel closure seals are hydrotested for pressure integrity; therefore, reactor coolant system hydrotests prior to startup are not included.

4.1.4.3   Operating Procedures

The reactor coolant system pressure vessel components are designed using the transition temperature method of minimizing the possibility of brittle fracture of the vessel materials. The various combinations of stresses are evaluated and employed to determine the system operating procedures

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 nil-ductility transition temperature (NDTT) and the design transition temperature (DTT), which is equal to NDTT + 60 F, the pressure vessels will be operated such that the stress levels will be restricted to a value which will prevent brittle failure. These levels are:

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.

If the nominal stresses are held within the referenced stress limits (a through c above), brittle fracture will not occur. This statement is based on data which has been reported by Robertson[3] and Kihara and Masubichi[4] in published literature. It can be shown that stress limits can be controlled by imposing operating procedures which control pressure and temperature during heatup and cooldown.[3] This procedure will insure that the nominal stress levels do not exceed those as specified in a through c above.

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4.1.4.4   Quality Manufacture

Material selection is discussed in detail in 4.2.5.

After receipt of the material a program of qualification of all welding and heat treating processes which could affect mechanical or metallurgical properties of the material during fabrication is undertaken. The purpose of this program is to establish the properties of the material as received, and to certify that the mechanical properties of the materials in the finished vessels are consistent with those used in the design analysis. This program consists of:

a.   Weld qualification test plates using production procedures and subjecting test plates to the heat treatments to be used in fabricating the vessels.

b.   Subjecting qualification test plates to all nondestructive tests to be employed in production, such as x-ray, dye-penetrant, magnetic particle, and ultrasonic. Acceptance standards are the same as used for production.

c.   Subjecting qualification test plates to destructive tests to establish the following:

(1)   Tensile strength.

(2)   Ductility.

(3)   Resistance to brittle fracture of the weld metal, base metal, and heat-affected zone (HAZ) metal.

After completion of the qualification test program, production welding and inspection procedures are prepared.

All plate or other materials are permanently identified, and the identity is maintained throughout manufacture so that each piece can be located in the finished vessels.

In-process and final dimensional inspections are made to insure that parts and assemblies meet the drawing requirements, and an "as-built" record is kept of these dimensions for future reference.

All welders are qualified or requalified as necessary in accordance with The Babcock & Wilcox Company and ASME IX requirements. Each lot of welding electrodes and fluxes is tested and qualified before release to insure that required mechanical properties and as-deposited chemical properties can be met. Electrodes are identified and issued only on an approved request to insure that the correct materials are used in each weld. All welding electrodes and fluxes are maintained dry and free from contamination prior to use. Records are maintained and reviewed by welding engineers to insure that approved procedures and materials are being used. Records are maintained for each weld joint and include the welder's name, essential weld parameters, and electrode heat or lot number.

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The several types of nondestructive tests performed during vessel fabrication are as follows:

a. Radiography, including x-ray, high voltage linear accelerator, or radioactive sources, will be used as applicable to determine the acceptability of pressure integrity welds and other welds as specifications require.

b. Ultrasonics is used to examine all pressure-integrity raw material, the bond between corrosion-resistant cladding to base material, and pressure-containing welds.

c. Magnetic Particle Examination is used to detect surface or near surface defects in machined weld grooves prior to welding, completed weld surfaces, and the complete external surface of the vessels including weld seams after final heat treatment.

d. Liquid Penetrant is used to detect surface defects in the weld deposit cladding, nonmagnetic materials, and closure studs.

The completed reactor vessel assembly will be shipped as a unit from the fabrication shop to the station site. The completed reactor closure head will be shipped in like manner.

4.1.5 CODES AND CLASSIFICATIONS

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested to applicable codes as listed in Table 4-9.

4.2 SYSTEM DESCRIPTION AND OPERATION

4.2.1 GENERAL DESCRIPTION

The reactor coolant system consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed coolant circulating pumps, an electrically heated pressurizer, and interconnecting piping. The system is arranged as two heat transport loops, each with two circulating pumps and one steam generator. Reactor system design data are listed in Tables 4-2 through 4-6, and a system schematic is shown in Figure 4-1. Elevation and plan views of the arrangement of the major components are shown in Figures 4-2 and 4-3.

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 ring flange.

The reactor vessel has six major nozzles for reactor coolant flow, 69 control rod drive assembly nozzles mounted on the reactor vessel head, and two emergency injection system nozzles—all located above the core. The vessel closure seal is formed by two concentric O-rings with provisions between them for leak

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detection. The reactor vessel, nozzle design, and seals incorporate the extensive design and fabrication experience accumulated by B&W. Fifty-one incore instrumentation nozzles are located on the lower head.

The reactor closure head 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. Pressure taps are provided in the annulus between the two O-rings to monitor leakage and to hydrotest the vessel closure seal after refueling.

The vessel is insulated with metallic reflective-type insulation supported on lugs welded to the outside of the vessel. Insulation panels are provided for the reactor closure head.

The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods in the withdrawn position. The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control cluster assemblies, surveillance specimens, and incore instrumentation.

The reactor vessel shell material is protected against fast neutron flux and gamma heating effects by a series of water annuli and the thermal shield located between the core and vessel wall. This protection is further described in 3.2.4.1.2, 4.1.4, and 4.3.1.

Stop blocks welded to the reactor vessel inside wall limit reactor internals and core vertical drop to 1/2 in. or less, and prevent rotation about the vertical axis in the unlikely event of a major internals component failure.

Surveillance specimens made from reactor steel are located between the reactor vessel wall and the thermal shield. These specimens will be examined at selected intervals to evaluate reactor vessel material NDTT changes as described in 4.4.3.

The reactor vessel general arrangement is shown in Figure 4-5, and the general arrangement of the reactor vessel and internals is shown in Figures 3-46 and 3-47.

Reactor vessel design data are listed in Table 4-2.

4.2.2.2   Pressurizer

The general arrangement of the reactor coolant system pressurizer is shown in Figure 4-6, and the design characteristics are tabulated in Table 4-3. The electrically heated pressurizer establishes and maintains the reactor coolant pressure within prescribed limits and provides a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation.

The pressurizer is a vertical cylindrical vessel connected to the reactor outlet piping by the surge line. The pressurizer vessel is protected from thermal shock by a thermal sleeve on the surge line and by a distribution baffle located above the surge line entrance to the vessel.

Relief valves are mounted on the top of the pressurizer and function to relieve any system overpressure. Each valve has one-half the required relieving capacity. The capacity of these valves is discussed in 4.3.4. The relief valves discharge to a quench tank located within the reactor building. The quench tank has a stored water supply to condense the steam. A relief valve protects the tank against overpressure should a pressurizer valve fail to reseat.

The pressurizer contains replaceable electric heaters in its lower section and a water spray nozzle in its upper section to maintain the steam and water at the saturation temperature corresponding to the desired reactor coolant system pressure. During outsurges, as the pressure in the reactor decreases, some of the water in the pressurizer flashes to steam to maintain pressure. Electric heaters are actuated to restore the normal operating pressure. During insurges, as pressure in the reactor system increases, steam is condensed by a water spray from the reactor inlet lines, thus reducing pressure. Spray flow and heaters are controlled by the pressurizer pressure controller.

Instrumentation for the pressurizer is discussed in 7.3.2.

4.2.2.3   Steam Generator

The general arrangement of the steam generators is shown in Figure 4-7, and design data are tabulated in Table 4-4.

The steam generator is a vertical, straight-tube-and-shell heat exchanger and produces superheated steam at constant pressure over the power range. Reactor coolant flows downward through the tubes, and steam is generated on the shell side. The high pressure parts of the unit are the hemispherical heads, the tubesheets, and the straight Inconel(*) tubes between the tubesheets. Tube supports hold the tubes in a uniform pattern along their length.

The shell, the outside of the tubes, and the tubesheets form the boundaries of the steam-producing section of the vessel. Within the shell, the tube bundle is surrounded by a shroud, which is in two overlapping sections with the upper section the larger of the two in diameter. The upper part of the annulus between the shell and baffle is the superheater outlet, while the lower part is the feedwater inlet-heating zone. Vents, drains, instrumentation nozzles, and inspection openings are provided on the shell side of the unit. The reactor coolant side has instrumentation connections on the top and bottom heads, manways on both heads, and a drain nozzle for 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. The unit is supported by a skirt attached to the bottom head.

Reactor coolant water enters the steam generator at the upper plenum, flows down the Inconel tubes while transferring heat to the secondary shell-side fluid, and

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*Inconel is a trade name of an alloy manufactured by the International Nickel Company. It also has substantial common usage as a generic description of a Ni-Fe-Cr alloy conforming to ASTM Specification SB-163. It is in the latter context that reference is made here.

exits through the lower plenum. Figure 4-8 shows the flow paths and steam generator heating regions.

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 a feed heating annulus (downcomer) formed by the shell and the baffle around the tube bundle. The steam which heats the feedwater to saturation is drawn into the downcomer by condensing action of the relatively cold feedwater.

The saturated water in the downcomer forms a static head to balance the static head in the nucleate boiling section. This provides the 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. With low (less than 1 ft/sec) saturated water velocities entering the generating section, the secondary side pressure drops in the boiling section are negligible. The majority of the pressure drop is due to the static head of the mixture. Consequently, the downcomer level of water balances the mean density of the two-phase boiling mixture in the nucleate boiling region.

b. Nucleate Boiling

The saturated water enters the tube bundle, and the steam-water mixture flows upward on the outside of the Inconel tubes countercurrent to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until DNB, i.e., departure of nucleate boiling, is reached, and then film boiling and superheating occurs. The quality at which transition from nucleate boiling to film boiling occurs is a function of pressure, heat flux, and mass velocity.

c. Film Boiling and Superheated Steam

Dry saturated steam is produced in the film boiling region at the upper end of the tube bundle. Baffles are used in this region to obtain higher vapor velocity and crossflow for efficient heat transfer.

d. Superheated Steam

Saturated steam is raised to final temperature in the baffled superheater region. Shown on Figure 4-9 is a plot of heating surface and downcomer level versus load. As shown, the downcomer water level is proportional to steam flow from 15 - 100 per cent load. A constant minimum level is held below 15 per cent load. The amount of surface (or length) of the nucleate boiling section and the film boiling section is proportional to load. The surface available for superheating varies inversely with load, i.e., as load decreases the superheat section gains from the nucleate and film boiling regions.

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Mass inventory in the steam generator increases with load as the length of the heat transfer regions varies.

The simple concept with ideal counterflow conditions results in highly stable flow characteristics on both the reactor coolant and secondary sides. The hot reactor coolant fluid is cooled uniformly as it flows downward. The secondary side mass flow is low, and the majority of the pressure drop is due to the static effect of the mixture. The boiling in the steam generator is somewhat similar to "pool boiling", except that there is motion upward that permits some parallel flow of water and steam.

A plot of reactor coolant and steam temperatures versus power is shown in Figure 7-5. As shown, both steam pressure and average reactor coolant temperature are held constant over the load range from 15 to 100 per cent full power. Constant steam pressure is obtained by a variable two-phase boiling length (see Figure 4-8), and by the regulation of feed flow to obtain proper steam generator secondary mass inventory. In addition to average reactor coolant temperature, reactor coolant flow is also held constant. The difference between reactor coolant inlet and outlet temperatures increases proportionately as load is increased. Saturation pressure and temperature are held constant, thereby resulting in a variable superheater outlet temperature.

Figure 4-10, a plot of temperature versus tube length, shows the temperature differences between shell and tube throughout the steam generator at full load. The excellent heat transfer coefficients permit the use of a secondary operating pressure and temperature sufficiently close to the reactor coolant average temperature so that a straight-tube design can be used.

Control of the shell temperature is achieved by the use of direct contact steam that heats the feedwater to saturation, and the shell is bathed with saturated water from feedwater inlet to the lower tubesheet.

In the superheater section, the tube wall temperature approaches the reactor coolant fluid temperature since the steam film heat transfer coefficient is considerably lower than the reactor coolant heat transfer coefficient. By baffle arrangement in the superheater section, the shell section is bathed with superheated steam above the steam outlet nozzle, further reducing temperature differentials between tubes and shell.

A discussion of the B&W once-through steam generator development program is presented in Appendix 4A.

The steam generator design and stress analysis will be performed in accordance with the requirements of the ASME III as described in 4.3.1.1.

4.2.2.4   Reactor Coolant Pumps

The general arrangement of a reactor coolant pump is shown in Figure 4-11, and the pump design data are tabulated in Table 4-5. The reactor coolant pumps are vertical, single-speed, shaft-sealed units having bottom suction and horizontal discharge. Each pump has a separate, single-speed, water-jacketed, top-mounted motor, which is connected to the pump by a shaft coupling.

Shaft sealing is accomplished in the upper part of the pump housing using a throttle bushing, a seal chamber, a mechanical seal, and a drain chamber in series. Seal water is injected ahead of the throttle bushing at a pressure approximately 50 psi above reactor system pressure. Part of the seal flow passes into the pump volute through the radial pump bearing. The remainder flows out along the throttle bushing, where its pressure is reduced, to the seal chamber and is returned to the seal water supply system. The outboard mechanical seal normally operates at a pressure of approximately 50 psig and a temperature of 95 to 100 F. However, it is designed for full reactor coolant system pressure and, if seal chamber cooling were maintained, would continue to operate satisfactorily without seal water injection for several weeks. The outboard drain chamber would further prevent leakage to the reactor building if deterioration of the mechanical seal performance should occur.

