4.0 REACTOR COOLANT SYSTEM

The reactor coolant system, shown on Figures 4.2-1 and 4.2-1A, consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, connecting piping, pressurizer safety and relief valves, and relief tank, necessary for operational control. While instrumentation is a part of, and is shown on the flow diagram of the reactor coolant system, all instrumentation is discussed in Chapter 7.

4.1 DESIGN BASES

4.1.1 Performance Objectives

The principal design data for the reactor coolant system are given in Table 4.1-1.

The reactor coolant system transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Demineralized light water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal hydraulic performance presented in Chapter 3. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The reactor coolant system provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits, to acceptable values, any uncontrolled release to the secondary system or to other parts of the plant under conditions of either normal or abnormal reactor behavior. During transient operation, the system's heat capacity attenuates thermal transients generated by the core or steam generators. The reactor coolant system accommodates coolant volume changes within the protection system criteria presented in Chapter 7.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal-hydraulic effects are reduced to a safe level during the pump coastdown, which would result from a loss-of-flow situation. The layout of the system assures natural circulation capability following a loss of flow to permit decay heat removal without overheating the core. Part of the system's piping serves as part of the emergency core cooling system to deliver cooling water to the core during a loss-of-coolant accident.

4.1.2 Application of Design Criteria

The reactor coolant system is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards for material selection, design, fabrication and inspection conform to the applicable provisions of recognized codes and good nuclear practice (Section 4.1.6). Details of the quality assurance program test procedures and inspection acceptance levels are given in Sub-

Chapters 4.3 and 4.5. Particular emphasis was placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within tolerances appropriate to the application of the design methods of the code delineated in Section 4.1.6.

Reactor Coolant System piping and components containing operating pressure and their supporting structures are designed as seismic Class I. Details are given in Section 4.1.4.

The Reactor Coolant System is located in the containment, which was designed to seismic Class I criteria. This design also considered such events as accidents or other applicable natural phenomena. Details of the containment design are given in Chapter 5.

Records of the design, fabrication and construction of the major Reactor Coolant System components will be maintained for the life of the plant. Code records will be maintained for the mandatory period, and thereafter either by Westinghouse or the American Electric Power Company.

The applicable portions of the Missile Protection Criteria as stated in Sub-Chapter 1.4 apply to Class I equipment in this chapter. An additional discussion can be found in Section 4.2.4.

The operation of the reactor is such that the severity of a hypothetical ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling groups are inserted in the core at power. At full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to assure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value which precludes any resultant damage to the Reactor Coolant System pressure boundary, i.e., gross fuel dispersion in the coolant and possible excessive pressure surges.

The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core was evaluated as a theoretical, though not a credible accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the Reactor Coolant System and the reactor containment. The environmental consequences of rod ejection are less severe than from the hypothetical loss of coolant, for which public health and safety is shown to be adequately protected. Reference is made to Chapter 14.

4.1.3 Design Characteristics

Design data for the respective Reactor Coolant System components are listed in Tables 4.1-3 through 4.1-8 and Table 4.1-12.

4.1.3.1 Design Pressure

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valves set points, and the Protection System set point pressures are listed in Table 4.1-2. The selected design margin includes operating transient pressure changes from core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. Table 4.1-9 gives the design pressure drop of the Reactor Coolant System components.

4.1.3.2 Design Temperature

The design temperature for each component was selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 4.1-3 through 4.1-8.

4.1.3.3 Seismic Loads

The seismic loading conditions are established by the Operating Basis Earthquake (OBE) and Design Basis Earthquake (DBE). Definitions of these conditions are given in Chapter 2.

For the OBE loading condition, the Nuclear Steam Supply System (NSSS) is designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment needed for this purpose are required to remain operable. The seismic design for the DBE is intended to provide a margin in design that assures capability to shut down and maintain the reactor in a safe condition. In this case, it is necessary to ensure that the Reactor Coolant System components do not lose their capability to perform their safety function. This has come to be referred to as the “no-loss-of-function” criteria and the loading condition as the “no-loss-of-function” loading condition.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Sub-Chapter 2.9.

In addition, design stress limits associated with emergency and faulted conditions, as defined in Sub-Chapter 2.9, were applied to systems and components which were felt capable of liberating sufficient energy from pipe whip as to potentially damage the containment, increase the severity of a LOCA or damage a safeguard system. These included portions of the Main Steam and Feedwater piping, Steam Generator drains, Accumulator Piping, Containment Spray and Residual Heat Removal System Sprays, and other major piping systems within the containment. The loading combinations and stress limits criteria were based on tables in Sub-Chapter 2.9.

Design and construction practices in accordance with these criteria assure the integrity of the Reactor Coolant System under seismic loading.

4.1.3.4 Support Load Stresses

The criteria applied in the design of the principal Reactor Coolant System component supports, restraints, snubbers and guides are defined in Sub-Chapter 2.9. An integrated dynamic analysis of the primary loop piping, the NSSS equipment and NSSS equipment supports was performed. The integrated system was analyzed for both seismic and pipe rupture conditions. For the seismic analysis, building response spectra at building support interfaces were used as input. For the rupture condition, breaks at various critical locations in both the primary and secondary piping were considered. Time history forcing functions associated with the various breaks were used as input to the integrated dynamic model.

Results from these analyses were combined with operating conditions to obtain the resultant state of stress and strain in both the piping system and the support system. These results showed that the NSSS support system and primary loop piping system were both within an acceptable state of stress and strain for the postulated loading conditions.

4.1.4 Cyclic Loads

The reactor coolant system and its components are designed to accommodate 10 percent of full power step changes in plant load and 5 percent of full power per minute ramp changes over the range from 15 percent full power up to and including but not exceeding 100 percent of full power without reactor trip. The reactor coolant system can accept a complete loss of load from full power with reactor trip.

Reactor coolant system components were designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal power changes, reactor trip, and startup and shutdown operations. The number of thermal and loading cycles used for design purposes are given in Table 4.1-10. These thermal and loading cycles were also used for the Unit 1 rerating program except as noted in the table. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Sub-Chapter 4.3.

A renewed operating license extends the license term an additional 20 years for CNP, Units 1 and 2. This extension was justified based on design transient cyclic loads defined in Table 4.1-10. The reactor coolant system was originally qualified using a conservative estimate of design cycles for a 40 year life. However, design life is dependent in part on fatigue cycles, not years of service. In evaluations performed for CNP, the actual number of cycles was extrapolated to 60 years. For the major reactor coolant system components, the extrapolated numbers of cycles over a 60-year life will not exceed the design cycles. The actual transient cycles are tracked and documented to ensure they remain below the allowable number of design cycles, as further discussed in Chapter 15 of the UFSAR.

To provide the necessary high degree of integrity for the equipment in the reactor coolant system, the transient conditions selected for equipment fatigue evaluation were based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients and accident conditions. To a large extent, the specific transient operating conditions considered for equipment fatigue analyses were based upon engineering judgment and experience. The transients chosen are representative of transients which prudently should be considered to occur during plant operation and which are sufficiently severe or may occur frequently to be of possible significance to component cyclic behavior.

For fatigue calculation of Class I systems and components, 20 OBE occurrences of 20 cycles each for a total of 400 occurrences was considered acceptable. However, for the reactor vessel a more conservative 10 OBE occurrences of 20 cycles each for a total of 200 occurrences was considered acceptable.

In accordance with Technical Specifications, CNP tracks the number of transient occurrences listed in the following sections. Each transient condition is discussed in order to make clear the nature and basis for the various transients.

4.1.4.1 Heatup and Cooldown

For design evaluation, the heatup and cooldown cases are represented by continuous heatup or cooldown at a rate of 100°F per hour, which corresponds, to a heatup or cooldown rate under abnormal or emergency conditions. The heatup occurs from ambient to the no load temperature and pressure condition and cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will usually not be attained because of other limitations such as:

a. Criteria for prevention of non-ductile failure, which establish maximum permissible temperature rates of change as a function of plant pressure and temperature.

b. Slower initial heatup rates when using pumping energy only.

c. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

The heatup and cooldown rates, imposed by plant operating procedures, are limited to no more than 60°F per hour for heatup and 100°F per cooldown for normal operation. Ideally, heatup and cooldown would occur only before and after refueling. In practice, additional unscheduled plant cooldowns may be necessary for plant maintenance. The frequency of maintenance shutdowns is expected to decrease as the plant matures.

As experience was gained with Yankee-Rowe, the number of shutdowns decreased; for example Core II ran for a year from 1962 to 1963 with no cooldown. Table 4.1-11 is a summary of the Yankee-Rowe plant outage for the period 1964 through to 1969.

4.1.4.2 Unit Loading and Unloading

The unit loading and unloading cases considered for the original design are conservatively represented by a continuous and uniform ramp power change of 5% per minute between 15% load and full load. This load swing is the maximum possible consistent with operation with automatic reactor control. The reactor coolant temperature will vary with load as programmed by the temperature control system. The number of each operation is specified at 18,300 times over the life of the plant.

For Unit 1 rerating conditions these cases are conservatively represented by a continuous and uniform ramp power change of 5% per minute between 0% load and full load. The number of each operation is specified at 11,680 times over the life of the plant.

The Unit 1 Babcock & Wilcox (BWI) Model 51R replacement steam generators have been analyzed for both the pre-Measurement Uncertainty Recapture (MUR) power uprate power rating (3264 MWt) and the 3600 MWt power uprate condition for a continuous and uniform ramp power change of 5% per minute between 0% and full load, with 11,680 cycles as described above.

4.1.4.3 Step Increase and Decrease of 10%

The ±10% step change in load demand is a control transient, which is assumed to be a change in turbine control valve opening, which might be occasioned by disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without a reactor trip following a ±10% step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15% and 100% full load, the power range for automatic reactor control. In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that the peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed set point, at a sufficiently slow rate, to prevent excessive pressurizer pressure decrease.

Following a step load decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the reactor coolant system average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine generator load as determined by the first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step load increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient. The number of each operation is specified at 2000 times over the life of the plant.

4.1.4.4 Large Step Decrease in Load

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate the secondary side steam dump system. The plant is designed to accept a step decrease of 50% from full power by use of a steam dump system that provides a heat sink to accept approximately 40% of full load steam flow. The remaining 10% of the total step change is assumed by the reactor rod control system as noted in Chapter 3.

The number of occurrences of this transient is specified at 200 times over the life of the plant. Reference to the Yankee-Rowe record indicates that this basis is adequately conservative.

4.1.4.5 Loss of Load

This transient applies to a step decrease in turbine load from full power occasioned by the loss of turbine load without immediately initiating a reactor trip and represents the most severe transient on the reactor coolant system. In this assumed case, the reactor and the turbine eventually trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System.

The number of occurrences of this transient is specified at 80 times over the life of the plant. Since redundant means of tripping the reactor upon turbine trip are provided as part of the Reactor Protection System, transients of this nature are not expected.

4.1.4.6 Loss of Power

This transient applies to a blackout situation involving the loss of offsite electrical power to the station and a reactor and turbine trip, on low reactor coolant flow, culminating in a complete loss of plant electrical power. Under these circumstances, the reactor coolant pumps are de-energized and, following the coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators, which at this time are receiving feedwater from the Auxiliary Feedwater

System operating from Diesel Generator power. Steam is removed for reactor cooldown through atmospheric pilot-operated relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times over the life of the plant.

4.1.4.7 Loss of Flow

This transient applies to a partial loss of flow accident from full power in which a reactor coolant pump is tripped out of service as a result of a loss of power to that pump. The consequences of such an accident at a high power level are a reactor and turbine trip on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal results in reactor coolant, at cold leg temperature, being passed through the steam generator and cooled still further. This cooler water then passes through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified at 80 times over the life of the plant.