A water-lubricated, self-aligning, radial bearing is located in the pump housing. An oil-lubricated radial bearing and a Kingsbury type, double-acting, oil-lubricated thrust bearing are located in the pump motor. The thrust bearing is designed so that reverse rotation of the shaft will not lead to pump or motor damage. Lube oil cooling is accomplished by cooling coils in the motor oil reservoir. Oil pressure required for bearing lubrication is maintained by internal pumping provisions in the motor, or by an external system if required for "hydraulic-jacking" of the bearing surfaces for startup.

Factory thrust, vibration, and seal performance tests will be made in a closed loop on the first pump at rated speed with the pump end at rated temperature and pressure. Sufficient testing will be done on subsequent units to substantiate that they conform to the initial test pump characteristics.

4.2.2.5   Piping

The general arrangement of the reactor coolant system piping is shown in Figures 4-2 and 4-3. Piping design data are presented in Table 4-6. In addition to the pressurizer surge line connection, the piping is equipped with welded connections for pressure taps, temperature elements, vents, drains, decay heat removal, and emergency high pressure injection water. Thermal sleeves are provided in the pressurizer surge line and the emergency high pressure injection line connections.

4.2.3   PRESSURE-RELIEVING DEVICES

The reactor coolant system is protected against overpressure by control and protective circuits such as the high pressure trip and code relief valves located on the top head of the pressurizer. The relief valves discharge into the quench tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 4-1. Since all sources of heat in the system, i.e., core, pressurizer heaters, and reactor coolant pumps, are interconnected by the reactor coolant piping with no intervening isolation valves, all relief protection can conveniently be located on the pressurizer.

4.2.4   ENVIRONMENTAL PROTECTION

The reactor coolant system is surrounded by concrete shield walls. These walls provide shielding to permit access into the reactor building for inspection and maintenance of miscellaneous rotating equipment during full power operation and

for periodic calibration of incore detectors. These shielding walls act as missile protection for the reactor building liner plate.

Lateral bracing will be provided near the steam generator upper tube sheet elevation to resist lateral loads, including those resulting from seismic forces, pipe rupture, thermal expansion, etc. Additional bracing is provided at a lower elevation to restrain the 36 in. ID vertical pipe leg from whipping.

Concrete slabs over the reactor coolant system are also provided for shielding and missile damage protection.

4.2.5   MATERIALS OF CONSTRUCTION

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 4-10.

All reactor coolant system materials exposed to the coolant are corrosion-resistant materials consisting of 304 or 316 SS, weld deposit 304 SS cladding, Inconel (Ni-Cr-Fe), and 17-4 PH (H1100). These materials were chosen for specific purposes at various locations within the system because of their superior compatibility with the reactor coolant.

Periodic analyses of the coolant chemical composition will be performed to monitor the adherence of the system to the reactor coolant water quality listed in Table 9-4. Maintenance of the water quality to minimize corrosion is performed by the chemical addition and sampling system which is described in detail in 9.2.

The feedwater quality entering the steam generator will be held within the limits listed in Table 9-3 to prevent deposits and corrosion inside the steam generators. This required feedwater quality has been successfully used in comparable once-through, nonnuclear steam generators. The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a combination of elements in varying degrees is present. The necessary conditions are a susceptible alloy, an aggressive environment, a stress, and time.

It is characteristic of stress corrosion that combinations of alloy and environment that result in cracking are usually quite specific. Environments that have shown to cause stress-corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism and the presence of chlorides and free oxygen. With regard to the former, experience has shown that chemical "hideout" or deposition of chemicals on the surface of tubes can occur in a "steam blanketed" area within a steam generator. In the presence of this environment, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. The once-through steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions indicated that Inconel Alloy 600 has excellent resistance to general and pitting-type corrosion in severe operating water conditions. Extensive operating experience with Inconel units has confirmed this conclusion. Inconel steam generator tubing is being used in the

German Nuclear Ship, "Otto Hahn", and the tubing has been tested under service temperature and pressure conditions and in a high O_2 water environment as follows:

Water Temperature	572 - 650 F
Pressure	2,000 psig
pH	6.9
O_2	0.040 ppm

Tubing inspected after 5,000 hours showed no signs of stress-corrosion cracking or other detrimental effects even though the O_2 was 0.040 ppm and considerably above the maximum that will be used in the Oconee Nuclear Station steam generators (Tables 9-3 and 9-4). This reaffirms the conclusion that Inconel is a satisfactory material for this service.

All external insulation of reactor coolant system components will be compatible with the component materials. The reactor vessel is insulated with metallic reflective insulation on the cylindrical shell exterior. The closure flanges and the top and bottom heads in the area of corrosion-resistant penetrations will be insulated with low halide-content insulating material. All other external corrosion-resistant surfaces in the reactor coolant system will be insulated with low or halide-free insulating material as required.

The reactor vessel plate material opposite the core is purchased to a specified Charpy V-notch test result of 30 -lb or greater at a corresponding nil-ductility transition temperature (NDTT) of 10 F or less, and the material will be tested to verify conformity to specified requirements and to determine the actual NDTT value. In addition, this plate will be 100 per cent volumetrically inspected by ultrasonic test using both normal and shear wave.

The remaining material in the reactor vessel, and other reactor coolant system components, is purchased to the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility which will ensure a low NDTT, and thereby to give assurance that the finished vessel can be initially hydrostatically tested and operated at room temperature without restrictions. The stress limits established for the reactor vessel are dependent upon the temperature at which the stresses are applied. As a result of fast neutron absorption in the region of the core, the material ductility will change. The effect is an increase in the NDTT. The predicted end-of-life NDTT value of the reactor vessel opposite the core is 260 F or less. The predicted neutron exposure and NDTT shift are discussed in 4.1.4.

The unirradiated or initial NDTT of pressure vessel base plate material is presently measured by two methods. These methods are the drop weight test per ASTM E208 and the Charpy V-notch impact test (Type A) per ASTM E23. The NDTT is defined in ASTM E208 as "the temperature at which a specimen is broken in a series of tests in which duplicate no break performance occurs at a 10 F higher temperature". Using the Charpy V-notch test, the NDTT is defined as the temperature

at which the energy required to break the specimen is a certain "fixed" value. For SA 302B steel the ASME III Table N-332 specifies an energy value of 30 ft-lb. This value is based on a correlation with the drop weight test and will be referred to as the "30 ft-lb fix". A curve of the temperature versus energy absorbed in breaking the specimen is plotted. To obtain this curve, 15 tests are performed which include three tests at five different temperatures. The intersection of the energy versus temperature curve with the 30 ft-lb ordinate is designated as the NDTT.

The available data indicates differences as great as 40 degrees between curves plotted through the minimum and average values respectively. The determination of NDTT from the average curve is considered representative of the material and is consistent with procedures as specified in ASTM E23. In assessing the NDTT shift due to irradiation, the translation of the average curves is used.

The material for these tests will be treated by the methods as outlined by ASME III Paragraph N-313. The test coupons will be 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 to certify the material as delivered to B&W. The exact test coupon locations are reviewed and approved by B&W to insure compliance with the applicable ASME Code and specifications. In accordance with ASME III Paragraphs N-712 and N-713, B&W performs Charpy V-notch impact tests on heat-affected zone (HAZ), base metal, and weld metal on all pressure vessel test plates.

Differences of 20 to 40 F in NDTT have been observed between T/4 and the surface in heavy plates. The T/4 location for Charpy V-notch impact specimens is conservative since the NDTT of the surface material is lower than that of the internal material.

The reactor vessel design includes surveillance specimens which will permit an evaluation of the neutron exposure-induced shift on the material nil-ductility transition temperature properties.

The material irradiation surveillance program is described in 4.4.3.

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4.2.6 MAXIMUM HEATING AND COOLING RATES

The normal reactor coolant system operating cycles are given in Tables 4-7 and 4-8 and described in 4.1.4. The normal system heating and cooling rate is 100 F/hr.

It is anticipated that the reactor vessel permissible heatup rate can exceed 100 F/hr. Consequently, there are sufficient electrical heaters installed in the pressurizer to permit a heatup rate, starting with minimum water level, of 150 F/hr. The exact final rates are determined during the detail design and stress analysis of the vessel.

The fastest cooldown rates which result from the break of a main steam line are discussed in 14.1.2.9.

4.2.7 LEAK DETECTION

To minimize leakage from the reactor coolant system all components are interconnected by an all-welded piping system. Some of the components have access openings of a flanged-gasketed design. The largest of these is the reactor vessel closure, which has a double metal O-ring seal with provisions for monitoring for leakage between the O-rings. Other openings and appurtenances to the reactor coolant system which are possible sources of leakage are tabulated in detail along with the maximum expected rates of leakage in Section 11.

With regard to the reactor vessel, the probability of a leak occurring is considered to be remote on the basis of reactor vessel design, fabrication, test, inspection, and operation at temperatures above the material NDTT as described in 4.3.1. Reactor vessel closure leakage will be zero 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 innermost O-ring seal. A stress analysis on a similar vessel design indicates this leak rate would be approximately 10 cc/min through the seal monitoring taps to a drain, and no leakage will occur past the outer O-ring seal. The exact nature of this transient condition, and the resulting small leak rate, will be determined by a detailed stress analysis.

In the unlikely event that an extensive leak should occur from the system into the reactor building during reactor operation, the leakage will be detected by one or more of the following methods:

    a. Instrumentation in the control room will indicate the addition rate of makeup water required to maintain normal water level in the pressurizer. Deviation from normal makeup and letdown to the reactor coolant system will provide an indication of the magnitude of the leak.

    b. Control room instrumentation will indicate additional reactor building atmosphere particulate or radioactive gas activity.

    c. Control room instrumentation will indicate the existence of a change in the water level in the reactor building sump.

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If any one of the above methods indicates an excessive reactor coolant leakage rate during operation, the reactor will be taken to a cold shutdown, and the cause of the problem will be determined.

4.3   SYSTEM DESIGN EVALUATION

4.3.1   SAFETY FACTORS

The reactor coolant system is designed, fabricated, and erected in accordance with proven and recognized design codes and quality standards applicable for the specific component function or classification. These components are designed for a pressure of 2,500 psig at a nominal temperature of 650 F. The corresponding nominal operating pressure of 2,185 psig allows an adequate margin for normal load changes and operating transients. The reactor system components are designed to meet the codes listed in Table 4-9.

Aside from the safety factors introduced by code requirements and quality control programs, as described in the following paragraphs, the reactor coolant system functional safety factors are discussed in Sections ⅃ and 14.

4.3.1.1   Pressure Vessel Safety

The safety of the nuclear reactor vessel and all other reactor coolant system pressure vessels is dependent upon four major factors: (1) design and stress analysis, (2) material selection and fabrication, (3) quality control, and (4) proper operation. The special care and detail used in implementing these factors in pressure vessel manufacture are briefly described as follows:

4.3.1.1.1   Design and Stress Analysis

These pressure vessels are designed to the requirements of the ASME III. This code is a result of ten years of effort by representatives from industry and government who are skilled in the design and fabrication of pressure vessels. It is a comprehensive code based on the most applicable stress theory. It requires a stress analysis of the entire vessel under both steady-state and transient operations. The result is a complete evaluation of both primary and secondary stresses, and the fatigue life of the entire vessel. This is a contrast with previous codes which basically established a vessel thickness during steady-state operations only.

In establishing the fatigue life of these pressure vessels, using the design cycles from Table 4-7, the fatigue evaluation curves of ASME III are employed.

Since ASME III requires a complete stress analysis, the designer must have at his disposal the necessary analytical tools to accomplish this. These tools are the solutions to the basic mathematical theory of elasticity equations. In recent years the capability and use of computers have played a major part in refining these analytical solutions. The Babcock & Wilcox Company has confirmed the theory of plates and shells by measuring strains and rotations on the large flanges of actual pressure vessels and finding them to be in agreement with

those predicted by the theory. B&W has also conducted laboratory deflection studies of thick shell and ring combinations to define the accuracy of the theory, and is using computer programs developed on the basis of this test data.

The analytical procedure considers all process operation conditions. A detail design and analysis of every part of the vessel is prepared as follows:

a. The vessel size and configuration are set to meet the process requirements, the thickness requirements due to pressure and other structural dead and live loads, and the special fillet contour and transition taper requirements at nozzles, etc., required by ASME III.

b. The vessel pressure and temperature design transients given in Table 4-7 are employed in the determination of the pressure loading and temperature gradient, and their variations with time throughout the vessel. The resulting combinations of pressure loading and thermal stresses are calculated. Computer programs are used in this development.

c. An evaluation of the stresses through the vessel is performed using as a criteria the allowable stresses per ASME III. This code gives safe stress level limits for all the types of applied stress. These are membrane stress (to insure adequate tensile strength of the vessel), membrane plus primary bending stress (to insure a distortion-free vessel), secondary stress (to insure a vessel that will not progressively deform under cyclic loading), and peak stresses (to insure a vessel of maximum fatigue life).