4.1.4.8 Reactor Trip from Full Power

A reactor trip from full power may occur for a variety of causes resulting in temperature and pressure transients in the Reactor Coolant System and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of secondary steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the Reactor Protection System causes the control rods to move into the core.

The number of occurrences of this transient is specified at 400 times over the life of the plant.

4.1.4.9 Hydrostatic Test Conditions

The pressure tests are outlined below:

a. Primary Side Hydrostatic Test Before Initial Startup at 3107 psig

   The pressure tests covered by this section include both shop and field hydrostatic tests, which occurred as a result of component, or system testing. This hydro test was performed at a water temperature, which is compatible with reactor vessel material Design Transition Temperature (DTT) requirements and a maximum test pressure of 3107 psig. In this test, the primary side of the steam generator was pressurized to 3107 psig coincident with the secondary side pressure of 0 psig. The Reactor Coolant System is designed for 5 cycles of this hydro test.

b. Secondary Side Hydrostatic Test Before Initial Startup

The secondary side of the steam generator was pressurized to 1356 psig with a minimum water temperature of 70°F coincident with the primary side of 0 psig. The Unit 1 steam generators may experience 5 cycles of this test. The Unit 2 steam generators may experience 5 cycles of this test.

4.1.4.10 Primary Side Leak Test

Subsequent to each time the primary system has been opened, a leak test will be performed. During this test the primary system pressure is, for design purposes, assumed to be raised to 2500 psia with the system temperature above Design Transition Temperature, while the system is checked for leaks.

In actual practice, the primary system will be pressurized to below 2500 psia to prevent the pressurizer safety valves from lifting during the leak test.

4.1.4.11 Pressurizer Surge and Spray Line Connections

The surge and spray nozzle connections at the pressurizer vessel are subject to cyclic temperature changes resulting from the transient conditions described previously. The various transients are characterized by variations in reactor coolant temperature, which in turn result in water surges into or out of the pressurizer.

The surges manifest themselves as changes in system pressure which, depending upon whether an increase or decrease in pressure occurs, result in introducing spray water into the pressurizer to reduce pressure or in actuating the pressurizer heaters to increase pressure to the equilibrium value. To illustrate a load change cycle as it affects the pressurizer, consider a design step increase in load. The pressurizer initially experiences an outsurge with a drop in system pressure which actuates the pressurizer heaters to restore system pressure. As the Reactor Control System reacts, the reactor coolant temperature is increased which causes an insurge into the pressurizer raising system pressure. As pressure is increased, the heaters go off and, at the pressure set point; the spray valves open to limit the pressure rise and restore system pressure. Thus the pressurizer surge nozzle is subjected to a temperature increase on the outsurge followed by a temperature decrease on the insurge during this load transient. The pressurizer spray nozzle is subjected to a temperature decrease when the spray valve opens to admit reactor coolant cold leg water into the pressurizer. The pressurizer experiences a reverse situation during a load decrease transient, i.e., an insurge followed by an outsurge.

It is assumed that the spray valve opens to admit spray water into the pressurizer once, at the design flowrate, for each design step change in plant load. Thus the number of occurrences for the spray nozzle corresponds to that shown for the other components in Table 4.1-10.

During plant cooldown, spray water is introduced into the pressurizer to cool it down. The maximum pressurizer cooldown rate is specified at 200°F per hour, which is twice the rate specified for the other Reactor Coolant System components.

4.1.4.12 Accident Conditions

The effect of the accident loading was evaluated in combination with normal loads to demonstrate the adequacy to meet the stated plant safety criteria.

A brief description of each accident transient considered follows. In each case one occurrence is evaluated.

a. Reactor Coolant Pipe Break

This accident involves the rupture of a Reactor Coolant System pipe resulting in a loss of primary coolant. It was conservatively assumed that the system pressure and temperature would be reduced rapidly and that the Safety Injection System would be initiated to introduce 70°F water into the Reactor Coolant System. The safety injection signal will also result in a turbine and reactor trip. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal is still at no-load temperature conditions when the 70°F safety injection water is introduced into the system.

b. Steam Line Break

For component evaluation, the following conservative conditions were considered:

1. The reactor is initially in a hot, no-load, just critical condition, assuming all rods in except the most reactive rod, which is assumed to be stuck in its fully withdrawn position.

2. A steam line break occurs inside the containment resulting in a reactor and turbine trip.

3. Subsequent to the break, there is no return to power and the reactor coolant temperature cools down to 212°F.

4. The centrifugal charging pumps restore the reactor coolant pressure to 2500 psia.

The above conditions result in the most severe temperature and pressure variations which the component will encounter during a steam break accident.

c. Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to a safety injection signal on low pressurizer pressure. When the accident occurs, some of the reactor coolant blows down into the affected steam generator causing the level to rise. If the level rises to a pre-selected setpoint, a high level alarm will occur and the feedwater regulating valve will close.

It is expected this accident will result in a transient which is no more severe than that associated with a reactor trip. For this reason, it requires no special treatment in so far as fatigue evaluation is concerned. Further detail about the sequence of events may be found in Section 14.2.4 (Units 1 and 2).

4.1.5 Service Life

The service life of the Reactor Coolant System pressure containing components depends upon the end-of-life material radiation damage, unit operational thermal cycles, design and manufacturing quality standards, environmental protection, maintenance standards and adherence to established operating and maintenance procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and therefore it is the only component which is subject to material radiation damage effects.

The NDTT shift of the vessel material and welds during service due to radiation damage effects is monitored by a radiation damage surveillance program. Details are given in Sub-Chapter 4.5.

Reactor vessel design was based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material as a result of operation.

To establish the service life of the Reactor Coolant System components as required by the ASME (Section III) Boiler and Pressure Vessel Code for "A" vessels, unit operating conditions were established for the initial 40 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients. The numbers of operating transients during the 60-year licensed life are not projected to exceed the number of transients assumed for the initial plant design life.

The number of thermal and loading cycles used for design purposes is listed in Table 4.1-10.

4.1.6 Codes and Classifications

Pressure-containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-12. Refer to Sub-Chapter 4.5 for a discussion of Inservice Inspection.

Reactor Coolant System piping has been designed and supported in accordance with the USAS B31.1-1967 Code for Pressure Piping. The Code requirement that the piping shall be arranged and supported with consideration of vibration was met by means of variable spring hangers, rigid supports, constant support hangers, pipe anchors, guides and snubbers. The Code does not specifically require any vibrational test programs. However, during the normal course of the preoperational test program, specific attention was directed at evaluating possible vibration problems during performance of specific transients associated with the required preoperational tests. Excessive vibrations or deficiencies, determined by visual examinations, which were indicative of possible vibration problems, were investigated and corrected when necessary. This was done to verify that the piping and piping restraints within the Reactor Coolant System pressure boundary were adequately designed to withstand dynamic effects resulting from transient conditions.

4.2 SYSTEM DESIGN AND OPERATION

4.2.1 General Description

The Reactor Coolant System consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, loop piping and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system is shown in Figures 4.2-1 and 4.2-1A.

Reactor Coolant System design data are listed in Tables 4.1-1 through 4.1-8 and Table 4.1-12.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray, to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Chapter 7. Spring-loaded safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

4.2.2 Components Description

4.2.2.1 Reactor Vessel

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core support structures, control rods, thermal shield and other parts directly associated with the core. The reactor vessel closure head contains head adaptors. These head adaptors are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contain Acme threads for the assembly of control rod drive mechanisms (Unit 1 and Unit 2 design has eliminated the Core Exit Thermocouple (CET) ACME threads). A dedicated nozzle, near the center of the reactor head, connects to vent piping, which vents to the upper containment volume, to provide reactor vessel head venting of non-condensable gas while maintaining adequate core cooling and containment integrity (Both Unit 1 and Unit 2 have a dedicated reactor head vent nozzle). For further details see Sub-Section 4.2.2.6. The seal arrangement at the upper end of these adaptors consists of omega seal weld to the CRDM pressure housing. The upper end of the instrument adaptor consists of a mechanical sealing assembly. The vessel has inlet and outlet nozzles located in a horizontal plane just below the vessel flange but above the top of the core. Coolant enters the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles.

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear in-core detection instrumentation. Each tube is attached to the inside of the bottom head by a partial penetration weld.

The reactor vessel is designed to provide the smallest and most economical volume required to contain the reactor core, control rods and the necessary supporting and flow-directing internals. Inlet and outlet nozzles are spaced around the vessel. Outlet nozzles are located on opposite sides of the vessel to facilitate optimum layout of the Reactor Coolant System equipment. The inlet nozzles are tapered from the coolant loop-vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leak-off connections; one between the inner and outer ring, and one outside of the outer O-ring. Piping and associated valving are provided to direct any leakage to the reactor coolant drain tank. Leakage will be indicated by a high-temperature alarm from a detector in the leakoff line.

Ring forgings have been used in the following areas of the reactor vessel:

A. Vessel Flange

B. Eight Primary Nozzles

C. The Unit 1 and Unit 2 closure heads are one-piece forgings.

Core structural load bearing members were made from annealed type 304 stainless steel, so there is no possibility that they may become furnace sensitized[2]. The only exception is the core barrel itself, which required stress relief during manufacture at temperatures over 750°F, to minimize the possibility of severe sensitization while maintaining the necessary conditions for relieving residual fabrication stresses.

Other pressure or strength bearing stainless steel components or parts in the reactor vessel and associated Reactor Coolant System that may become furnace sensitized [1] during the fabrication sequence include:

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[2] The term "furnace sensitized" is interpreted as austenitic stainless steel wrought material and weld metal components which have been post weld heat treated in accordance with ASME Section III requirements, and which on the basis of its composition and thermal history would not be expected to pass ASTM-A-393.

4.2.2.1.1 Unit 1

A. Reactor Vessel

1. Primary nozzle safe ends - Type 316 forging

B. Steam Generators

1. Primary nozzle safe ends - weld metal buttered ends - Type 316 LN forgings (Unit 1 Replacement Steam Generator). [3]

4.2.2.1.2 Unit 2

A. Reactor Vessel

1. Primary nozzle safe ends - Type 316 forging overlaid with weld prior to final post weld heat treatment.

2. Monitor Tubes.

B. Steam Generators

1. Primary nozzle safe ends - weld metal buttered ends.

Westinghouse has evaluated the use of sensitized stainless steel and reactor components in pressurized water reactors. The results of this evaluation are summarized in Reference 1, which covers the nature of sensitization, conditions leading to stress corrosion and associated problems with both sensitized and non-sensitized stainless steel. The results of extensive testing and service experience that justify the use of stainless steel in the sensitized condition for components in Westinghouse PWR Systems is presented in Reference 1.

Although Westinghouse testing and evaluation showed justification for the use of sensitized stainless steel, extra modifications were made during the fabrication of these vessels, when it was revealed that the work would not significantly affect the delivery schedules and would result in a somewhat more conservative design. Also, for the Unit 2 replacement steam generator lower assemblies the use of severely sensitized stainless steels was prohibited on the primary nozzle safe ends.[2]

The cylindrical portion of the reactor vessel below the refueling seal ledge is permanently insulated with a metallic reflective-type insulation supported from the reactor coolant nozzles. This insulation consists of inner and outer sheets of stainless steel spaced 3 inches apart with

[3] Through an R & D program performed by Babcock & Wilcox (BWI), it was demonstrated that the primary nozzle safe end material does not sensitize when subjected to the PWHT times that the Unit 1 Replacement Steam Generator (RSG) was subjected to.

multilayers of stainless steel. Removable panels of the metallic reflective insulation described above are provided for the reactor vessel head and closure region.

These panels are supported on the refueling seal ledge and vent shroud support ring. The rest of the closure head is insulated with removable panels of at least three inches of the reflective insulation described. The bottom head is also insulated with reflective insulation, which is not removable.

Schematics of the reactor vessel are shown in Figures 4.2-2 and 4.2-2A. The materials of construction are given in Table 4.2-1 and the design parameters are given on Table 4.1-3. A description of the reactor vessel internals is given in Chapter 3.

4.2.2.2 Pressurizer

The pressurizer provides a point in the Reactor Coolant System where liquid and vapor can be maintained in equilibrium under saturated conditions for control purposes.

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief and safety valve connections are located in the top head of the vessel. The heaters are removable for maintenance or replacement. A vent connection is provided on the piping ahead of the power-operated relief valves to vent non-condensible gases or steam from the pressurizer to the upper containment volume. For further details see Sub-Section 4.2.2.6.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line, which is attached to the bottom of the pressurizer, connects the pressurizer to the hot leg of one reactor coolant loop.

During an insurge, the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the set-point of the power-operated relief valves. The spray valves on the pressurizer are modulating, air operated, control valves. In addition, the spray valves can be operated manually by a switch in the control room. A small continuous spray flow is provided through a manual bypass valve around the power-operated spray valves to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excessive cooling of the spray piping.

During an outsurge, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during insurges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop. A screen at the surge line nozzle and baffles in the lower section of the pressurizer prevents cold insurge water from flowing directly to the steam/water interface and it assists mixing.

The volume of the pressurizer is equal to or greater than, the minimum volume of steam, water, or total of the two, which satisfies all of the following requirements:

1. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.
2. The water volume is sufficient to prevent the heaters from being uncovered during a step-load increase of ten percent of full power.
3. The steam volume is large enough to accommodate the surge resulting from the design step load reduction of full load with reactor control and steam dump without the water level reaching the high-level reactor trip point.
4. The steam volume is large enough to prevent water relief through the safety valves following a loss of load with the high water level initiating a reactor trip.
5. The pressurizer will not empty following reactor trip and loss of load.
6. The Emergency Core Cooling Signal will not be activated during reactor trip and turbine trip.

The general configuration of the pressurizer is shown in Figure 4.2-3 and the design data are given in Table 4.l-4.

4.2.2.2.1 Pressurizer Spray

Two separate, automatically controlled spray valves with remote manual overrides are used to regulate pressurizer spray. In parallel with each spray valve is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open, and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping to the pressurizer forms a water seal, which prevents steam buildup back to the control, valves. The spray rate from one valve is sufficient to prevent the pressurizer pressure from reaching the operating (set) point of the power relief valves during a step reduction in power level of ten percent full load.

The pressurizer spray lines and valves are large enough to provide adequate spray flow using as the driving force the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force. The line may also be used to assist in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

A flow path from the Chemical and Volume Control System to the pressurizer spray line is also provided. This additional facility provides auxiliary spray to the vapor space of the pressurizer during cooldown if the reactor coolant pumps are not operating. The thermal sleeve on the pressurizer spray connection is designed to withstand the thermal stresses resulting from the introduction of cold spray water.

4.2.2.2.2 Surge Line

The surge line is sized to limit the pressure drop during the maximum anticipated surge to less than the difference between the maximum allowable pressure in the reactor vessel and the loops (at the point of highest pressure) and the pressure in the pressurizer at the maximum allowable accumulation with the safety valves discharging.

The surge line and the thermal sleeves at each end are designed to withstand the thermal stresses resulting from volume surges of relatively hotter or colder water, which may occur during operation.

4.2.2.3 Pressurizer Relief Tank

The pressurizer relief tank condenses and cools the discharge from the pressurizer safety and relief valves, as well as several smaller relief valves. The tank normally contains water and a predominantly nitrogen atmosphere; however, provision is made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen.

The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any non-condensable gases from the Reactor Coolant System, which might collect in the pressurizer vessel.

Steam is discharged through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature. The tank is equipped with an internal spray and a drain, which are used to cool the tank following a discharge. The tank is protected against a discharge exceeding the design value by two rupture discs, which discharge into the reactor containment. The tank is carbon steel with a corrosion-resistant coating on the wetted surfaces. A flanged nozzle is provided on the tank for the pressurizer discharge line connection. This nozzle and the discharge piping and sparger within the vessel are austenitic stainless steel.

The tank design is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110 percent of the volume above the 100%-power pressurizer water level set-point. The tank is not designed to accept a continuous discharge from the pressurizer. The volume of water in the tank is capable of absorbing the heat from the assumed discharge, with an initial temperature of 120°F and increasing to a final temperature of 210°F. If the temperature in the tank rises above 126°F during plant operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the Waste Disposal System.

The spray rate is designed to cool the tank from 200°F to 120°F in approximately one hour following the design discharge of pressurizer steam. The volume of nitrogen gas in the tank is selected to limit the maximum pressure following a design discharge to 50 psig.

The rupture discs on the relief tank have a relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure is twice the calculated pressure resulting from the maximum safety valve discharge described above. The tank and rupture discs holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

Principal design parameters of the pressurizer relief tank are given in Table 4.1-4.

4.2.2.3.1 Discharge Piping

The discharge piping (from the safety and power-operated relief valves to the pressurizer relief tank) is sized to prevent back-pressure at the safety valves from exceeding 20 percent of the set-point pressure at full flow. The pressurizer safety and power relief valves discharge lines are stainless steel.

4.2.2.4 Steam Generators

The steam generators are vertical shell and U-tube heat exchangers with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Manways are provided for access to both sides of the divided head. Feedwater enters the steam generators and is distributed through a feedwater ring located just below the moisture separators. Thermal sleeves are provided in the feedwater piping elbows at the steam generator inlet (Unit 2 only). For the Unit 1 Babcock & Wilcox Model 51R replacement steam generators, the thermal sleeves are welded to a transition ring which is then welded to the main feedwater nozzle forging on the steam drum shell side of the steam generator. Feedwater flow is out of the top of the feedwater ring through “J” tubes, down between the steam generator shell and tube bundle wrapper and into the tube bundle just above the tube sheet. The “J” tubes prevent rapid drainage of the feedwater ring due to a drop in steam generator water level and thus eliminate or reduce the possibility of water hammer in the feedwater line. Steam is generated on the shell side of the tube bundle and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

The units are primarily constructed of carbon steel. The heat transfer tubes are Inconel, the primary side of the tube sheets are clad with Inconel, and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel.

The Unit 2 and Unit 1 steam generators of this type are shown in Figures 4.2-4 and 4.2-4a through 4.2-4e respectively. Design data are given in Table 4.1-5.

Each steam generator is designed to produce 25 percent of the steam flow required at full-power operation. The internal moisture-separating equipment is designed to insure that the moisture carryover will not exceed 0.045 percent by weight for Unit 1 and 0.15 percent by weight for Unit 2 under the following conditions:

a. Steady-state operation up to 105 percent of full-load steam flow, with water at the normal operating level.

b. Loading or unloading at a rate of five percent of full power steam flow per minute in the range from 15% to 105% of full load steam flow. The Unit 1 Babcock & Wilcox Model 51R replacement steam generators have been analyzed for both the pre-Measurement Uncertainty Recapture power uprate power rating (3264 MWt) and the 3600 MWt power uprate condition for a continuous and uniform ramp power change of 5% per minute between 0% and full load. The Unit 2 Model 51 replacement steam generators have been analyzed for both the pre-Measurement Uncertainty Recapture power uprate power rating (3425 MWt) and the 3600 MWt power uprate condition for a continuous and uniform ramp power change of 5% per minute between 0% and full load, with 11,680 cycles as described above.

c. A step-load change of ten percent of full power in the range from 15% to 105% full load steam flow.

In 2000, the Unit 1 steam generators were replaced. The Babcock & Wilcox (BWI) Model 51R replacement steam generators consist of a lower replacement steam generator subassembly (RSGSA), replacement steam drum internals, and replacement feedring fabricated by BWI plus the re-used existing Model 51R steam drum pressure boundary. The procurement of the replacement steam generator subassemblies did not affect the original design basis. Where appropriate, the tables and subsections of Chapter 4 have been revised to reflect the design enhancements of the Unit 1 RSG.

During the summer of 1988, Unit 2 steam generators were replaced. This entailed the procurement of new replacement lower assemblies and refurbishment of the upper assemblies and internals (steam dome). The procurement of the replacement steam generator assemblies did not affect the original design basis. Where appropriate, the tables and subsections of Chapter 4.0 have been revised to reflect the design enhancements of the replacement steam generator lower assemblies with their refurbished upper assemblies.

The Unit 1 and Unit 2 design pressure limit for primary-to-secondary pressure differential is 1600 psi. Certain operating conditions (e.g., low full-power vessel average temperature, high steam generator tube plugging levels, and reactor coolant system pressure controlled to 2250 psia) can result in the maximum primary-to-secondary pressure gradient to exceed the 1600 psi limit during normal transients. Calculations indicate that a minimum full-power steam pressure of 679 psia is necessary such that the maximum primary-to-secondary pressure gradient remains less than or equal to 1600 psi during normal transients for either Unit 1 or Unit 2. In order to provide additional conservatism relative to the design differential pressure limit, the minimum full-power steam pressure shall be restricted to 690 psia when reactor coolant system pressure is controlled to 2250 psia.

4.2.2.5 Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage mixed flow pump, which employs a controlled leakage seal assembly. The principal design parameters for the pumps are listed in Table 4.1-6. The reactor coolant pump estimated performance and NPSH characteristics are shown in Figure 4.2-8.

Reactor coolant is drawn up through the pump impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The rotor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pump in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, a second seal which directs the controlled leakage out of the pump, and a third seal which minimizes the leakage of water out of the pump. The second and third seals drain to the reactor coolant drain tank.

The shaft seal section consists of the number 1 controlled leakage, film riding face seal, a shutdown seal (SDS) assembly and the number 2 and number 3 rubbing face seals. The seals are contained within the main flange and seal housing. The SDS is housed within the number 1 seal area and is a passive device actuated by high seal flow temperature resulting from a loss of seal injection and component cooling water (CCW) cooling to the thermal barrier cooling coil.

In the event of a loss of seal injection and CCW flow to the thermal barrier heat exchanger, reactor coolant begins to travel along the RCP shaft and displaces the cooler seal injection water. Once the temperature within the number 1 seal reaches the actuation temperature range of the SDS, the SDS will activate to limit leakage from the RCS through the RCP seal package. The loss of reactor coolant through the RCP seal package is limited when the SDS polymer ring activates (clamps down) around the number 1 seal sleeve.

A portion of the high pressure water flow from the charging pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System around the thermal barrier cooling coil and through a labyrinth seal on the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount, which leaks through the second seal, is also collected and removed from the pump.

Component cooling water is supplied to the motor bearing oil coolers and the thermal barrier-cooling coil. Should the seal injection water flow be lost or interrupted, Reactor Coolant flows across the thermal barrier-cooling coil in the reverse direction and is cooled. It then becomes the source of water to the pump radial bearing and to the pump seals.

The squirrel cage induction motor driving the pump is air-to-water cooled, and has oil lubricated thrust and radial bearings. A water-lubricated bearing provides radial support for the pump shaft. An oil collection system is provided for each reactor coolant pump motor to minimize the fire potential from spillage. The fire protection/suppression system is described in Section 9.8.1.

A flywheel on the shaft above the motor provides additional inertia to extend flow coastdown. An inadvertent, early actuation of the SDS on the pump shaft, with the shaft still rotating, will not adversely impact RCP coastdown. Each pump contains a ratchet mechanism to prevent reverse rotation. The reactor coolant pump flywheel is shown in Figure 4.2-6.

Precautionary measures, taken to preclude missile formation from primary coolant pump components, assure that the pumps will not produce missiles under any anticipated accident condition.