Much of the stress analysis mentioned in the above listing is statically indeterminate. Hence, when an evaluation of these stresses, as mentioned in c above, shows them to be in excess of permissible ASME III values, corrective design changes are made and the procedure reiterated.

A design report is prepared and submitted to the jurisdictional authorities and regulatory agencies, i.e., state, insurance, etc. This report defines in sufficient detail the design basis, loading conditions, etc., and will summarize the conclusions to permit independent checking by interested parties.

4.3.1.1.2   Quality Control

In-process and final dimensional inspections are made to insure that parts and assemblies meet the drawing requirements, and an "as-built" record is kept of these dimensions for reference. A temperature-controlled gage-room is maintained to keep all measuring equipment in proper calibration, and personnel supervising this work are trained in formal programs sponsored by gage equipment manufacturers.

The practice of applied radiography is being continually improved to enhance flaw detection. Present procedures are:

a. All welds are properly prepared by chipping and grinding valleys between stringer beads so that radiographs can be properly interpreted.

b. All radiographs are reviewed by two people knowledgeable and skilled in their interpretation.

c. An 0.080 in. lead filter is used at the film to absorb "broadbeam scatter", when using high voltage equipment (above 1 mev).

d. Fine grain or extra fine grain film is used for all exposures.

e. Densities of radiographs are controlled by the use of densitometers.

f. Double film technique is used on all gamma-ray exposures as well as high voltage exposures.

g. Films are processed through an automatic processor which has a controlled replenishment, temperature, and process cycle, all contributing to better quality.

h. Energies used are controlled to be in the optimum range.

Ultrasonics is one of the most useful inspection tools. It is being used as follows:

a. In addition to radiography, pressure-containing welds are inspected by ultrasonics.

b. In order to detect laminations which are normally parallel to the surface, plates are also inspected by a shear wave.

c. The bond between cladding and base material is inspected by ultrasonics.

d. All plate is 100 per cent volumetrically inspected by ultrasonics using both normal and shear wave.

e. Personnel conducting ultrasonic inspections are given extensive training.

The magnetic particle examination is used to aid in detecting surface and near surface defects, and is employed on both parts and finished vessels as follows:

a. Welds are inspected with the magnetic particle method after removal of backup strips.

b. Weld preparations are inspected by the magnetic particle method.

c. A complete external surface inspection of the entire vessel including weld seams is performed after all heat treatment.

d. Personnel using this method are trained by B&W in addition to attending formal programs conducted by the equipment manufacturers.

The liquid penetrant examination is used to aid in detecting surface defects and is particularly adaptable to the nonmagnetic materials such as stainless steel. It is presently being used as follows:

a. Inspection of weld-deposited cladding.

b. Inspection of reactor vessel studs.

c. Personnel using this method of examination are trained by B&W in addition to attending formal programs conducted by the equipment manufacturers.

The primary purpose of the above quality control procedures and methods is to locate, define, and determine the size of material defects to allow an evaluation of defect acceptance, rejection, or repair.

The size of defect that can conceivably contribute to the rupture of a vessel depends not only on the size effect, but also on the orientation of the defect, the magnitude of the stress field, and temperature. The correlation of these major parameters has been done by Pellini and Puzak[2] who have prepared a "fracture analysis diagram" which is the basis of vessel operation from cold startup and shutdown to full pressure and temperature operation.

The diagram predicts that, for a given level of stress, larger flaw sizes will be required for fracture initiation above the NDTT temperature. For example, at stresses in the order of 3/4 yield strength, a flaw in the order of 8 to 10 in. may be sufficient to initiate fracture at temperatures below the NDTT temperature. However, at NDTT + 30 F, a flaw of 1-1/2 times this size may be required for initiation of fracture. While at a temperature of NDTT + 60 F, brittle fracture is not possible under elastic stresses because brittle fracture propagation does not take place at this temperature. Fractures above this temperature are of the predominantly ductile type, and are dependent upon the member net section area and section modulus as they establish the applied stress.

Stud forgings will be inspected for flaws by two ultrasonic inspections. An axial longitudinal beam inspection will be performed. The rejection standard will be loss-of-back-reflection greater than that from a 1/2 in. diameter flat bottom hole. A radial inspection will be made using the longitudinal beam technique. This inspection will carry the same rejection standards as the axial inspection. In addition to ultrasonic test, liquid penetrant inspection will be performed on the finished studs.

The stress analysis of the studs will include a fatigue evaluation. It is not expected that fatigue evaluation will yield a significantly high usage factor for the 40 year design life. Therefore, there will be no planned frequency for stud replacement. If an indication is found when the studs are inspected

during refueling, as described below, the stud will be replaced.

One-third of the studs will be visually examined and dye-penetrant inspected at every refueling. Any positive indications found will be cause for rejection.

The reactor vessel closure contains sixty 6-1/2 in. diameter studs.

The stud material is A-540, Grade B23 (ASME III, Case 1335) which has 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. This means:

At Operating Conditions (2185 psig):

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.

4.3.1.1.3   Operation

As previously mentioned in 4.1.4, pressure vessel service life is dependent on adherence to established operating procedures. Pressure vessel safety is also dependent on proper vessel operation. Therefore, particular attention is given to fatigue evaluation of the pressure vessels and to the factors that affect fatigue life. The fatigue criteria of ASME III are the bases of designing for fatigue. They are based on fatigue tests of pressure vessels sponsored by the AEC and the Pressure Vessel Research Committee. The stress limits established for the pressure vessels are dependent upon the temperature at which the stresses are applied.

As a result of fast neutron absorption in the region of the core, the reactor vessel material ductility will change. The effect is an increase in the nil-ductility transition temperature (NDTT). The determination of the predicted NDTT shift is described in 4.1.4.1. This 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 above the design transition temperature (DTT). Below the 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. These stress levels define an operating coolant pressure temperature path or envelope for a stated heatup or cooldown rate that must be followed. Additional information on the determination of the operating procedures is provided in 4.1.4.1, 4.1.4.2, and 4.1.4.3.

4.3.1.1.4   Additional Pressure Vessel Safety Factors

Additional methods and procedures used in pressure vessel design, not previously mentioned in 4.3.1.1 above but which are considered conservative and provide an

additional margin of safety, follow:

a. Use of a stress concentration factor of 4 on assumed flaws in calculating stresses.

b. Use of minimum specified yield strength of the material instead of the actual values.

c. Neglecting the increase in yield strength resulting from irradiation effects.

d. The design shift in NDTT as given in 4.1.4.1 is based on maximum predicted flux levels at the reactor vessel inside wall surface, whereas the bulk of the reactor vessel material will experience a lesser exposure of radiation and consequently a lower change in NDTT over the life of the vessel.

e. Because the irradiation dosage is higher at the inside surface of the reactor pressure vessel wall, the surveillance specimens will be subjected to a greater degree of irradiation, and therefore a larger shift in the NDTT value than will be experienced by the vessel. The specimens lead the vessel with respect to irradiation effects and impart a degree of conservatism in the evaluation of the capsule specimens. The material irradiation surveillance program is described in 4.4.3.

f. Results from the method of neutron flux calculations, as described in 3.2.2.1.7, have increased the flux calculations by a factor of 2 in predicting the nvt in the reactor vessel wall. The conservative assumptions, uncertainties, and comparisons of calculational codes used in determining this factor are discussed in detail in 3.2.2.1.7.

The foregoing discussion presents a detailed description of quality design, fabrication, inspection, and operating procedures used to insure confidence in the integrity of pressure vessels. Experience reported by Reference (5), B&W, and the satisfactory experience of B&W customers support the conclusion that pressure vessel rupture is incredible.

4.3.1.2   Piping

Total stresses resulting from thermal expansion and pressure, and mechanical and seismic loadings, are all considered in the design of the reactor coolant piping. The total stresses which can be expected in the piping are within the maximum code allowables. The pressurizer surge line connection, and the high pressure injection connections, are equipped with thermal sleeves to limit stresses from thermal shock to acceptable values. All materials and fabrication procedures will meet the requirements of the specified code. All material will be ultrasonically inspected. All welds will be radiographically inspected. All interior surfaces of the interconnecting piping are clad with stainless steel to eliminate corrosion problems and to reduce coolant contamination.

4.3.1.3   Steam Generator

Because the basic concept of the once-through steam generator would indicate the possible existence of differential thermal expansion-induced stresses in either the tubes or shell, an evaluation of the thermal loadings has been performed using the most severe design transients from Table 4-7.

The basic structural premise of the steam generator is that the tubesheets themselves are designed to take the full design pressure on either side of the tubesheet with zero pressure on the other side. That is, the tubes are not counted upon for any structural aid or support.

The steam line failure analyzed in 14.1.2.9 closely simulates the above design premise in a transient man r. Secondary temperature variations during the accident are essentially transient skin effects with the controlling temperature for the tubesheets and tubes being that of the reactor coolant. Thermal stresses for this case will be below ASME allowable values. Some tube deformation may occur but this will be restrained by the tube supports.

During normal power operation the tubes are hotter than the shell of the steam generator in the amount of 10 F to 20 F depending upon load. The effect is to put the tubes in a slight compression of 3,000 psi at the 20 F maximum temperature difference. This causes no adverse effect on the tubes since this stress is well below the allowable stress of 23,300 psi for SB-163 material (ASME III, Case 1336). Buckling of the tubes does not occur since these are supported laterally at 40 in. intervals along their length.

During startup and shutdown operations the tubes are hotter than the shell of the steam generator in the amount of 40 F. This places the tubes in a compressive stress of 6,000 psi which causes no adverse effect on the tubes since this stress is well below the allowable stress of 23,300 psi for this SB-163 material (ASME III, Case 1336). Buckling of the tubes does not occur since these are supported laterally at 40 in. intervals along their length. To demonstrate the structural adequacy of the steam generator at this condition, a laboratory unit was constructed at the same tube size, length, and material as the steam generator, but of seven tubes in number. It was structurally tested with a thermal difference of shell and tube of 80 F for 2,000 cycles. This severe thermal cycle test was performed with a tube-to-shell temperature difference twice as great as the maximum expected during startup and shutdown (Transients 1 and 2, Table 4-7). Destructive examination of the unit after this test indicated no adverse effects from fatigue, stress, buckling, or tube-to-tubesheet joint leakage.

4.3.2   RELIANCE ON INTERCONNECTED SYSTEMS

The principal heat removal systems which are interconnected with the reactor
coolant system are the steam and feedwater systems and the low pressure in-
jection and decay heat removal system. The reactor coolant system is depen-
dent upon the steam generators, and the steam, feedwater, and condensate sys-
tems for decay heat removal from normal operating conditions to a reactor
coolant temperature of approximately 250 F. All vital active components in
these systems are duplicated for reliability purposes.

The engineering flow diagram of the steam and feedwater systems is shown in
Figure 10-1. In the event that the condensers are not available to receive
the steam generated by decay heat, the water stored in the feedwater system
may be pumped into the steam generators and the resultant steam vented to the
atmosphere. A turbine-driven, 5 per cent capacity, boiler feed pump will
supply water to the steam generators.

The low pressure injection and decay heat removal system is used to remove
decay heat when the thermal driving head of the reactor coolant system is no
longer adequate to generate steam. A complete description of this system is
presented in 9.5. The heat received by this system is ultimately rejected
to the service water system which also contains sufficient redundancy to
guarantee proper operation. A schematic diagram of the service water system
is presented in Figure 9-9.

4.3.3   SYSTEM INTEGRITY

Integrity of the reactor coolant system is insured by proper materials selec-
tion, fabrication quality control, design, and operation. All components in
the reactor coolant system are fabricated from materials initially having a
low nil-ductility transition temperature (NDTT) to eliminate the possibility
of propagation-type failures. Where material properties are subject to change
throughout Unit lifetime, such as the case with the reactor vessel, provisions
are included for materials surveillance specimens. These will be periodically
examined, and any required temperature-pressure restrictions will be incorporat-
ed into reactor operation to insure operation above NDTT.

The coolant system is designed in accord with ASME pressure vessel and ASA
power piping codes as covered in 4.1. Relief valves on the pressurizer are
sized to prevent system pressure from exceeding the design point by more than
10 per cent.

As a further assurance of system integrity, all components in the system will
be hydrotested at 3,125 psig prior to initial operation. The largest and most
frequently used opening in reactor coolant system, the reactor vessel head,
contains provisions for separate hydrostatic pressurization between the O-ring
type gaskets.

4.3.4   PRESSURE RELIEF

The reactor coolant system is protected against overpressure by safety valves
located on the top of the pressurizer.

The capacity of the pressurizer safety valves is determined from consider-
ations of: (1) the reactor protective system, (2) line pressure drop (static
and dynamic) between the point of highest pressure in the reactor coolant
system and the pressurizer, (3) the pressure drop in the safety valve dis-
charge line, and (4) accident or transient conditions which may potentially
cause overpressure.

Preliminary analysis indicates that the hypothetical case of withdrawal of a
regulating control cluster assembly bank from a relatively low power provides
the basis for establishing pressurizer safety valve capacity. The accident is
terminated by high pressure reactor trip with resulting turbine trip. This
accident condition produces a power mismatch between the reactor coolant sys-
tem and steam system larger than that caused by a turbine trip without immedi-
ate reactor trip, or by a partial load rejection from full load.

The safety valve capacity required to prevent overpressure for this hypotheti-
cal condition is approximately 600,000 lb/hr of steam. The safety valves are
sized in accordance with the requirements of ASME III.