Components of the reactor coolant pump motor have been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing.

The most adverse operating condition for the flywheel is visualized to be the loss-of-load situation. The following conservative design-operation conditions precluded missile production by the flywheel. The wheels were fabricated from rolled, vacuum-degassed, ASTM A-533 steel plates. Flywheel blanks were flame-cut from the plate, with allowance for exclusion of flame-affected metal. A minimum of six Charpy tests were made from each plate, three parallel and three normal to the rolling direction. The tests determined that each blank satisfied the design requirements. An NDTT less than +10°F is specified. The finished flywheels were subjected to 100% volumetric ultrasonic inspection. The finished machined bores were also subjected to magnetic particle, or liquid penetrant examination.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 4.2-7) less than 50% of the minimum specified material yield strength at room temperature (100 to 150°F).

Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (References 2, and 3), to be 3900 rpm, more than three times the operating speed.

A fracture mechanics evaluation was completed for the reactor coolant pump flywheel.

To estimate the magnitude of fatigue crack growth during plant life, an initial radial crack length of 10% of the distance through the flywheel (from the keyway to the flywheel outer radius) was conservatively assumed. The analysis assumed 6000 cycles of pump starts and stops. The existing analysis is valid for the period of extended operation associated with license renewal.

The reactor coolant pump motor bearings are of conventional design, the radial bearings are the segmented pad type, and the thrust bearings are tilting pad Kingsbury bearings. All are oil lubricated; the lower radial bearing and the thrust bearings are submerged in oil, and the upper radial bearing is oil fed from an impeller integral with the thrust runner. Low oil levels would signal an alarm in the control room. Each motor bearing contains embedded temperature detectors; therefore initiation of failure, separate from loss of oil, would be indicated and alarmed in the control room as high bearing temperature. This would alert the operator to take corrective action. Even if the bearing proceeded to failure, the low melting point Babbitt metal on the pad surfaces would ensure that no sudden seizure of the bearing would occur. In this event the motor would continue to drive, as it has sufficient reserve capacity to operate, even under such conditions. However, it would draw excessive currents and at some stage would shut down because of the high current.

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft would fail in torsion just below the coupling to the motor. This would constitute a loss of coolant flow in the one loop; the effect of which is analyzed in Chapter 14. Following the seizure, the motor would continue to run without any overspeed, and the flywheel would maintain its integrity, as it would still be supported on a shaft with two bearings.

There are no other credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing would be precluded by shearing of the graphitar in the bearing. Any seizure in the seals would result in a shearing of the anti-rotation pin in the seal ring. An inadvertent actuation of the shutdown seal on the shaft will not interrupt core cooling flow provided by the RCP. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions would be initially by high temperature signals from the bearing water temperature detector, excessive No. 1 seal leakoff indications, and off-scale #1 seal leakoff indications, respectively. Following these signals, pump vibration levels would be checked. These would show excessive levels, indicating some mechanical trouble.

The design specifications for the reactor coolant pumps include, as a design condition the stresses induced by a design basis earthquake. Beside evaluating the externally produced loads on the nozzles and support lugs, an analysis was made of the effect of gyroscopic reactions on the flywheel, the bearings and in the shaft, due to rotational movements of the pump about a horizontal axis, during the maximum seismic disturbance.

The pump would continue to run unaffected by such conditions. In no case does any bearing stress in the pump or motor exceed or even approach a value, which the bearing could not carry.

The design requirements of the bearings are primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over long periods of time. To this end, the surface bearing stresses are held at a very low value, and even under the most severe seismic transients or accidents, do not begin to approach loads which cannot be adequately carried for short periods of time.

Because there are no established criteria for short-time stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

As is generally the case with machines of this size, the shaft dimensions are predicated on avoidance of shaft critical speed conditions, rather than actual levels of stress.

There are many machines as large as, and larger than these, that are designed to run at speeds in excess of first shaft critical. However, it is considered more desirable to operate below first critical speed, and the reactor coolant pumps are designed in accordance with this philosophy. This results in a shaft design, which, even under the most severe postulated transient, gives very low values of actual stress. While it would be possible to present quantitative data of imposed operational stress relative to maximum tolerable levels, if the mode of postulated failure were clearly defined, such figures would have little significance in a meaningful assessment of the adequacy of the shaft to maintain its integrity under operational transients. However, a qualitative assessment of such factors gives assurance of the conservative stress levels experienced during these transients.

In each of these cases, where the functional requirements of the component control its dimensions, it can be seen that if these requirements are met, the stress-related failure cases are more than adequately satisfied.

It is thus considered to be beyond the bounds of reasonable credibility that any bearing or shaft failure could occur that would endanger the integrity of the pump flywheel.

4.2.2.6 Reactor Coolant System Vents

The reactor coolant system is provided with a reactor vessel head vent and a pressurizer steam space vent to remove non-condensible gas or steam from the system. The vents are designed to pass a combined capacity equal to one-half of the reactor coolant system volume in one hour at system design pressure and temperature. They are designed to mitigate a possible condition of inadequate core cooling, inadequate natural circulation or inability to depressurize the system to permit initiation of the residual heat removal system as a result of a condition causing the accumulation of non-condensible gas or steam in the reactor coolant system. The reactor vessel head vent and the pressurizer steam space vent are designed as seismic Class I with two parallel, one inch nominal pipe size flow paths. Each path contains redundant safety grade, fail-closed solenoid valves. Orifices (1/4" reactor head vent path, 3/8" pressurizer steam space vent path) are installed upstream of the solenoid operated isolation valves to limit the maximum postulated flow, in the case of a pipe break down stream of the orifices, to less than the capacity of one centrifugal charging pump. Sealed-open, hand operated valves are installed upstream of the orifices. The solenoid valves in one flow path are powered independently from the valves in the second flow path and each valve has a separate control switch. All are normally closed and have stem position indicators to provide remote indication of valve position.

Downstream of the solenoid valves, RTDs are installed to detect leakage and provide an alarm. For the reactor vessel head vent, one 1/4" orifice is installed downstream of the solenoid operated isolation valves to limit piping and support loads during venting. Both vents discharge to the upper volume of the containment in an area, which will provide adequate dilution of any combustible gas.

4.2.2.7 Reactor Coolant Piping

The reactor coolant piping and fittings, which make up the loops, are austenitic stainless steel. The reactor coolant piping is made by a centrifugal casting process. All smaller piping which comprises part of the Reactor Coolant System boundary, such as the pressurizer surge line, spray and relief line, loop drains, and connecting lines to other systems is also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. The joints and connections in piping which comprise part of the reactor coolant boundary are welded, except for the pressurizer safety valves, where flanged joints are used. Thermal sleeves are installed at points in the system where high thermal stresses could develop due to rapid changes in fluid temperature during normal operational transients. These points include:

1. Charging connections from the Chemical and Volume Control System.
2. Return lines from the Residual Heat Removal Loop (also part of the Emergency Core Cooling System).
3. Both ends of the pressurizer surge line.
4. Pressurizer spray line connection to the pressurizer.

Thermal sleeves are not provided for the remaining injection connections of the ECCS since these connections are not in normal use.

Piping connections from auxiliary systems are made above the horizontal centerline of the reactor coolant piping, with the exception of:

1. Residual heat removal pump suction, which is 45o down from the horizontal centerline. This enables the water level in the reactor coolant system to be lower in the reactor coolant pipe while continuing to operate the residual heat removal system, should this be required for maintenance.
2. Loop drain lines and the connection for temporary level measurement of water in the reactor coolant system during refueling and maintenance operation.
3. The differential pressure taps for flow measurement are downstream of the steam generators on the 90o elbow.

4. RVLIS piping connections are located at the horizontal centerline of Loops 1 and 3. A connection for the Mid-Loop instrument piping is located 60º down from the horizontal centerline of Loop 2.

Penetrations into the coolant flow path are limited to the following:

1. The spray line inlet connections extend into the cold-leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.
2. The reactor coolant sample system taps are inserted into the main stream to obtain a representative sample of the reactor coolant.
3. The wide range temperature detectors are located in RTD wells that extend into the reactor coolant pipes.
4. Three thermowell-mounted narrow-range RTDs extend into the hot leg scoops to provide a representative hot leg temperature.
5. A thermowell-mounted narrow-range RTD extends into each reactor coolant cold leg pipe.

Principal design data for the reactor coolant piping are given in Table 4.1-7.

Piping was restrained for postulated break conditions to prevent plastic hinge formation, except for certain breaks where no damage to Class I systems or the containment liner could result which would violate the criteria discussed in Section 4.2.4.

Numerous pipe whip restraints were designed for postulated ruptures occurring within the reactor coolant boundary to limit the consequences of the postulated ruptures. The pipe whip restraints were designed for circumferential ruptures in the reactor coolant system and connecting systems at changes in direction of the piping and nozzle junctions when consequential damage from these ruptures might occur. They were also provided to limit the consequences of longitudinal ruptures having a jet force equal to that of a circumferential rupture. Longitudinal splits were postulated to occur at selected points within the reactor coolant boundary.

4.2.2.8 Valves

All valves in the reactor coolant system which are in contact with the coolant are constructed primarily of stainless steel. Other materials in contact with the coolant are special materials such as hard surfacing and packing.

Hard surfacing performed on austenitic stainless steel pressure boundary parts is controlled to minimize severe sensitization of the stainless steel. Seat rings are utilized in valve design to preclude sensitization of the reactor coolant pressure boundary wall.

Indication of valve position for the pressurizer safety and power-operated relief valves is provided by a four channel acoustic flow monitor. There are four accelerometers; one strapped to the discharge of each of the three pressurizer safety valves and one on the common discharge of the three power relief valves. Flow through any of these valves produces an acoustic energy input to the respective accelerometer and this is amplified on the assigned channel of the monitor which is located in the control room. Indication on four vertical rows of light emitting diodes represents a bar graph display of relative flow through the monitored valves.

4.2.2.8.1 Pressurizer Safety Valves

The pressurizer safety valves are totally enclosed pop-type valves. The valves are spring-loaded, self-activated and with back-pressure compensation designed to prevent system pressure from exceeding the design pressure by more than 110 percent, in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The set pressure of the valves is 2485 psig.

The 6” pipes connecting the pressurizer nozzles to their respective safety valves are shaped in the form of a loop seal. Piping is connected to the bottom of each loop seal to drain any condensate that accumulates in the loop seal. An acoustic flow monitor and a temperature indicator on each valve discharge alerts the operator to the passage of steam due to leakage or valve lifting.

4.2.2.8.2 Power Relief Valves

The pressurizer is equipped with 3 power-operated relief valves, which limit system pressure for a large power mismatch and thus lessen the likelihood of an actuation of the fixed high-pressure reactor trip. The relief valves operate automatically or by remote manual control. The original design for 3 PORVs was to provide 100% load rejection capability. Since the load rejection capability has been reduced to 50%, the third PORV is now considered an installed spare. The 50% load rejection transient is an ANS Condition 1 event, also known as a Normal Operating Transient, or Plant Condition 1. ANSI/ANS Standard 51.1 does not consider the effects of single failure for Condition 1 events. The operation of these valves also limits the undesirable operation of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power-operated relief valves. An acoustic flow monitor and a temperature indicator on the common discharge of the relief valves alerts the operator to the passage of steam due to leakage or valve opening. Indication of valve position is also provided by limit switches on each valve.

During startup and shutdown, a manually energized safeguard circuit is in service while the reactor coolant system temperature is below 266°F for Unit 1 and 299°F for Unit 2. This allows automatic opening of that Unit's two power relief valves at ≤435 psig for low temperature overpressure protection (LTOP) of the reactor vessel. This safeguard circuit ensures that the reactor pressure remains below the ASME Section III, Appendix G “Protection Against Nonductile Failure” limits in the case of an LTOP event.

The PORVs are spring-loaded-closed, air required to open valves. Normally this air is supplied by the plant control air source. To assure operability of the valves upon a loss of control air, a backup air supply is provided for two of the PORVs. The backup air supply consists of compressed air bottles. The backup air supply contains sufficient air for the required number of PORV strokes in a ten minute period during an LTOP event.

Design parameters for the pressurizer spray control, safety, and power relief valves are given in Table 4.1-8.

4.2.2.9 Reactor Coolant System Supports

1. Steam Generator Support

   Each steam generator is supported by a structural system consisting of four vertical support columns and upper and lower lateral restraints approximately 46½ feet apart. The vertical columns have a ball joint connection at each end to accommodate both the radial growth of the steam generator itself and the radial movement of the vessel from the reactor center.

   The lower lateral support consists of an inner frame, keyed and shimmed to the four steam generator support feet to accommodate radial growth of these feet. The inner frame is surrounded by an outer frame, which is embedded in both the primary shield and crane wall concrete. The connection between the inner and outer frame consists of a series of shimmed points, which act as both guides and limit stops to allow for expansion from the center of the reactor. The lower lateral support restrains both torsional and translational movements.

   The upper lateral support consists of a ring band, which is shimmed to the steam generator at twelve locations around the circumference. Attached to this band are lugs 180o apart which are shimmed and guided to a structural framing system which is embedded in the crane wall and steam generator enclosure wall concrete. Hydraulic snubbers are also connected 180O apart on the band and tied to other embedded frames in a direction coincident with the direction of movement away from the reactor center. The upper lateral support restrains rapid translational movements in all horizontal directions.

2. Reactor Vessel Supports

   The reactor vessel is supported by four of its eight nozzles by four individual weldments embedded in the primary shield concrete. Each nozzle pad bears on a shoe that is supported by a heavy U-shaped weldment, which wraps around the shoe. The U-shaped weldment is water-cooled at the junction of the outer flange and the web by two continuous welded angles on either side of the web. The U-shaped weldment bears vertically on two shims and is restrained horizontally by a series of shims and bearing plates. These bearing plates and shims are connected to an outer weldment, which completely surrounds the U-shaped weldment and is embedded in the concrete.

The reactor support system allows the reactor to expand radially from its vertical centerline but resists rotational motion in all orthogonal planes. The nozzle horizontal centerlines translate in the vertical direction relative to the shoes.

3. Pressurizer Support

   The pressurizer is supported on a ring girder, which is in turn supported on a concrete slab. Horizontally, the vessel is restrained at two elevations approximately 27 feet apart.

   The lower restraint consists of anchor bolts in slightly oversize holes in the ring girder. The upper restraint consists of four individual weldments embedded in concrete that allow the pressurizer to expand radially, but resist torsional and translational horizontal movements.

4. Reactor Coolant Pump Support

   Each reactor coolant pump is supported vertically by three ball joint ended columns. This structural column system resists both overturning and vertical movement while allowing for expansion from the center of reactor. Excessive torsional and horizontal translational movements are resisted by a combination of lateral thrust columns anchored into the crane wall concrete.

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 by relief and safety valves connected to the top head of the pressurizer. The safety valves are currently analyzed for steam discharge only. The power operated relief valves are analyzed for steam or water discharge. However, evaluations have shown that the pressurizer will not become water solid before at least 10 minutes following a spurious Safety Injection or a feedline break. The relief and safety valves discharge into the pressurizer relief tank, which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 4.2-1A, and the valve design parameters are given in Table 4.1-8. The valves are further discussed in Sub-Section 4.2.2.8.

4.2.4 Protection against Proliferation of Dynamic Effects

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into certain areas of the containment building during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plate and all essential equipment inside the containment against blowdown jet forces and pipe whip to meet the missile protection criteria of Section 1.4.1 and the following:

1. A break of a steam or feedwater pipe inside the containment must not cause a break in a steam or feedwater pipe of another loop.

2. The leak tightness of the containment liner must not be damaged by a whip or blowdown jet force of a pipe which is part of the reactor coolant pressure boundary or which is necessary to function after a LOCA.

The concrete deck over the Reactor Coolant System also provides shielding and missile damage protection.

Reactor coolant pressure boundary equipment and piping are supported and provided with restraints to resist the actions of seismic, thermal expansion and pipe rupture effects.

4.2.5 Materials of Construction

The materials used in the Reactor Coolant System are selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

Reactor Coolant System materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steel and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 4.2-2. Reactor coolant chemistry is further discussed in Section 4.2.8.

The secondary side water chemistry is controlled to minimize corrosion and sludge buildup in the steam generators. Plant procedures list the limits for containments in the steam generators. The levels of these contaminates are normally maintained well below the limits.

The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, stress, and time.

It is a characteristic of stress corrosion that combinations of alloy and environment, which result in cracking, are usually quite specific. Environments, which have been shown to cause stress-corrosion cracking of stainless steels, are free alkalinity in the presence of chlorides, fluorides, and free oxygen. Experience has shown that 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 under very specific conditions, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses, which resulted from normal manufacturing procedures. The steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel alloy has excellent resistance to general and pitting-type corrosion in severe operating water conditions. Extensive operating experience with Inconel units has confirmed this conclusion.

External insulation of Reactor Coolant system components is compatible with component materials. The cylindrical shell exterior, closure flanges and bottom head of the reactor vessel are insulated with stainless steel, metallic, reflective insulation. The closure head is insulated with stainless steel, metallic, reflective insulation. Other external corrosion-resistant surfaces in the Reactor Coolant System are insulated with low or halide-free insulating material as required.

The remaining material in the reactor vessel, and other Reactor Coolant System components, meets the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility which ensures a low RT_{NDT} temperature, and thereby gives assurance that the finished vessels can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the reactor vessel are dependent upon the temperatures at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the RT_{NDT} temperature.

The effects and methods of calculating the cumulative fast neutron (E > 1 MeV) exposure of the vessel wall material is described in Section 4.5.

To evaluate the RT_{NDT} temperature shift of welds, heat affected zones and base material for the vessel; test coupons of these material types have been included in the reactor vessel surveillance program described in Sub-Section 4.5.1.3.

The methods used to measure the initial RT_{NDT} temperature of the reactor vessel base plate material are given in Sub-Section 4.5.1.3.

4.2.6 Maximum Heating and Cooling Rates

The Reactor Coolant System operating cycle used for design purposes is given in Table 4.1-10 and described in Section 4.1.5. The normal system heating and cooling rate is 60°F/hr. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level, of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant.

The fastest cooldown rates, which result from the hypothetical case of a break of a main steam line, are discussed in Chapter 14.

Surface thermocouples on each Steam Generator above the level of the tubesheet are provided to permit a direct measurement of Steam Generator temperature, to determine that no more than a 50°F difference exists with the Reactor Coolant System cold leg temperature prior to starting a reactor coolant pump in the inactive loop with the Reactor Coolant System in the water solid condition. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer, or if the SG/RCS Delta T is less than 50°F.

4.2.7 Leakage

The existence of leakage from the Reactor Coolant System to the lower containment compartment regardless of the source of leakage is detected by one or more of the following conditions:

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

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

c. An increase in the amount of coolant make-up water which is required to maintain normal level in the pressurizer, or an increase in containment sump level.

4.2.7.1 Leakage Prevention

Reactor Coolant System components are manufactured to exacting specifications which exceeds normal code requirements (as outlined in Section 4.1.6). In addition, because of the welded construction of the Reactor Coolant System and the extensive non-destructive testing to which it is subjected (as outlined in Sub-Chapter 4.5), it is considered that leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the Reactor Coolant System is permitted by the reactor coolant pump seals. Also all sealed joints are potential sources of leakage even though the most appropriate sealing device is selected in each case. Thus, because of the large number of joints and the difficulty of assuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable.

4.2.7.2 Locating Leaks

Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

Methods of leak location, which can be used during plant shutdown, include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the Reactor Coolant System in the leaking fluid and then left behind by the evaporation process. Portable sonic detectors sensitive to ultrasonic frequencies provide another means for locating small leaks.

4.2.7.3 Leak Detection Methods

a. Containment Air Particulate and Containment Radiogas Monitors

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

The sensitivity of the air particulate monitor to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal base line leakage into the containment. The sensitivity is greatest where base line leakage is low, as has been demonstrated by the experience of Indian Point Unit No. 1, Yankee Rowe, and Dresden Unit 1. Where containment air particulate activity is below the threshold of detectability, operation of the monitor with stationary filter paper would increase leak sensitivity to a few cubic centimeters per minute. Assuming a low background of containment air particulate radioactivity, a reactor coolant corrosion product radioactivity (Fe, Mn, Co, Cr) of 0.2 μCi/cc (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactive solids into the containment air, the air particulate monitor is capable of detecting leaks as small as approximately 0.0013 gal/min (5 cc/minute) within thirty minutes after they occur. If only ten percent of the particulate activity is actually dispersed in the air, the threshold of detectable leakage is raised to approximately 0.013 gpm (50 cc/minute).

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

The containment radiogas monitor is inherently less sensitive (threshold at

10-6μCi/cc) than the containment air particulate monitor, and would function only in the event that significant reactor coolant gaseous activity exists due to fuel cladding defects. Assuming a reactor coolant gas activity of 0.3 μCi/cc, the occurrence of a leak of two to four gpm would double the background (predominantly argon-41) in less than one hour. In these circumstances this instrument would be a useful backup to the air particulate monitor.

b. Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the containment. This instrumentation has not nearly the sensitivity of the air particulate monitor, but has the advantage of being sensitive to vapor originating from all sources, the reactor coolant, the steam, and the feedwater systems. Plots of containment air dew point variations above a base-line maximum should be sensitive to incremental leakage equivalent to 0.2 to 1.0 gpm.

c. Liquid Inventory in the Process Systems and in the Containment Sump

An increase in the amount of coolant make-up water, which is required to maintain normal level in the pressurizer, will be indicated by an increase in charging flow or change in volume control tank level. Further details of the operation of the charging system is supplied in Chapter 9.

Gross leakage will be indicated by a rise in normal containment sump level and periodic operation of containment sump pumps. A run time meter is provided to monitor the frequency of operation and running time of each containment sump pump.

4.2.8 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.

Reactor coolant water chemistry specifications are listed in Table 4.2-2. Periodic analysis of the coolant chemical composition are performed to verify that the reactor coolant water quality meets these specifications. Maintenance of the water quality to minimize chemical control is maintained with the Chemical and Volume Control System and Sampling System, which are described in Chapter 9.

4.2.9 Reactor Coolant Flow Measurements

Elbow taps are used in the Reactor Coolant System as an instrument device that indicates the status of the reactor coolant flow (Reference 4). The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \left(\frac{\omega}{\omega_o}\right)^2 ,$$

where:

ΔP_o is the referenced pressure differential with the corresponding referenced flow rate ω_o ΔP is the pressure differential with the corresponding flow rate ω.

The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within ± 10% and field results have shown the repeatability of the trip point to be within ± 1%. The analysis of the loss of flow transient presented in Sub-Chapter 14.1 assumes instrumentation error of ± 3%.

4.2.9.1 Reactor Coolant Margin To Saturation

A digital subcooling monitor is provided to display in the control room either the temperature or pressure margin available for the sub-cooled operating condition below the corresponding saturation pressure or saturation temperature.

The device selects the highest temperature reading from 8 core exit thermocouples and 8 hot and cold leg RTD's, and the lower pressure reading from two RVLIS reactor coolant wide range pressure sensors, and then calculates the corresponding saturation conditions, and displays the available margin of subcooling below saturation, in either temperature (°F) or pressure (psi).