4.3.5   REDUNDANCY

The reactor coolant system contains two steam generators and four 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 busses as shown in
Figure 8-1.

Two core flooding nozzles are located on the opposite sides of the reactor
vessel. Each nozzle is connected to a core flooding tank and a line from
the low pressure injection system. The high pressure injection lines are
connected to the reactor coolant system on two of the four coolant inlet
pipes, again to further insure water injection.

4.3.6   SAFETY ANALYSIS

The components of the reactor coolant system are interconnected by an all-
welded piping system. Since the reactor inlet and outlet nozzles are all
located above the core, there is never any danger of the reactor coolant un-
covering the core when any other system component is drained for inspection
or repair.

4.3.7   OPERATIONAL LIMITS

Reactor coolant system heat up and cool down rates are described in detail in
4.1.4 and 4.2.6.

The component stress limitations dictated by material NDTT considerations are
described in 4.1.4 and 4.3.1.

The reactor coolant system is designed for 2,500 psig at 650 F. The normal
operating conditions will be 2,185 psig at an average system temperature of
579 F at full power. In this mode of operation, the reactor vessel outlet
temperature is 603 F. Additional temperature and pressure variations at
various power levels are shown on Figure 7-5.

Reactor trip signals will be fed to the safety and protective system as a result of high coolant temperature, high pressure, low pressure, and low flow, i.e., flux-flow comparator. By relating low flow to the reactor power, operation at partial power is feasible with less than four reactor coolant pumps operating. The reactor operating limits are as shown in the following tabulation:

<div align="center">

Performance Vs Pumps In-Service

</div>

Reactor Coolant Pumps Operating	4 Pumps	3 Pumps	2 Pumps (Same Loop)	2 Pumps (2 Loops)
Maximum Reactor Power, %	100	86	64	60
Reactor Coolant Flow, %	100	74	46	38

Reactor operating limits under natural circulation conditions are discussed in 14.1.2.6.3.

Further discussions of the bases for the selection of operational limits are presented in 7.1.2.4.

The reactor coolant system is designed for continued operation with 1 per cent of the fuel rods in the failed condition. The tolerable radioactivity content of the coolant is based on long term saturation activities with 1 per cent failed fuel.

4.4   TESTS AND INSPECTIONS

4.4.1   COMPONENT IN-SERVICE INSPECTION

All reactor vessel internals are removeable to facilitate vessel inspection should it become necessary.

The pressure shell of the steam generator is completely inspectable on the outside by removing the thermal insulation. Direct visual examination, and magnetic particle or ultrasonic techniques, can be used if necessary. The reactor coolant side of the heads and tubesheets can be inspected by direct visual techniques by removing the 16 in. manway covers in the heads. The inside of the cylindrical portion of the main pressure shell can be inspected through manholes and handholes. The feedwater spray nozzles are removable for inspecting the shell nearby.

The reactor building arrangement provides sufficient space for inspection of the external surfaces of the reactor coolant piping.

4.4.2   REACTOR SYSTEM TESTS AND INSPECTIONS

The assembled reactor coolant system will be subjected to the following tests and inspections during final unit construction and initial startup phases. These tests are in addition to the tests in compliance with code requirements.

4.4.2.1 Reactor Coolant System Precritical and Hot Leak Test

The objective of this test is to demonstrate satisfactory preliminary operation of the entire system and its individual components, to check and evaluate operating procedures, and to determine reactor coolant system integrity at normal operating temperature and pressure.

4.4.2.2 Pressurizing System Precritical Operational Test

This test demonstrates satisfactory preliminary operation of the pressurizer and its individual components. Spray valve adjustments and heater control adjustments are tested.

4.4.2.3 Pressurizer Surge Line Temperature Gradient Test

The temperature at the midpoint of the pressurizer surge line is determined after a period of steady state operation to check temperature gradients.

4.4.2.4 Relief System Test

In this test all relief valves are set and adjusted, and operating procedures are evaluated.

4.4.2.5 Unit Power Startup Test

This test determines performance characteristics of the entire unit in short periods of operation at steady state power levels.

4.4.2.6 Unit Power Heat Balance

The purpose of this test is to determine the actual reactor heat balance at various power levels to provide the necessary data for calorimetric calibration of the nuclear instrumentation and reactor coolant system flow rate.

4.4.2.7 Unit Power Shutdown Test

The purpose of this test is to check and evaluate the operating procedures used in shutting down the unit and to determine the overall unit operating characteristics during shutdown operations.

4.4.3 MATERIAL IRRADIATION SURVEILLANCE

Surveillance specimens of the reactor vessel shell section material are installed between the core and inside wall of the vessel shell to monitor the NDTT of the vessel material during operating lifetime.

The type of specimens included in the surveillance program will be Charpy V-notch (Type A) and tensile specimens for measuring the changes in material properties resulting from irradiation. This is in accordance with ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors".

The reactor vessel material surveillance program will utilize a total of eight specimen capsules. Four capsules will be located close to the inside reactor

vessel wall and directly opposite the center portion of the core for each reactor vessel. The locations of the capsule holder tubes are shown in Figures 3-46 and 3-47. In these positions, the irradiation received by the specimens will be approximately three times that received by the reactor vessel.

The material from each reactor vessel will have its initial NDTT determined by the Charpy V-notch impact correlation with drop weight tests. The predicted shift or change in the NDTT of the vessel material resulting from irradiation is discussed in detail in 4.1.4.1 and is shown in Figure 4-12 relative to years of operation.

The influence of neutron irradiation on the reactor vessel material properties will be evaluated periodically during unit shutdowns for refueling as tabulated below. This will be accomplished by testing samples of the material from each reactor vessel which are contained in the surveillance specimen capsules. These capsules contain steel coupons from plate, weld, and heat-affected zone material used in fabricating the reactor vessels. Dosimeters are placed with the Charpy V-notch impact specimens and tensile specimens. The dosimeters will permit evaluation of flux as seen by the specimens and vessel wall. To prevent corrosion the specimens are enclosed in stainless steel sheaths.

The irradiated samples are tested to determine the material properties such as tensile, impact, etc., and the irradiated NDTT which may be measured in a manner similar to the initial NDTT. These test results can be compared with the then-existing data on the effects of neutron flux and spectrum on engineering materials.

Schedule for Capsule Removal

Unit No.	Capsule Number	Exposure Time (Years from start of unit)
1	1	1
	2	5
	3	10
	4	15
2	5	5
	6	15
	7	Spares
	8	Spares

The measured neutron flux and NDTT may then be compared and evaluated with the initial NDTT and predicted NDTT shift to monitor the progress of radiation-induced changes in the vessel materials. As the end of reactor design life nears, a significant increase in measured NDTT in excess of the predicted NDTT shift could be investigated by review of the vessel stress analysis and operating records. If necessary or required in accordance with the advanced knowledge available at that time, the vessel transient limitations on pressure and temperature may be altered so that vessel stress limits, as stated in 4.1.4.3 for heatup and cooldown, are not exceeded.

4.5   REFERENCES

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

(2) Pellini, W. S. and Puzak, P. P., Fracture Analysis Diagram Procedures for the Fracture-Safe Engineering Design of Steel Structures, Welding Research Council Bulletin 88, May 1963.

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

(4) Kihara, H. and Masubichi, K., Effects of Residual Stress on Brittle Fracture, Welding Journal, Volume 38, April 1959.

(5) Miller, E. C., The Integrity of Reactor Pressure Vessels, ORNL-NSIC-15, May 1966.

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Table 4-1

Tabulation of Reactor Coolant System Pressure Settings

Item	Pressure, psig
Design Pressure	2,500
Operating Pressure	2,185
Code Relief Valves	2,500
High Pressure Trip	2,350
High Pressure Alarm	2,300
Low Pressure Alarm	2,150
Low Pressure Trip	2,000

Table 4-2

Reactor Vessel Design Data

Item	Data
Design/Operating Pressure, psig	2,500/2,185
Hydrotest Pressure (cold), psig	3,125
Design/Operating Temperature, F	650/600
Overall Height of Vessel and Closure Head, ft-in.	37-4
Straight Shell Thickness, in.	8-7/16
Water Volume, ft^3	4,150
Thickness of Insulation, in.	3
Number of Reactor Closure Head Studs	60
ID of Flange, in.	165
ID at Shell, in.	171
Inlet Nozzle ID, in.	28
Outlet Nozzle ID, in.	36
Core Flooding Water Nozzle, in.	~~8~~ 14

TABLE 4-1, 4-2

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Table 4-3

Pressurizer Design Data

Item	Data
Design/Operating Pressure, psig	2,500/2,185
Hydrotest Pressure (cold), psig	3,125
Design/Operating Temperature, F	670/650
Normal Water Volume, ft^3	800
Normal Steam Volume, ft^3	700
Surge Line Nozzle Diameter, in.	10
Overall Height, ft-in.	44-0

Table 4-4

Steam Generator Design Data

Item	Data per Unit
Design Pressure, Reactor Coolant/Steam, psig	2,500/1,050
Hydrotest Pressure (tube side-cold), reactor coolant psig	3,125
Design Temperature, Reactor Coolant/Steam, F	650/600
Reactor Coolant Flow, lb/hr	65.66 x 10^6
Heat Transferred, Btu/hr	4.21 x 10^9
Steam Conditions at Full Load, Outlet Nozzles:	
Steam Flow, lb/hr	5.30 x 10^6
Steam Temperature, F	570 (35 F superheat)
Steam Pressure, psig	910
Feedwater Temperature, F	455
Overall Height, ft-in.	73 - 2-1/2
Shell OD, in.	147-1/16
Reactor Coolant Water Volume, ft^3	2,030

TABLE 4-3, 4-4

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4-30

Table 4-5

Reactor Coolant Pump Design Data

Item	Data per Unit
Number of Pumps	4
Design Pressure, psig	2,500
Hydrotest Pressure (cold), psig	3,125
Design Temperature, F	650
Operating Speed, rpm	1,180
Pumped Fluid Temperature, F	60 to 580
Developed Head, ft	370
Capacity, gpm	88,000
Hydraulic Efficiency, %	~~86~~ *88*
Seal Water Injection, gpm	~~50 to 65~~ *45 to 50*
Seal Water Return, gpm	~~55 to 62~~ *43 to 48*
Pump Nozzle ID, in.	28
Overall Unit Height, ft	24
Water Volume, ft^3	~~76~~ *95*
Motor Stator Frame Diameter, ft	8
Pump-Motor Moment of Inertia, lb-ft^2	70,000

Motor Data:

Type	Squirrel-Cage Induction, Single Speed
Voltage	~~4,000~~ *6,600*
Phase	3
Frequency, cps	60
Starting	Across-The-Line
Input (hot reactor coolant), kw	~~5,600~~ *5,400*
Input (cold reactor coolant), kw	~~7,350~~ *7,420*

TABLE 4-5

Table 4-6

Reactor Coolant Piping Design Data

Item	Data
Reactor Inlet Piping ID, in.	28
Reactor Outlet Piping ID, in.	36
Pressurizer Surge Piping, in.	10 Sch. ~~120~~ *160*
Design/Operating Pressure, psig	2,500/2,185
Hydrotest Pressure (cold), psig	3,125
Design/Operating Temperature, F	650/603
Design/Operating Temperature (pressurizer surge line), F	670/650
Water Volume, ft^3	1,900

Table 4-7

Transient Cycles

Transient Description	Design Cycles	Estimated Actual Cycles
1. Heatup, 70 to 579 F and Cooldown, 579 to 70 F	480	80
2. Heatup, 540 to 579 F and Cooldown, 579 to 540 F	1,440	770
3. Ramp Loading and Ramp Unloading	12,000	9,000
4. Step Loading Increase	2,000	1,500
5. Step Unloading Decrease	2,000	1,500
6. Step Load Reduction to Auxiliary Load	160	120
7. Reactor Trip From Full Power	400	300
8. Miscellaneous Transients	10	5

Above cycles are based on 40 year design life.

TABLE 4-6, 4-7

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4-32

Table 4-8

Design Transient Cycles

Transient No. (See Table 4-7)	Frequency
1. Heatup, 70 to 579 F and Cooldown, 579 to 70 F	12 per Year
2. Heatup, 540 to 579 F and Cooldown, 579 to 540 F	36 per Year
3. Ramp Loading and Ramp Unloading	6 per Week
4. Step Loading Increase	1 per Week
5. Step Loading Decrease	1 per Week
6. Step Loading Reduction to Auxiliary Load	4 per Year
7. Reactor Trip From Full Power	10 per Year

Table 4-9

Reactor Coolant System Codes and Classifications

Component	Code	Classification
Reactor Vessel	ASME[a] III	Class A
Steam Generator	ASME[a] III	Class A
Pressurizer	ASME[a] III	Class A
Reactor Coolant Pump		
~~Volute and~~ Casing	ASME[a] III	Class A
Motor	IEEE,[b] NEMA,[c] and ASA[d]	
Coolant Piping	ASA[e] B31.1-1955 and Associated Nuclear Code Cases	

[a] American Society of Mechanical Engineers, Boiler and Pressure Vessel Code. Section III covers Nuclear Vessels.

[b] Institute of Electrical and Electronics Engineers.

[c] National Electrical Manufacturers Association.