The Plant Process Computer (PPC) may also be used to display the margin of subcooling temperature (°F) on a trend recorder in the control room. The computer uses a calculated saturation temperature derived from the lowest valid value of the RC System wide range pressure inputs, the atmospheric pressure constant and the steam tables in conjunction with one of the following:

1. Hottest of the valid Hot and Cold Leg Wide Range RCS Temperature Inputs (RTD's);
2. Hottest of the valid Incore Thermocouples;
3. Average of the valid Hot and Cold Leg Wide Range RCS Temperature Inputs (RTD's);
4. Average of the valid Incore Thermocouples.

4.2.10 Loose Parts Detection

A loose parts monitoring system is used to detect loose metallic parts impacting within the reactor coolant system. Metallic debris may appear as a result of outage work or wear of system internals. Carried through the system, such debris may damage internal primary system components. The loose parts monitoring system provides early detection of loose metallic parts to minimize damage. The system was designed to meet the intent of Regulatory Guide 1.133, Rev. 1.

The system consists of redundant accelerometers at the reactor vessel and steam generators. A metallic impact will result in minute accelerations in the reactor coolant system component material, which will be detected by the accelerometers. The system alarms when impacts above a previously established threshold occur. Bypassing of alarms based on plant conditions is controlled by plant procedures.

4.2.11 Reactor Vessel Water Level

A Reactor Vessel Level Instrumentation System (RVLIS) is provided to indicate the relative vessel water level or the relative void content of fluid in the vessel during post-accident conditions. This level indication assists the operator in recognizing conditions, which may lead to high temperatures that could damage the vessel or its internals. Level indicators and recorders are located in the control rooms.

Sensors measuring the differential pressure between the vessel head and the bottom and between the head and the hot legs provide the basis for level indication. Because flow through the vessel affects differential pressure measurement, three level indication ranges are provided by separate sensors. One range monitors water level from the vessel bottom to the head during full flow conditions in the reactor vessel. The remaining two ranges monitor the entire vessel level and partial water level (top reactor head to hot leg) at zero forced flow conditions (no reactor coolant pump operating).

The differential pressure measurements are compensated for process effects using reactor coolant system pressure and temperature measurements. They are also compensated for environmental temperature effects on the RVLIS sensing lines using temperature measurements at representative sensing line locations.

4.2.11.1 References for Section 4.2

1. “Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems WCAP 7735 (Westinghouse Class 3), July 1971.”
2. Ernest L. Robinson, “Bursting Tests of Steam-Turbine Disk Wheels”, Transactions of the A.S.M.E., July 1944.
3. “Application of the Griffith-Irwin Theory of Crack Propagation to the Bursting Behavior of Disks, Including Analytical and Experimental Studies”, by D. H. Winne and B. M. Wundt, ASME, December 1, 1957.
4. J. W. Murdock, “Performance Characteristic of Elbow Flowmeters”, Transactions of the ASME, September, l964.

4.3 SYSTEM DESIGN EVALUATION

4.3.1 Safety Factors

The safety of the reactor vessel and other Reactor Coolant System pressure containing components and piping is dependent on several major factors including, design and stress analysis, material selection and fabrication, quality control and operations control.

4.3.1.1 Reactor Vessel - (Unit 1)

1. The following components of the reactor pressure vessel were analyzed in detail through systematic analytical procedures.
   A. Control Rod Housings
   B. Closure Head Flange and Shell
   C. Main Closure Studs
   D. Inlet Nozzles (and Vessel Support Pads)
   E. Outlet Nozzles (and Vessel Support Pads)
   F. Vessel Wall Transition
   G. Core-Barrel Support Pads
   H. Bottom Head to Shell Juncture
   I. Bottom Head Instrument Penetrations, etc.

4.3.1.2 Method of Analysis - (Unit 1)

Item (A). An interaction analysis was performed on the CRDM housing. The flange was assumed to be a ring and the tube a long cylinder. The different values of Young's Modulus and coefficients of thermal expansion of the tubes were taken into account in the analysis. Local flexibility was considered at appropriate locations. The closure head was treated as a perforated spherical shell with modified elastic constants. The effects of redundants on the closure head were assumed to be local only. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation was made for the J weld.

Item (B). The closure head, closure head flange, vessel flange, vessel shell and closure studs were all evaluated in the same analysis. An analytical model was developed by dividing the actual structure into different elements such as sphere, ring, long cylinder and cantilever beam, etc. An interaction analysis was performed to determine the stresses due to mechanical and thermal loads. These stresses were evaluated in light of the strength and fatigue requirements of the ASME Boiler and Pressure Vessel Code Section III.

Item (C). A similar analysis was performed for the vessel flange to vessel shell juncture and for the main closure studs.

Items (D) & (E). For the analysis of nozzle and nozzle to shell juncture, the loads considered were internal pressure, operating transients, pipe reactions (thermal, gravity and seismic), the static weight of vessel, earthquake loading and expansion and contraction, etc. A combination of methods was used to evaluate the stresses due to mechanical, thermal and external loads resulting from seismic pipe reactions, earthquake and pipe break, etc. For fatigue evaluation, peak stresses resulting from external loads and thermal transients are determined by concentrating the stresses as calculated by the above described methods. Combining these stresses enabled the fatigue evaluation to be performed.

Item (F). The vessel wall transition was analyzed by means of a standard interaction analysis. The thermal stresses are determined by the skin method, which assumed that the inside surface of the vessel is at the same temperature as the reactor coolant and the mean temperature of the shell remains at the steady state temperature. This method is considered conservative.

Item (G). The thermal, mechanical and pressure stresses were calculated at various locations on the core barrel support pad and at the vessel wall. Mechanical stresses were calculated by the flexure formula for bending stress in a beam, pressure stresses were taken from the analysis of the vessel to bottom head juncture and thermal stresses were determined by the conservative method of skin stresses. The stresses due to the cyclic loads were multiplied by a stress concentration factor, where applicable, and used in the fatigue evaluation.

Item (H). The standard interaction analysis and skin stress methods were employed to evaluate the stresses due to mechanical and thermal stresses respectively. The fatigue evaluation was made on a cumulative basis where superposition of all transients was taken into consideration.

Item (I). An interaction analysis was performed by dividing the actual structure into an analytical model composed of different structural elements. The effects of the redundants on the bottom head were assumed to be local only. It was also assumed that for any condition where there was interference between the tube and the head, no bending at the weld can exist. Using the mechanical and thermal stresses from this analysis a fatigue evaluation was made for the J weld.

4.3.1.3 Reactor Vessel - (Unit 2)

1. The following components of reactor pressure vessel were analyzed in detail through systematic analytical procedures.
   - A. Control Rod Drive Mechanism (CRDM) Penetrations
   - B. Head and Shell Flanges and Closure Studs
   - C. Reactor Coolant Nozzle (and Vessel Support Pad) Outlets
   - D. Reactor Coolant Nozzle (and Vessel Support Pad) Inlets
   - E. Main Shell Thickness Transitions
   - F. Core Barrel Support Pads
   - G. Instrumentation Tubes

4.3.1.4 Method of Analysis - (Unit 2)

Item (A). Two models were made for the stress analysis of the CRDM tubes of the top head. A three-dimensional finite element model was used to analyze the J-groove weld region, while a two-dimensional model was used to analyze the bi-metallic weld region.

The analysis includes the bimetallic weld between the tube flange and the tube and the partial penetration weld between the tube and the top head. In the models, the vessel head extends in the meridional direction to a point halfway between the tube and its nearest neighbor.

For temperature calculations, three-dimensional heat transfer was assumed. A finite element program for axisymmetric bodies was used to compute the stresses of the model. Another computer program was utilized for the fatigue analysis. This program is coded to perform the fatigue analysis in accordance with the ASME Code Section III.

Item (B). For the thermal analysis, a finite element method was used in calculating the thermal profile. The model used was three-dimensional. The circumferential boundaries were chosen to represent symmetry of the studs and near symmetry of the perforated region.

The stress analysis was performed using a three-dimensional finite element method of analysis.

A complete fatigue analysis was performed using all operating transients including the effects of thermal and pressure loading. Usage factors were calculated using linear damage criteria.

Items (C) & (D). The outlet and inlet reactor coolant nozzles were analyzed as follows:

A complete thermal analysis was made for each operating transient. Temperature distributions were calculated by finite differences, using an axisymmetric model.

At critical times during the transients the thermal stresses, as well as stresses due to internal pressure, were calculated using an axisymmetric finite-element model. The finite element method was selected so that secondary and peak stresses associated with material and thickness changes could be studied.

The stresses produced by mechanical loads were calculated using a finite-element model, which included the effects of circumferential variations in the loads. The mechanical loads, consisting of seismic loads, dead weight loads, and thermal expansion loads were applied to the safe ends, the vessel support pads and the internal projection of the nozzle.

The calculated stresses were evaluated using the criteria of Section III of the ASME Code. As part of this evaluation a complete fatigue analysis was performed at selected points on the nozzle surface, and fatigue usage factors were calculated.

Item (E). Both the cylinder to bottom head transition and the transition between the 8½” and 10½” wall thickness were analyzed.

Temperature distributions at the shell transitions were calculated by a finite difference technique using an axisymmetric model.

Primary and secondary stresses due to pressure and thermal gradients were calculated using the KALNINS thin-shell program. The stresses were evaluated by the criteria of Section III of the ASME Code and a fatigue analysis was performed.

Item (F). The core support pads were analyzed as follows:

Using a plane two dimensional model, temperature distributions in the pad were calculated. These calculations were made using a finite difference technique. Thermal stresses were calculated using a plane finite-element model.

Stresses in the support pad, due to mechanical loads on the support pads, were calculated using formulas for the bending and torsion of rectangular beams. The stresses in the shell due to these loads were found using the results of Bijlaard as described in WRC Bulletin 107.

Fatigue usage factors were calculated, taking into account stress concentration effects.

Item (G). The stress analysis of the CRDM Penetration indicated that the outermost tube is much more critical than the center one. Therefore, for the investigation of the instrumentation tubes only the outermost model was made. The analysis of this model follows the same method as described for the CRDM Penetrations.

4.3.1.5 General

The location and geometry of the areas of discontinuity and/or stress concentration are shown in Figures 4.3-1, 4.3-2, and 4.3-3. The identification letters refer to Unit 1 as described above.

For the original design and rerating program, a summary of the estimated primary plus secondary stress intensity for components of the Unit 1 reactor vessel and the estimated cumulative fatigue usage factors for the components of the Unit 1 reactor vessel is given in Tables 4.3-1 and 4.3-2.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from nuclear power plants in service at the time of the evaluation, such as Yankee-Rowe.

The conservatism of the design fatigue curves used in the fatigue analysis has been demonstrated by the Pressure Vessel Research Committee (PVRC) in a series of cyclic pressurization tests of model vessels fabricated according to the Code. The results of the PVRC tests showed that no crack initiation was detected at any stress level below the code allowable fatigue curve and that no crack progressed through a vessel wall in less than three times the allowable number of cycles. Similarly, fatigue tests have been performed on irradiated pressure vessel steels with comparable results (Reference 1).

The vessel design pressure is 2485 psig while the normal operating pressure is 2235 psig. The resulting operating membrane stress is therefore amply below the code allowable membrane stress to account for operating pressure transients.

The allowable pressure-temperature relationship as a function of rate of temperature change was determined according to the method given in Appendix G2000 of the ASME Code, Section III.

The actual shift in RT_{NDT} temperature will be established periodically during plant operation by testing of vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for any increase in the RT_{NDT} temperature caused by irradiation, the limits given in the plant operating manual on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. For further details see Sub-Chapter 4.4.

The vessel closure contains fifty-four 7-inch diameter studs. The stud material is SA-540, which has a minimum yield strength of 104,400 psi at design temperature. The membrane stress in the studs when they are at the steady-state operational condition is less than half this value. This means that about half of the fifty-four studs have the capability of withstanding the hydrostatic end load on vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

As part of the plant operator training program, operating personnel receive instruction in reactor vessel design, fabrication and testing, as well as precautions necessary for pressure testing and operating modes. The need for record keeping is stressed; such records being helpful for future summation of time at power level and temperature which tend to influence the properties of the irradiated material in the core region.

4.3.1.6 Piping

The analysis of the Reactor Coolant Loop/Supports System was based on an integrated analytical model, which included the effects of the supports and the supported equipment.

A three-dimensional, multi-mass, elastic-dynamic model was constructed to represent the Reactor Coolant Loop/Supports System. The seismic spectrum at the concrete to support interface, obtained from an elastic-dynamic model of the reactor containment internal structure, was used as input to the piping analysis.

The dynamic analysis employed displacement method, lumped parameter, stiffness matrix formulations and assumptions that all components behave in a linear elastic manner. The proprietary computer code WESTDYN was used in this analysis.

The evaluation of damage propagation from pipe whip and blowdown jet forces was based upon piping configuration, location of barriers and supports, locations of postulated breaks, separation of redundant parts of the system and location of other systems and equipment in relation to the system under consideration. Confirmatory dynamic analyses of the Reactor Coolant Loop were performed for postulated circumferential as well as longitudinal ruptures. A total of eight guillotine and longitudinal ruptures were postulated and their consequences evaluated. The locations and types of rupture were chosen to realistically encompass the most severe loading on the component supports.

Subsequent to the initial design considerations discussed in the previous paragraph, the allowances of Generic Letter 84-04, “Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops” were adopted. This Leak-Before-Break (LBB) methodology eliminates the design requirement to consider the dynamic effects of postulated main coolant loop ruptures. The LBB methodology eliminates consideration of pipe whip effects, blowdown jet forces, and main coolant loop and reactor vessel support loads, for the previously postulated ruptures, on attached and adjacent SSCs.

Cook Nuclear Unit 1 has performed Mechanical Stress Improvement Process, MSIP@, on the D. C. Cook Unit 1 RPV inlet and outlet nozzles. The MSIP mitigates the potential formation of cracks from PWSCC by placing compressive residual stresses on the inside surface of the weld region by compressing the pipe through controlled plastic deformation. There will be no leak when there is no through-wall flaw, and it is beneficial for the LBB defense. However, LBB evaluation is performed with postulated through-wall flaws. Structural evaluations show that all LBB margins are satisfied. Margins remain valid for other critical locations after MSIP application. Also, as part of this evaluation, Flaw Crack Growth was performed, and the results show that the crack will not grow through the pipe wall.

4.3.1.7 Normal Operating Loads

System design operating parameters were used as the basis for the analysis of equipment, coolant piping and equipment support structures for normal operating loads. The analysis was performed using a static model to predict deformation and stresses in the system under normal operating conditions. The analysis with respect to the piping and vessels was in accordance with the provisions of USAS B31.1 and ASME Section III, respectively. Results of the analysis gave six generalized action components, three bending moments and three forces. These moments and forces were resolved into stresses in the piping in accordance with applicable codes. Stresses in the structural supports were determined by the material and section properties assuming linear elastic small deformation theory.

4.3.1.8 Seismic Loads

Analysis for seismic loads was based on a dynamic modal analysis. The appropriate floor spectral accelerations were used as input forcing functions to the detailed dynamic model. The loads developed from the dynamic model were incorporated into a detailed support model to determine the support member stresses.

4.3.1.9 Blowdown Loads

Analysis of blowdown loads resulting from loss-of-coolant accidents were based on the time-history responses of simultaneously applied blowdown forcing functions on a single broken loop dynamic model. The forcing functions were defined at points in the system loop where changes in cross section or direction of flow occur such that differential loads are generated during the blowdown transient. The loads developed from the dynamic models were incorporated into detailed support models to determine equipment support member stresses.

4.3.1.10 Combined Blowdown and Seismic Loads

The stresses in components resulting from normal loads and the worst case blowdown analysis were combined with the worst case seismic analysis to determine the maximum stress for the combined loading case. This is considered a very conservative method since it is highly improbable that both maxima will occur at the same instant. These stresses were combined to determine that the Reactor Coolant Loop/Supports System does not lose its intended functions for this highly improbable situation. The limiting stress criteria used in the analysis are defined in Sub-Chapter 2.9.

Under a postulated main steam pipe break, the resulting stresses in unbroken piping attached to the steam generator were calculated to be quite low and are well within the limiting criteria for the Faulted Condition. See Sub-section 5.2.2.7 for details of the structural design requirements for postulated pipe breaks.

4.3.1.11 Steam Generators

4.3.1.11.1 Unit 1

For Donald C. Cook Unit 1, the Babcock & Wilcox Model 51R replacement steam generators consist of a lower replacement steam generator subassembly (RSGSA), replacement steam drum internals, and replacement feedring fabricated by BWI plus the re-used existing steam drum pressure boundary.

Structural and seismic evaluations of Model 51R primary and secondary side pressure boundaries demonstrate that these components satisfy ASME III, Division 1, Class 1 design requirements for Service Levels A, B, C, and D (Normal, Upset, Emergency and Faulted conditions, respectively). Model 51R internal components are required to withstand all specified loadings to maintain heat transfer capability during and following a design basis (Safe Shutdown) earthquake. The structural design basis for steam generator internals is that the limiting primary side LOCA or a secondary side pipe rupture combined with a simultaneous SSE do not result in a simultaneous release of primary and secondary coolant or loss of the heat transfer capability of the Model 51R. Internals loading during these postulated events do not result in rupture, collapse, or deformation of the Model 51R tubes; rupture of the secondary side shell; or pull the tubes from the tubesheet. In the case of a primary pressure loss accident, the primary-secondary design pressure differential can reach approximately 1100 psig. This pressure differential is less than the primary-secondary design pressure differential of 1600 psi for normal operating conditions. Thus, safe shutdown capability is maintained. The seismic evaluation of the Unit 1 Model 51R internals and the Unit 1 Model 51R structural evaluation is documented in the steam generator stress reports. For the re-used existing steam drum pressure boundary, the Level A and B fatigue analysis includes consideration of accumulated transient cycles up to the time of steam generator replacement plus a 40 year design life after steam generator replacement.

Classical methods and finite element modeling (where required for pressure and thermal transients) are employed to prove that the components examined meet the ASME Code allowable stresses. For seismic loading, the loads were first established by seismic analysis using the specified response spectrum and then these loads on components were used in the subsequent stress analysis. The design and hydrotest primary stresses in the Model 51R meet the design and hydrotest allowables of ASME III as shown in the following sections.

The requirements of Subsection NB-3221 for design stresses are met as follows:

$P_m \leq S_m$ at design temperature

$P_l \leq 1.5\ S_m$ at design temperature

$P_l + P_b \leq 1.5\ S_m$ at design temperature

Where:

P_m = General primary membrane stress

P_l = Local primary membrane stress

P_b = Primary bending stress

S_m = Design stress intensity value

S_y = Yield strength

S_u = Ultimate strength

The criteria for normal and upset loads are the ASME Levels A & B allowables for the range of primary plus secondary stress. The requirements of Subsection NB-3222 and NB-3223 are met as follows:

Range of $(P_m + P_b + Q) \leq 3\ S_m$ at operating temperature

Where:

Q = Secondary Stress

For pressure boundary components and tubing, it is also shown that the cumulative fatigue usage factor remains below 1.0 for all Service Level A, B, and test condition operating cycles.

The criterion for Service Level C loading conditions is to maintain integrity of tube, tube supports (lattice grid) and steam drum internals for emergency conditions of Level C. The requirements of Subsection NB-3224 are met as follows:

$P_m \leq$ greater of $(1.2\ S_m$ or $S_y)$ at operating temperature

$P_L + P_b \leq$ greater of $(1.8\ S_m$ or $1.5\ S_u)$ at operating temperature

The criterion for Service Level D loading condition (combined main steam line break and design basis earthquake) is to ensure tube integrity by proving that tube rupture and leakage cannot occur. The requirements of Subsection NB-3225 are met as follows:

$P_m \leq$ lesser of $(2.4\ S_m$ or $0.7\ S_u)$ at operating temperature

$P_L + P_b \leq$ lesser of $(3.6\ S_m$ or $1.05\ S_u)$ at operating temperature

The requirements of Subsection NB-3226 for hydrotest are met as follows:

for $P_m \leq 0.67\ S_y$ at test temperature

$P_m + P_b \leq 1.35\ S_y$ at test temperature

for $0.67\ S_y < P_m \leq 0.9\ S_y$ at test temperature

$P_m + P_b \leq (2.15\ S_y - 1.2\ P_m)$ at test temperature

4.3.1.11.2 Unit 2

Calculations confirm that the steam generator tube sheet will withstand the loading (which is quasi-static rather than a shock loading) caused by loss of reactor coolant. The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 44,800 psi. This is well below the ASME Section III material yield strength of 56,600 psi at 650°F. Because the pressure in the primary channel head would drop to zero under the condition postulated, no damage will result to the channel head.

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2235 psig across the tubes and tube sheet from the primary side or a maximum pressure differential of 805 psi across the tubes and tube sheet from the secondary side. Under these conditions there is no rupture of the primary to secondary boundary, including tubes and tube sheet. This criterion prevents any violation of the containment boundary.

To meet this criterion, it has been established that under the postulated accident conditions, where a primary to secondary side differential pressure of 2235 psig exists, the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, does not exceed 70 percent of material tensile stress at the operating temperature.

Also, the computed primary membrane plus primary bending stress in the tube sheet ligament, averaged across the ligament width at the tube sheet surface location giving maximum stress, do not exceed 105 percent of material tensile stress at the operating temperature.

This criterion is considered applicable to faulted (Level D) circumstances in that it is consistent with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plants Components Division, Appendix F. The stresses in the actual tube sheet design are obtained by using the applicable stress criteria.

A complete tube sheet analysis was performed to verify the structural integrity of the primary-secondary boundary under blowdown plus seismic conditions.

The tubes have been designed to the requirements (including stress limitations) of Section III for normal operation, assuming 1600 psig as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

Tube material was selected for both units based on industry experience and requirements of Section III. Inconel tubing is used in both units due to its strength and resistance to corrosion.

In the case of a primary pressure loss accident, the secondary-primary pressure differential can reach 805 psig. This pressure differential is less than the primary-secondary design pressure differential of 1600 psi. No stresses in excess of those covered in the ASME Section III rules for design conditions are experienced on the tube sheet for this accident case. An analysis of the U-bend portion under external pressures forms the basis for a calculated collapse pressure of 1500 psi providing the tubes a factor of safety of 1.86 against collapse. Analysis of straight tubes of the same size indicate actual tube strengths are significantly higher than the above. A collapse pressure of 2415 psi was calculated for a straight tube.

In addition, consideration has been given to the superimposed effects of secondary side pressure loss and the design basis earthquake loading. Furthermore, the fluid dynamic forces on the internal components affecting the primary-secondary boundary (tubes) have been assessed as well. For this condition, the criterion is that no rupture of the primary to secondary boundary (tubes and tube sheet) occurs.

In the most severe case, the fluid dynamic forces on the internals under secondary steam break accident conditions indicate that the tubes are adequate to constrain the motion of the baffle plates, with some plastic deformation, while boundary integrity is maintained.

The ratio of the allowable stresses on various components (based on an allowable membrane stress of 0.7 of the nominal tensile stress of the material) to the computed stresses for postulated faulted conditions have also been calculated and documented in applicable steam generator stress reports.