[d] American Standards Association No. C50.2-1955 and C50.20-1954.

[e] American Standards Association No. B31.1.

TABLE 4-8, 4-9

4-33   00 00536

Table 4-10

Materials of Construction

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-302, Grade B
	Pressure Forgings	A-508, (Code Case 1332)
	Cladding, Stainless Weld Rod	SA-371, ER 308 SA-298, E 308
	Thermal Shield and Internals	SA-240, Type 304 and Inconel-X
Steam Generator	Pressure Plate	SA-212, Grade B
	Pressure Forgings	SA-105, Grade II
	Cladding for Heads, Stainless Weld Rod	SA-371, ER 308 SA-298, E 308
	Cladding for Tube Sheets	Ni-Cr-Fe
	Tubes	SB-163 (Code Case 1336)
Pressurizer	Shell, Heads, and External Plate	SA-212, Grade B
	Forgings and Nozzles	SA-105, Grade II
	Cladding, Stainless Weld Rod	SA-371, ER-308 SA-298, E 308
	Internal Plate	SA-240, Type 304
	Internal Piping	SA-312, Type 304
Piping	28 in. and 36 in.	
	Base Material to ASTM	A-212, Grade B, or A-106, Grade C
	Clad on Inner Surface to ASTM	A-371
	Using Electrode	Type ER-308
	10 in.	
	ASTM	A-376, Type 316, and A-403, Grade WP 316

TABLE 4-10

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Table 4.11

References For Figure 4-4
Increase in Transition Temperature Due to
Irraidation 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 of Progress, "Irradiation Effects on Reactor Structural Materials", 11-1-64/1-31-65.	A302B	Plate	550	3×10^{19}	135

[a] Transverse specimens.

TABLE 4-11

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Table 4-11 (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. 12, Oct. 1962, p 465-S.	A302B	Weld	500 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. 1, 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

00 00339
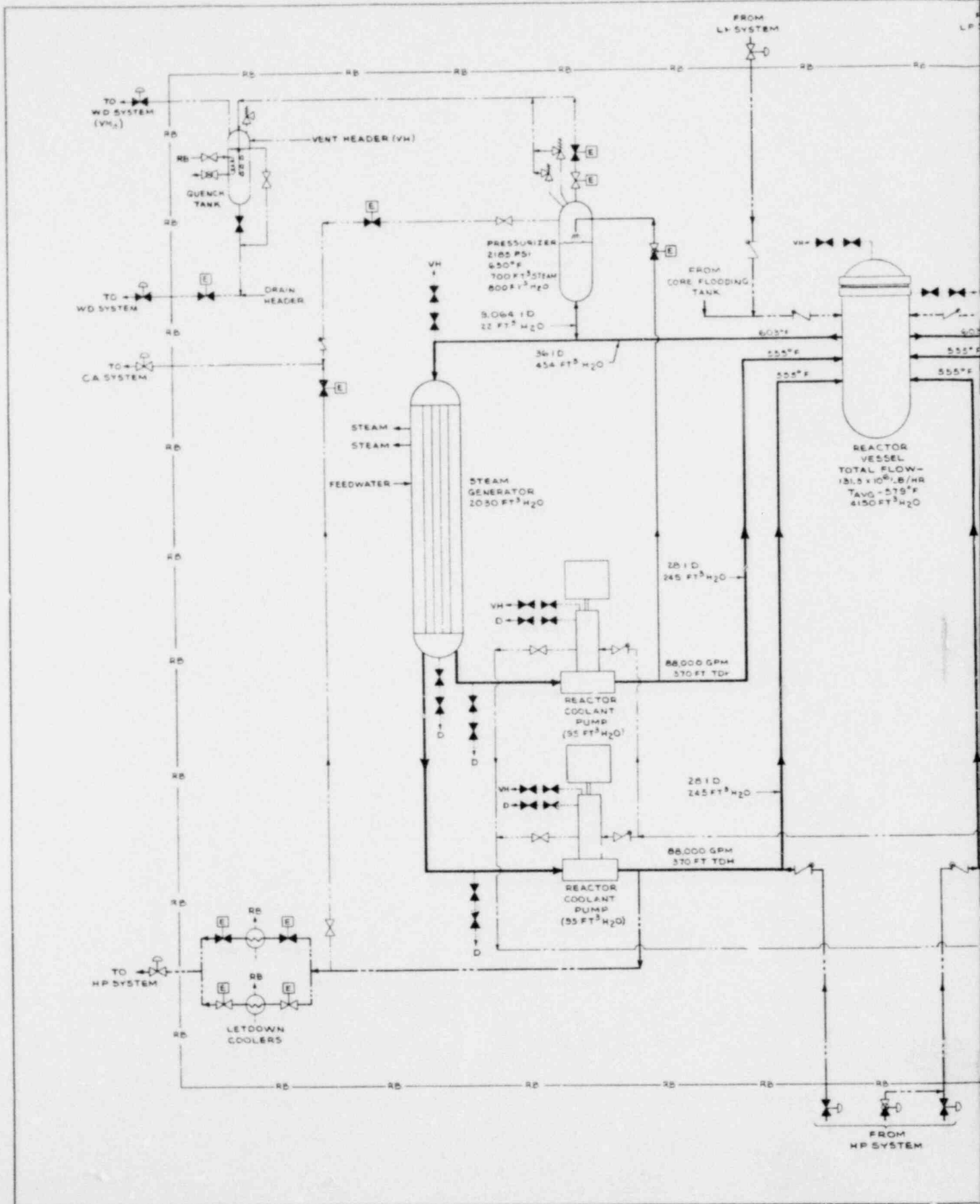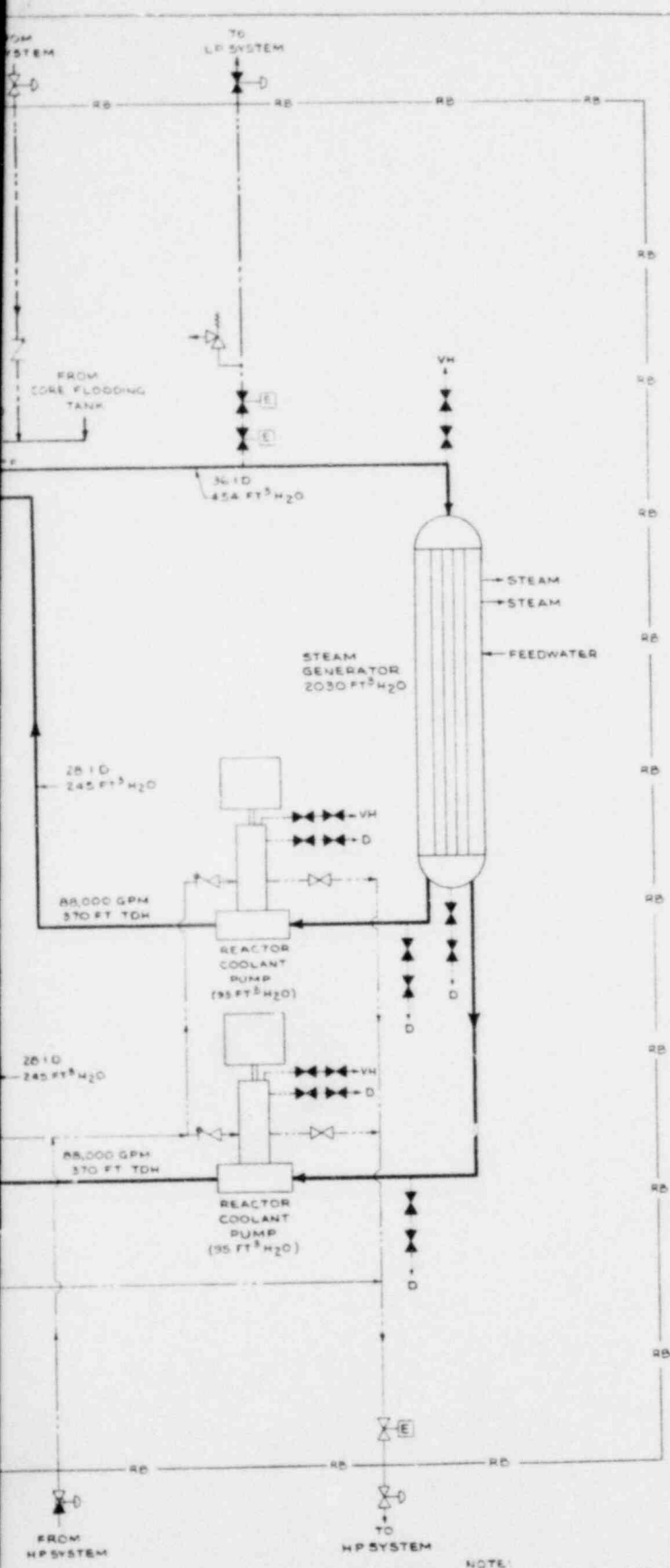
REACTOR COOLANT SYSTEM

NOTE:
FOR LEGEND NOMENCLATURE
SEE FIGURE 9-1
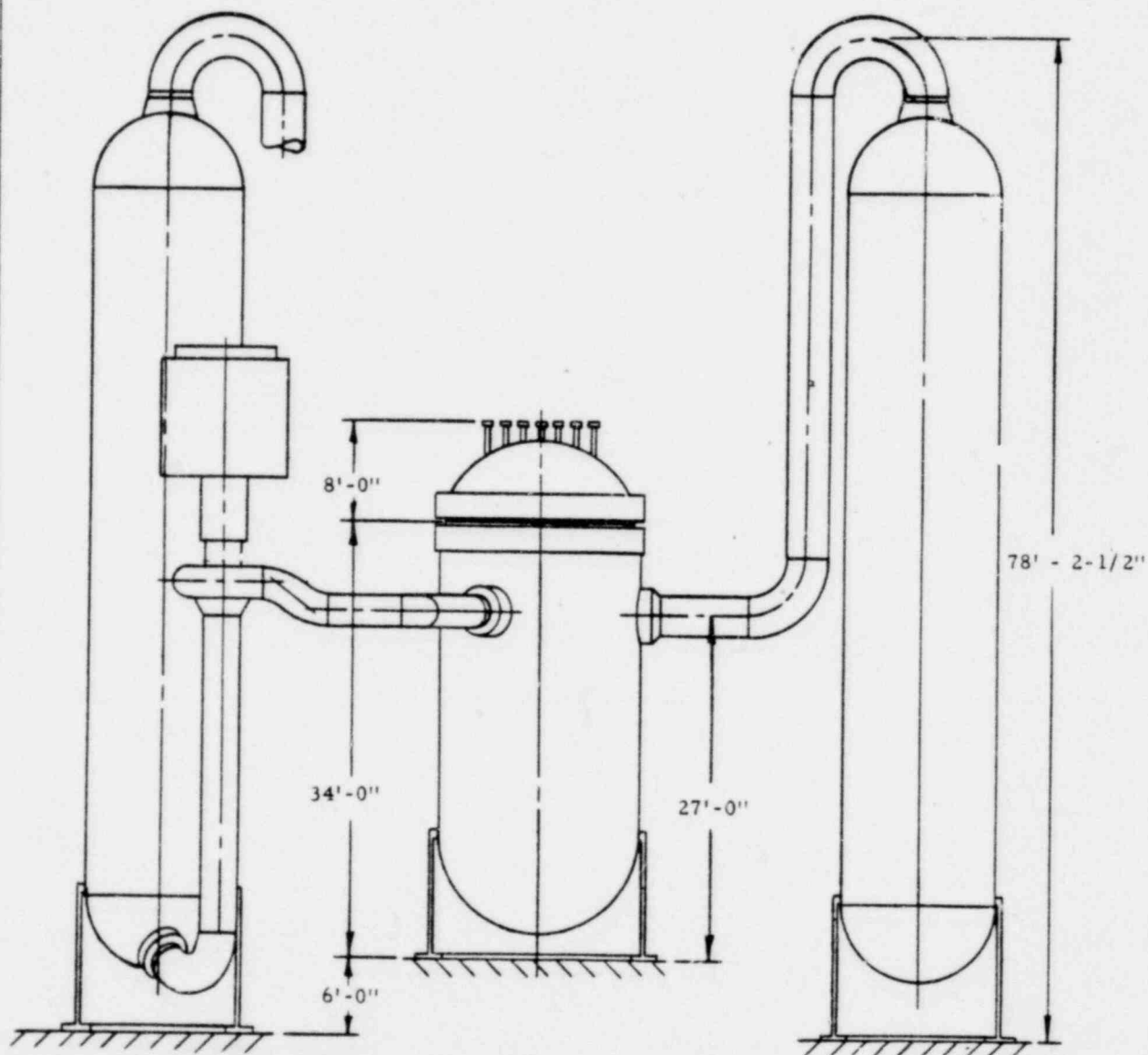
00 00341

REACTOR COOLANT SYSTEM ARRANGEMENT ELEVATION

OCONEE NUCLEAR STATION
FIGURE 4-2

30' - 6-9/16"

3' - 9-13/16"

5' - 1-9/16"

2' - 3-9/16"

12' - 2"

11' - 8-1/16"

60° 60°
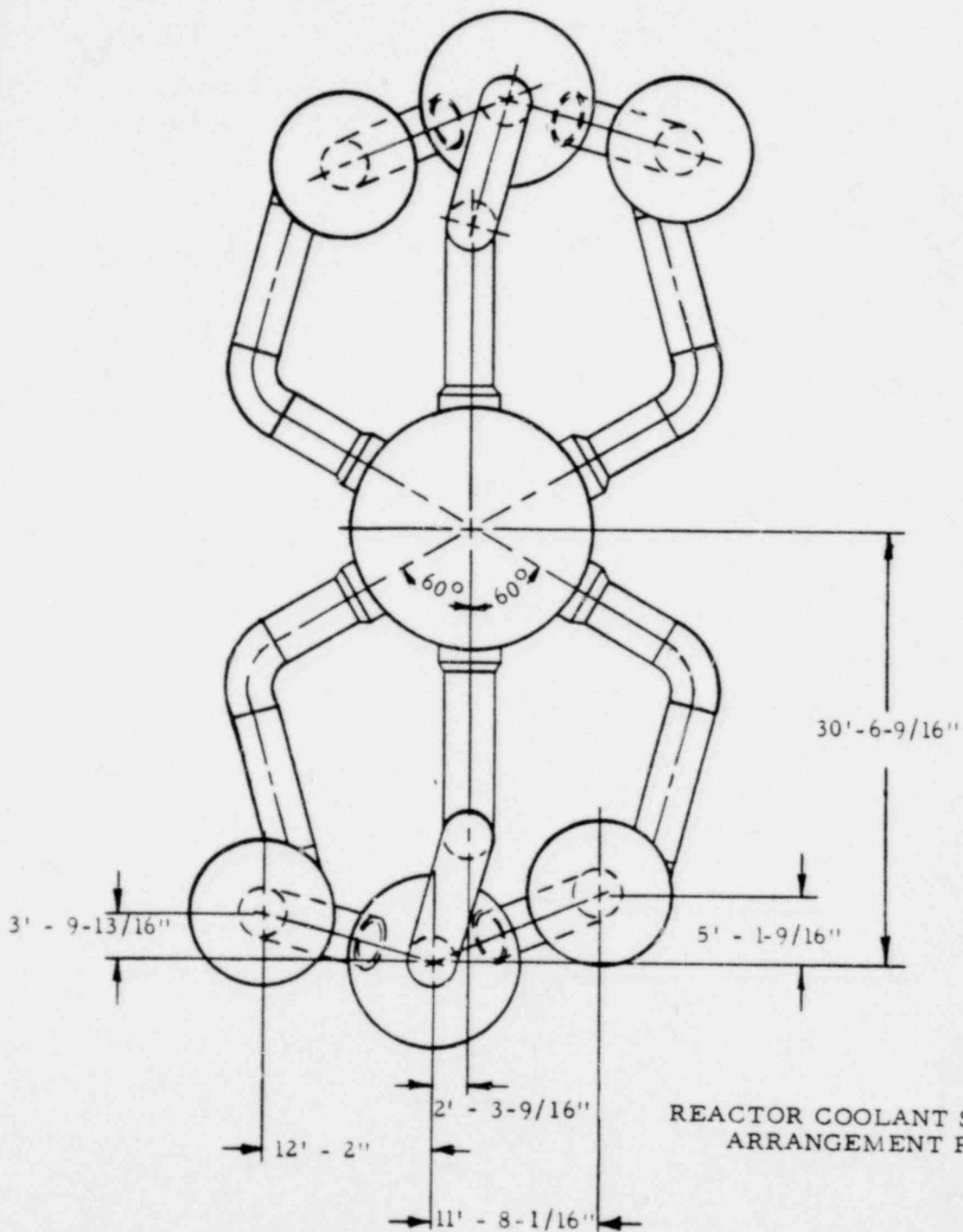
REACTOR COOLANT SYSTEM
ARRANGEMENT PLAN

OCONEE NUCLEAR STATION

FIGURE 4-3

00 00343
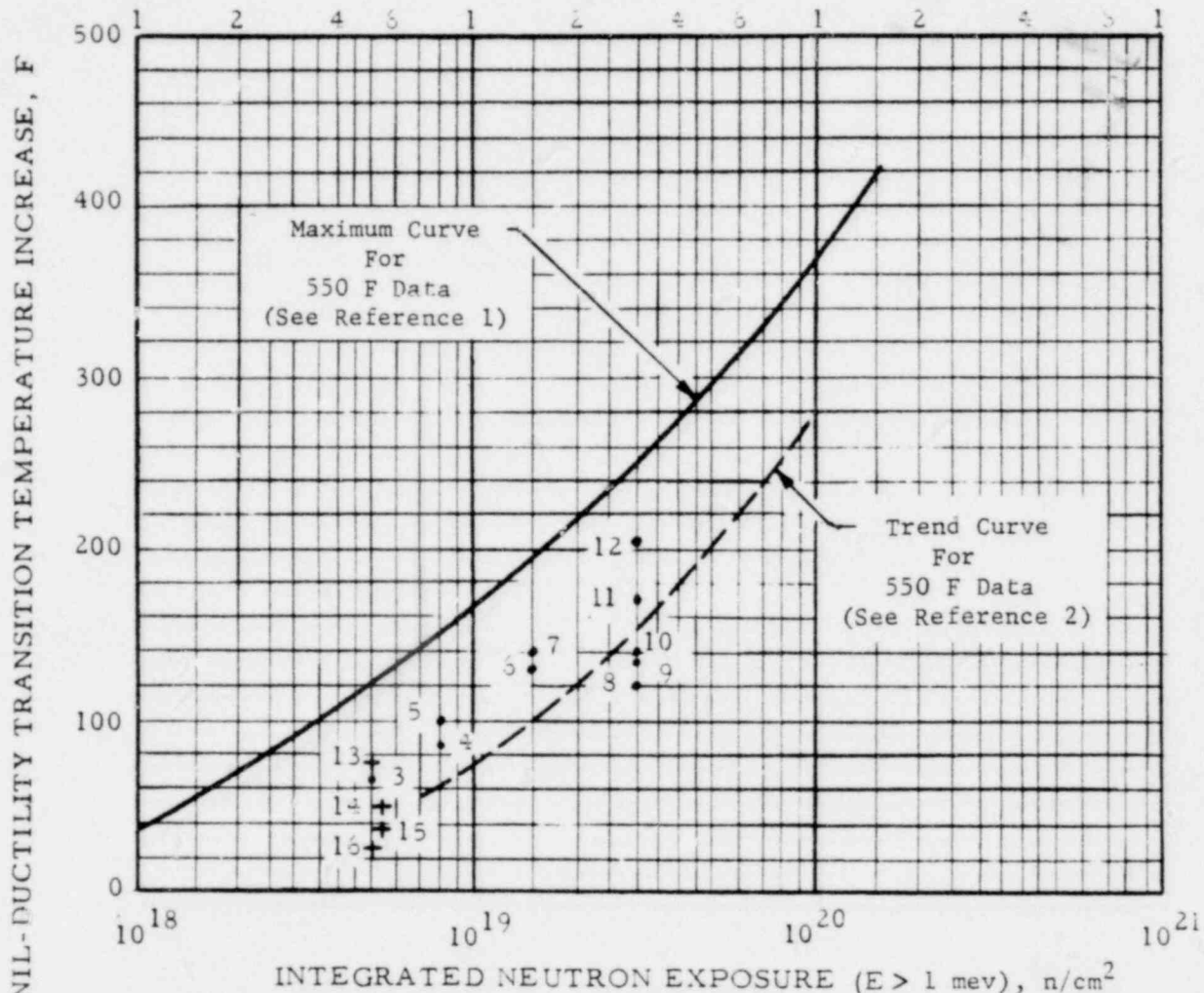
NIL-DUCTILITY TRANSITION TEMPERATURE INCREASE VERSUS
INTEGRATED NEUTRON EXPOSURE
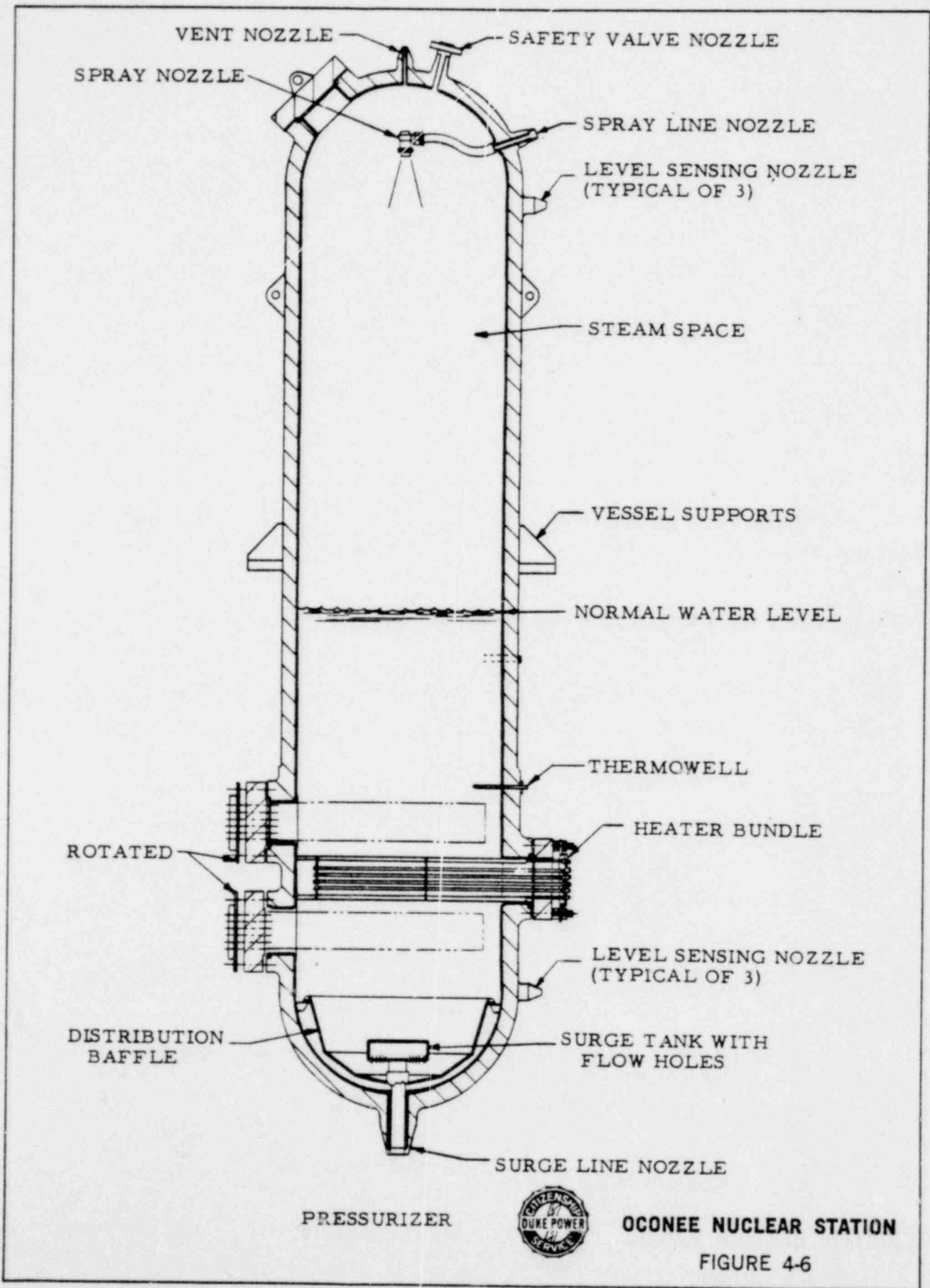FOR A302B STEEL

OCONEE NUCLEAR STATION
FIGURE 4-4

00 00344
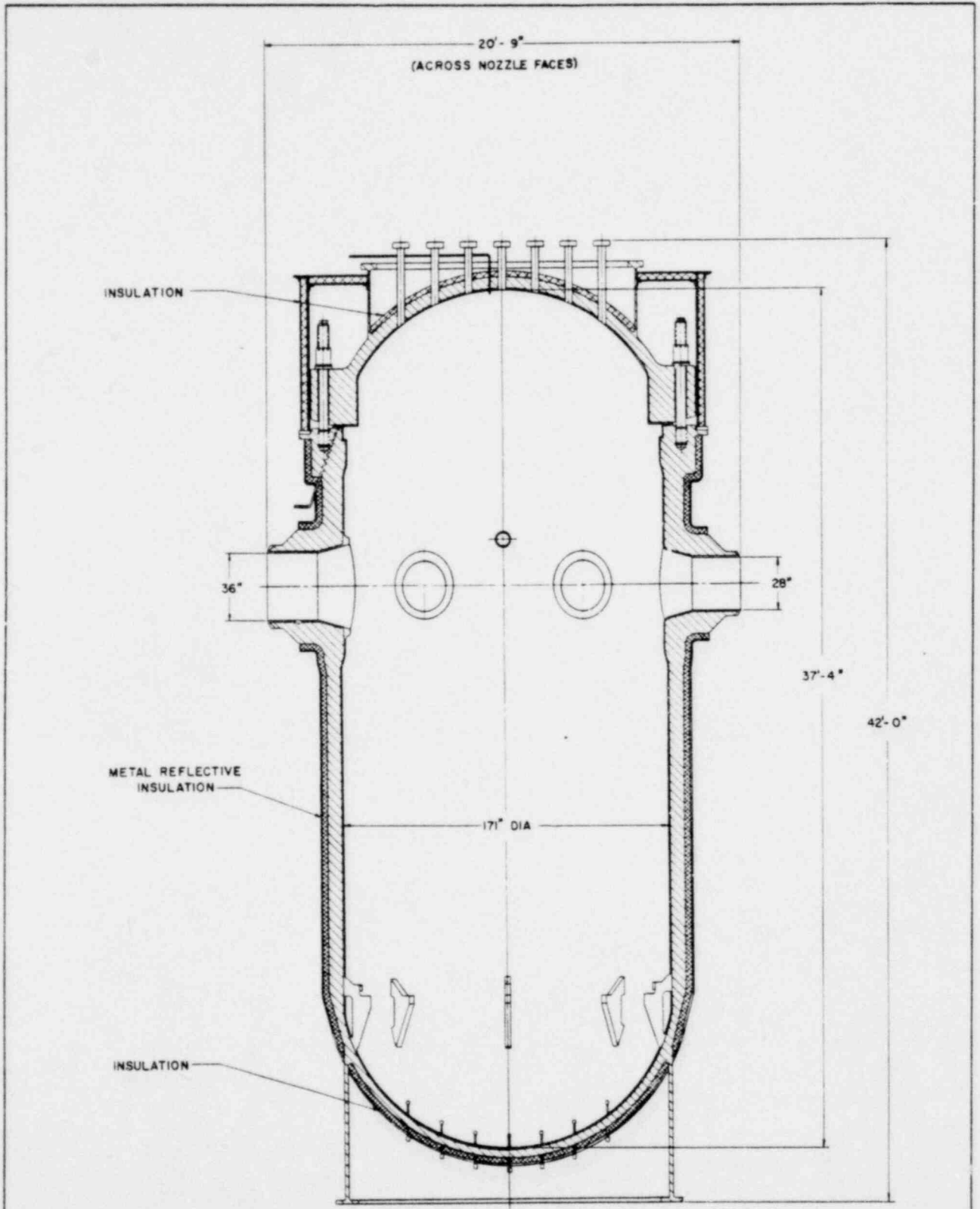
REACTOR VESSEL

OCONEE NUCLEAR STATION

FIGURE 4-5

REV: 4-1-67
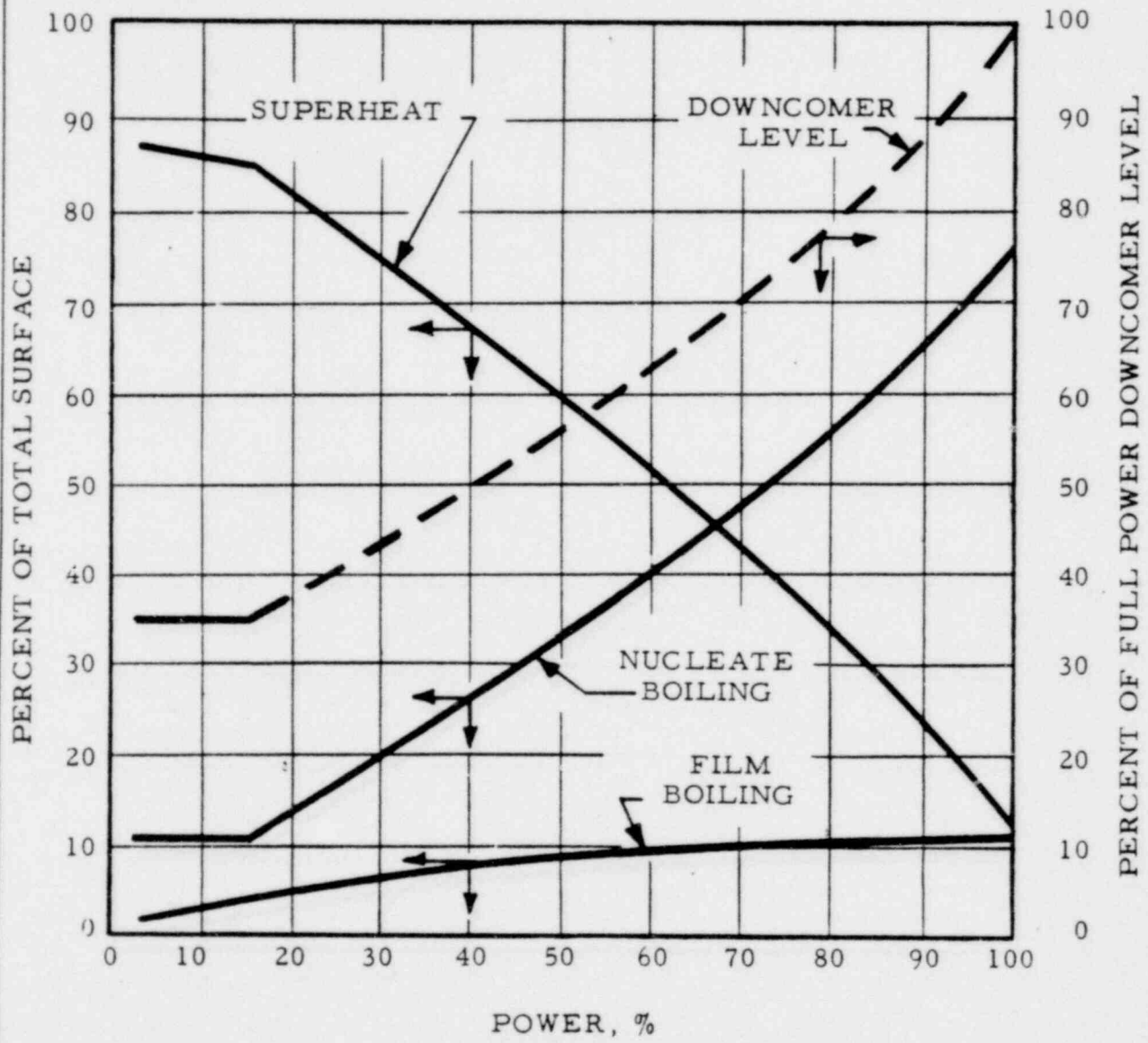DELETED NOZZLE

PRESSURIZER — OCONEE NUCLEAR STATION

FIGURE 4-6
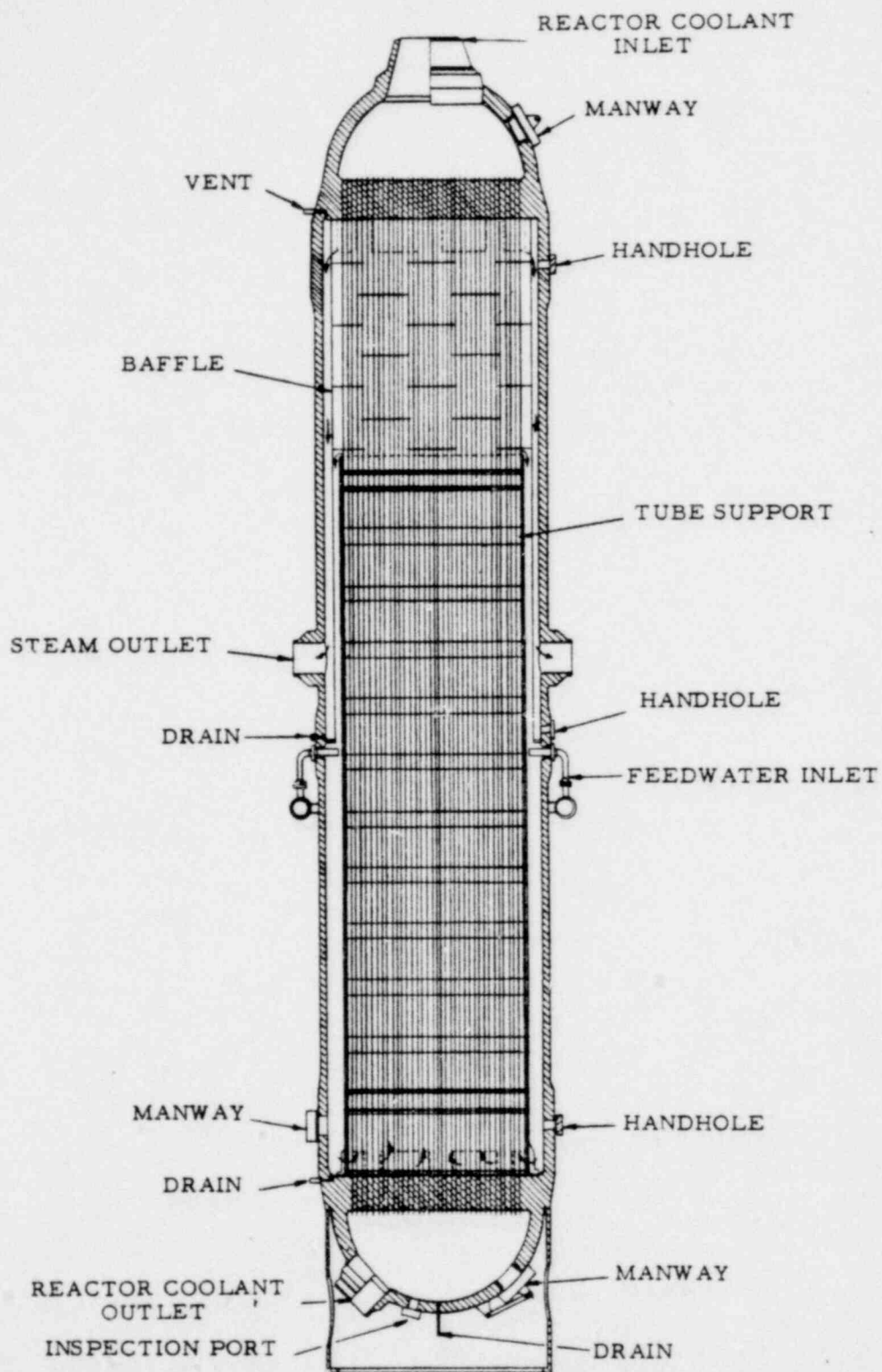
STEAM GENERATOR

OCONEE NUCLEAR STATION
FIGURE 4-7

000 347

STEAM GENERATOR HEATING REGIONS

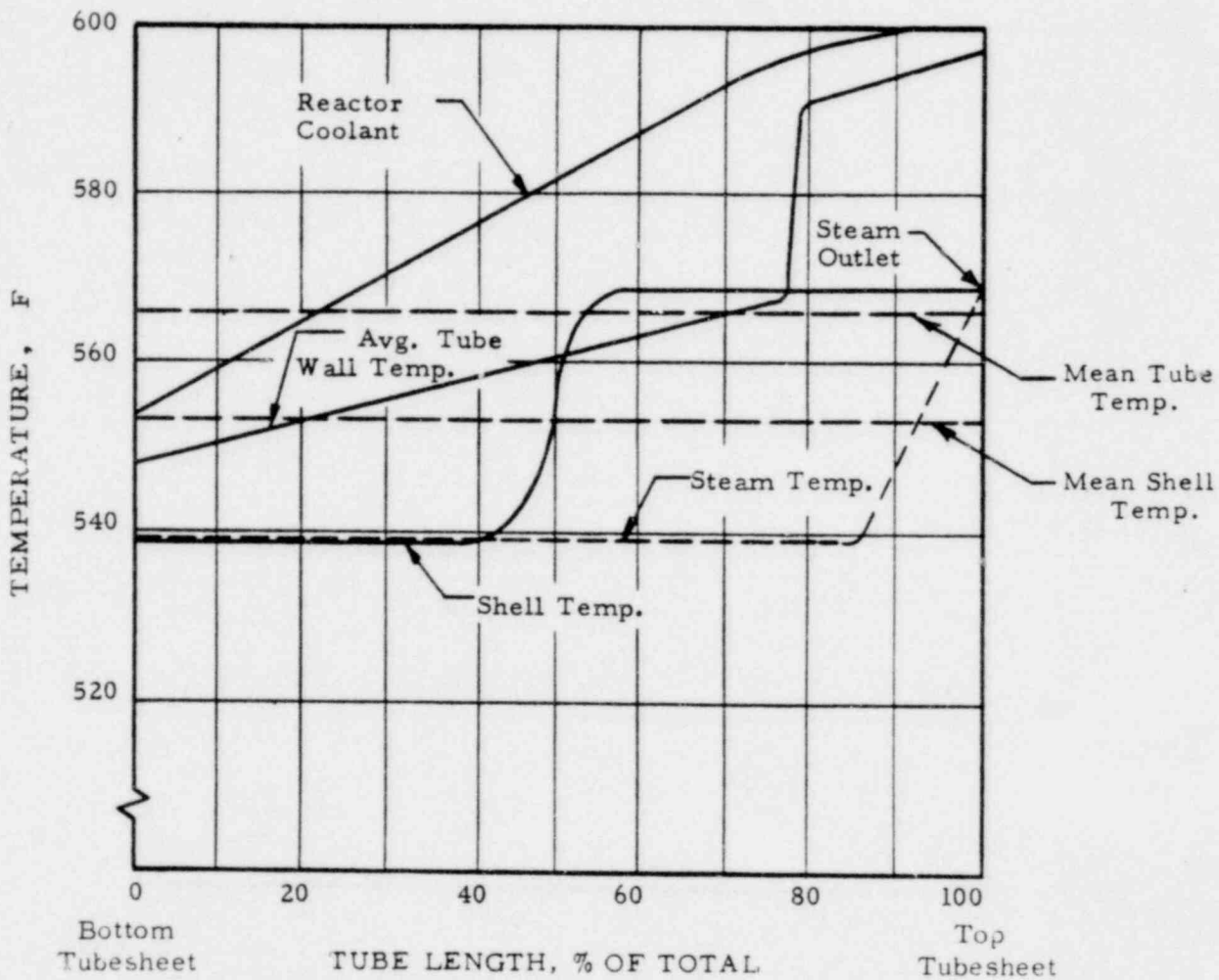OCONEE NUCLEAR STATION
FIGURE 4-8
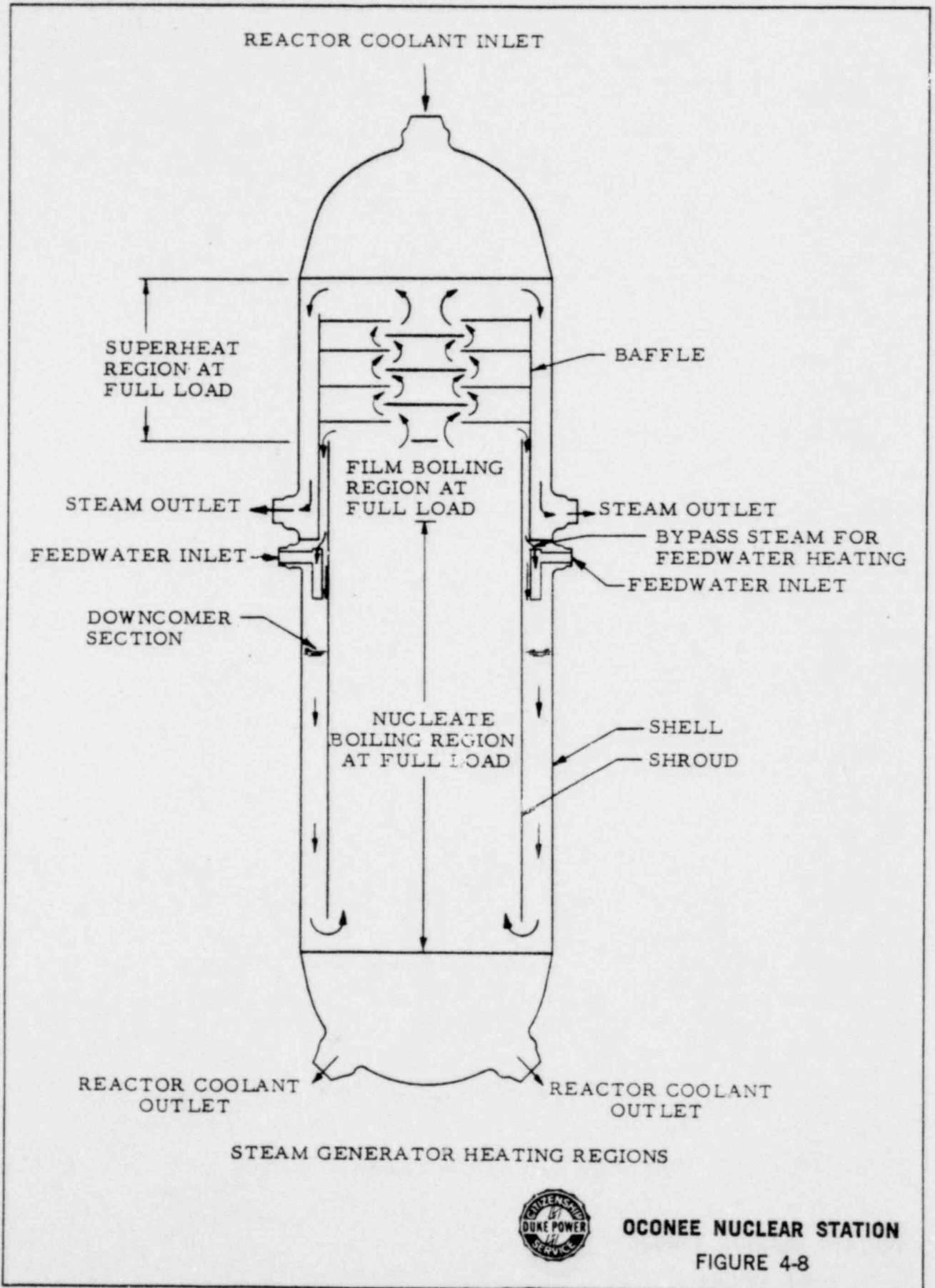
STEAM GENERATOR HEATING SURFACE AND
DOWNCOMER LEVEL VERSUS POWER

OCONEE NUCLEAR STATION

FIGURE 4-9
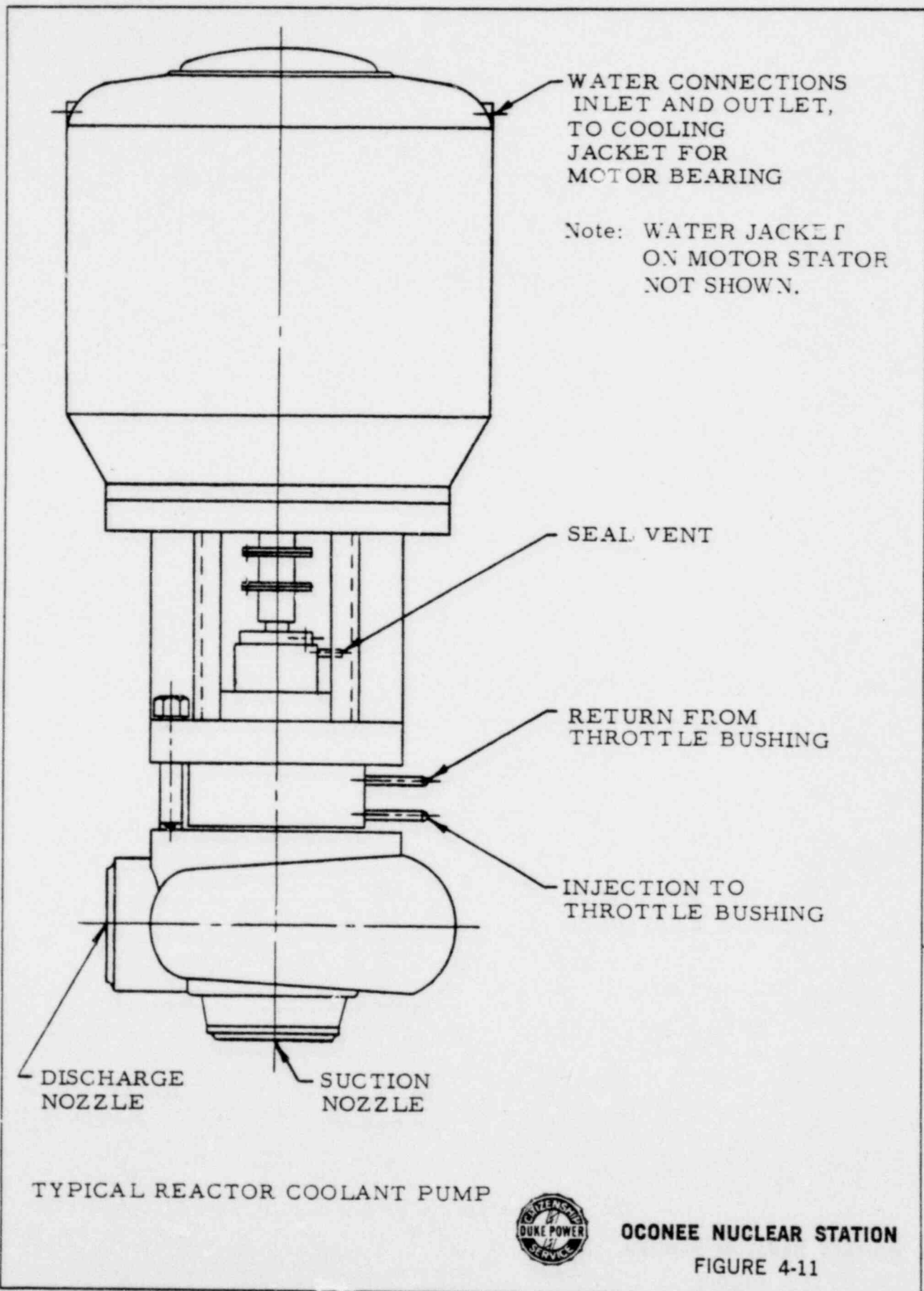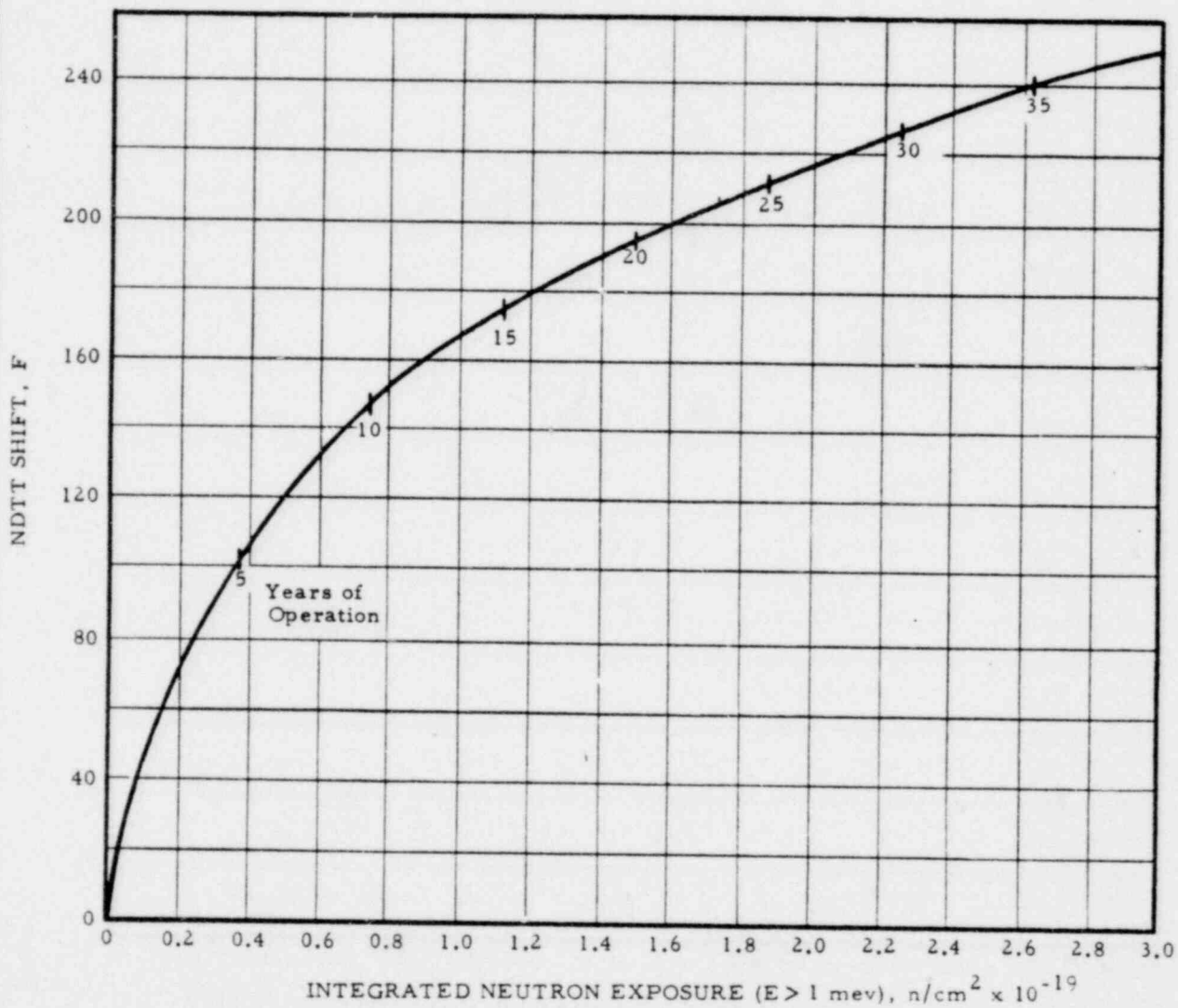
STEAM GENERATOR TEMPERATURES

OCONEE NUCLEAR STATION
FIGURE 4-10

000 550

WATER CONNECTIONS
INLET AND OUTLET,
TO COOLING
JACKET FOR
MOTOR BEARING

Note: WATER JACKET
ON MOTOR STATOR
NOT SHOWN.

SEAL VENT

RETURN FROM
THROTTLE BUSHING

INJECTION TO
THROTTLE BUSHING

DISCHARGE
NOZZLE

SUCTION
NOZZLE

TYPICAL REACTOR COOLANT PUMP

OCONEE NUCLEAR STATION

FIGURE 4-11

000 351

PREDICTED NDTT SHIFT VERSUS REACTOR
VESSEL IRRADIATION

OCONEE NUCLEAR STATION
FIGURE 4-12

000 352