The evaluation of Westinghouse steam generator tube sheets was performed according to rules of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, Article 4 - Design. The design criteria included consideration of steady-state and transient operation as specified in the design specification. Due to the complex nature of the tube-tube sheet-shell-head structure, the analysis of the tube sheet required the application of the results of related research programs (such as the design data on perforated plates resulting from Pressure Vessel Research Committee (PVRC) programs) and the utilization of current techniques in computer analysis, the application of which was verified by comparing analytical and experimental results for related equipment.

The introductory paragraph I-900 of the ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels, states that consideration may be given to the stiffening and staying effects of tubes in perforated plates, if applicable. Furthermore, it is noted that the stress analysis methods of Appendix I of Section III are accepted techniques for obtaining solutions to problems for which these procedures are applicable. It allows use of other valid analytical or experimental techniques, where necessary.

Although the Nuclear Pressure Vessel Code Article I-9 provides rules and techniques in analysis of perforated plates, it should be noted that the stress intensity levels for perforated plate are given for triangular perforation arrays. Westinghouse tube sheets contain square arrays. Hence, Westinghouse utilized its own data and that obtained from PVRC research in square array perforation patterns for development of similar charts for stress intensity factors and elastic constants. The resulting stress intensity levels and fatigue stress ranges were evaluated according to the stress limitation of the Code.

The Westinghouse analysis of the steam generator tube sheets was included as part of the Stress Report requirement for Class A Nuclear Pressure vessels. The evaluation was based on the stress and fatigue limitations outlined in Article 4 Design of ASME Section III. The stress analysis techniques utilized in evaluation of the tube sheet complex included all factors considered appropriate to conservative determination of the stress levels. The analysis of the tube sheet complex includes the effect of all appurtenances attached to the perforated region of the tube sheet considered appropriate to conservative analysis of stress for evaluation on the basis of ASME Section III stress limitations. The evaluation involved the heat conduction and stress analysis of the tube sheet, channel head, and secondary shell structure for particular steady-state design conditions for which Code stress limitations are to be satisfied and for discrete points during transient operation for which the temperature/pressure conditions must be known to evaluate stress maxima and minima for fatigue life usage. In addition, limit analyses were performed to determine tube sheet capability to sustain emergency operating conditions for which elastic analysis does not suffice. The analytic techniques utilized were computerized and significant stress problems were verified experimentally to justify the techniques where possible.

Generally, the analytic treatment of the tube-tube sheet complex included a determination of elastic equivalent plate stress within the perforated region by an interaction analysis utilizing effective elastic constants appropriate to the nature of the perforation array.

For the steam generators, the fatigue analysis of the complex was performed at potentially critical regions in the complex such as the junction between tube sheet and channel head or secondary shell as well as at many locations throughout the perforated region of the tube sheet. For the holes for which fatigue evaluation was done, several points around the hole periphery were considered to assure that the maximum stress excursion has been considered. The fatigue evaluation was computerized to include stress maxima-minima excursions considered on an intra-transient basis.

The evaluation of the tube-to-tube sheet juncture of Westinghouse PWR System steam generators is based on a stress analysis of the interaction between tube and tube sheet hole for the significant thermal and pressure transients that are applied to the steam generator in its predicted histogram of cyclic operation. The evaluation is based on the numerical limits specified in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

Of importance in the analysis of the interaction system was the behavior of the tube hole, where it was recognized that the hole behavior is a function of the behavior of the entire tube sheet complex with attached head and shell. Hence, the output of the tube sheet analysis giving equivalent plate stresses in the perforated region was utilized in determining the free boundary displacements of the perforation to which the tube is attached.

Analysis of the juncture for the fillet-type weld utilized in the Westinghouse steam generator design was made with consideration of the effect of the rolled-in joint in the weld region, as well as with the conservative assumption that the tube flexure relative to the perforation is not inhibited with the rolled-in effect. Analysis of the juncture for the flush tube weld utilized in the Westinghouse steam generator design for Cook Nuclear Plant Unit 2 was made with the conservative assumption that the tube flexure relative to the perforation is not aided by contact with the tube plate.

A wide range of computational tools were utilized in these solutions including finite element, heat conduction and thin shell computer programs.

In all cases evaluated, the Westinghouse steam generator tube sheet complex meets the stress limitations and fatigue criteria specified in Article 4 of ASME Section III as well as faulted condition limitations specified in the equipment design specifications.

In this way, the tube-tube sheet integrity of the Westinghouse steam generator was demonstrated under the most adverse conceivable conditions resulting from a major breach in either the primary or secondary system piping.

The results of the tube sheet complex are listed in the appropriate steam generator stress report.

4.3.1.12 Pressurizer

The Pressurizer was analyzed for fatigue conditions in accordance with Section III of the ASME Boiler and Pressure Vessel Code using the thermal and pressure transient conditions listed elsewhere in this chapter.

The pressurizer vessel was analyzed for the following:

1. Normal operational loadings which include:
    a. Weight of water based on the vessel filled with cold water and including insulation.
    b. Normal loadings exerted by connecting piping.
2. Seismic loadings which include:
    a. The Operational Basis Earthquake (OBE). The pressurizer vessel was designed to resist earthquake loadings simultaneously in horizontal and vertical directions and to transmit such loadings through the vessel supports to the foundation. The OBE results in mechanical loadings, which in combination with the normal operational loads is considered to be an upset condition. The components of loadings exerted by the external piping due to the OBE were included in this evaluation.
    b. The Design Basis Earthquake (DBE). Pressurizer vessel integrity is not impaired to prevent a safe and orderly shutdown of the reactor plant when the DBE loadings, both horizontal and vertical acting simultaneously, are imposed on the vessel. These loadings and the centers of gravity involved are determined on the basis of the vessel at normal operating pressure,

temperature, and water level. The components of loadings exerted by the external piping due to the DBE were included in this evaluation.

3. Pipe break loadings in combination with the normal operational loads. The moment and forces were considered as acting in combination with each force separately. The pipe break accident was considered to be a faulted condition with the exception that the stress intensity limits were those specified under the Design Basis Earthquake condition.
4. Normal operating loads plus the DBE loads plus the pipe break loads. The resulting stress intensities do not exceed the stress intensity limits of Paragraph N417.11 (faulted conditions) in Section III of the Code with the following exception. The combination of all primary stress intensities in the vessel supports are within the support material yield strength specified in the above code.

4.3.1.13 Reactor Coolant Pump

All the pressure-containing parts of the reactor coolant pump were analyzed in accordance with Article 4 of the ASME Boiler and Pressure Vessel Code, Section III. This includes the casing, the main flange and the main flange bolts. The analysis included pressure, thermal and cyclic stresses, which were compared with the allowable stresses in the code.

Mathematical models of the parts were prepared and used in the analysis, which proceeded in two phases.

1. In the first phase, the design was checked against the design criteria of the ASME Code, with pressure stress calculations, although thermal effects are included implicitly with the experience factors. By this procedure, the shells were profiled to attain optimum metal distribution, with stress levels adequate to meet the more limiting requirements of the second phase.
2. In the second phase, the interactivity forces needed to maintain geometric capability between the various components are determined at design pressure and temperature, and applied to the components along with the external loads to determine the final stress state of the components. These are finally compared with the Code allowable values.

There are no other sections of the Code which are specified as areas of compliance, but where Code methods, allowable stresses, fabrication methods, etc., are applicable to a particular component, these were used to give a rigorous analysis and a conservative design.

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 Safety Injection and Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators, the main steam, feedwater, and condensate systems for decay heat removal for normal operating conditions down to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit plant cooldown following a loss of all reactor coolant pumps.

The Steam and Power Conversion System is described in Chapter 10. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The Auxiliary Feedwater System will supply water to the steam generators in the event that the main feedwater pumps are inoperative.

The Safety Injection System is described in Chapter 6. The Residual Heat Removal System is described in Chapter 9.

4.3.3 System Integrity

As part of the quality control on materials, Charpy V-notch toughness tests were run on the ferritic material used in fabricating pressure-retaining parts of the reactor vessel, steam generator and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. In addition, drop-weight tests and Charpy V-notch transition temperature tests were performed on the reactor vessel materials.

As an assurance of system integrity, components in the system are hydrotested at 3107 psig prior to initial operation. (3106 psig for Unit 1 RSGs)

4.3.4 Pressure Relief

The Reactor Coolant System is protected against overpressure by safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10 percent, in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The capacity of the pressurizer safety valves was determined from considerations of: (1) the reactor protective system, and (2) accident or transient conditions which may potentially cause overpressure.

The combined capacity of the pressurizer safety valves is equal to or greater than the maximum surge rate resulting from a complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant were assumed to open when the steam pressure reaches the secondary plant safety valve settings.

The design and installation criteria for mounting of the pressure-relieving devices (pressurizer relief and safety valves) are as follows:

1. Piping is able to withstand normal plus DBE loads and normal transient operating conditions.
2. Piping is to be able to withstand the worst condition arising from the operational mode of multiple discharges acting simultaneously.
3. Stresses in piping and its supports are to be held within the limits presented in Sub-Chapter 2.9.

4.3.5 System Incident Potential

Analysis of system incidents are discussed in Chapter 14.

4.3.6 References for Section 4.3

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18. WCAP-12135, Volume 1, "Donald C. Cook Nuclear Plant Units 1 and 2 Rerating Engineering Report," September 1989.
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4.4 SAFETY LIMITS AND CONDITIONS

4.4.1 System Heatup and Cooldown Rates

Operating limits for the Reactor Coolant System with respect to heatup and cooldown rates are defined in the Technical Specifications.

Assurance of adequate fracture toughness of the reactor coolant system is provided by compliance with the requirements for fracture toughness testing included in Section III of the ASME Boiler and Pressure Vessel Code and the Code of Federal Regulations, 10 CFR 50, Appendices G and H.

The original heatup and cooldown curves for the plant were based on the actual measured fracture toughness properties of the vessel materials determined in accordance with the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, as implemented by the Code Case #1514. Allowable pressures as a function of the rate of temperature change and the actual temperature relative to the nilductility transition reference temperature (RT_{NDT}) of the reactor vessel were established according to the methods given in Appendix G 2000, “Protection Against Non-Brittle Failure”, of Section III of the ASME Boiler and Pressure Vessel Code.

The original curves are based on a temperature scale relative to the RT_{NDT} of the vessel, including appropriate estimates of RT_{NDT} caused by radiation. Predicted ΔRT_{NDT} values were derived by using a curve depicting RT_{NDT} vs Fluence ($N/cm^2 > 1$ MeV) for various copper contents of vessel weld material, and the fluence of 1/4T corresponding to the maximum for the service period applicable. Initial RT_{NDT} included an assumed ΔRT_{NDT} corresponding to that predicted after 2 integrated full power years of operation.

The heatup and cooldown curves are updated based on information obtained from our radiation surveillance program in accordance with the Technical Specifications.

The use of RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material automatically provides additional conservatism for other vessel regions which are exposed to much lower radiation levels. Therefore, the flanges, nozzles, and other such regions less affected by radiation are favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}. Changes in fracture toughness of the core region plates or forgings, weldments and associated heat affected zones due to radiation damages are monitored by a surveillance program which conforms with ASTM E-185, “Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels”. The evaluation of the radiation damage in this surveillance program is based on pre-irradiation and post-irradiation testing by Charpy V-notch and tensile specimens and post-irradiation testing of wedge opening loading specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals.

4.4.2 Reactor Vessel, Pressurized Thermal Shock

As required by 10 CFR 50.61, the projected values of Reference Temperature, Pressurized Thermal Shock (RT_{PTS}) have been calculated for all of the Unit No. 1 and 2 reactor vessel beltline region materials for the fluence received to date and for the projected fluence levels received up to the expiration date of the operating licenses. The calculations revealed that none of the materials in either unit will exceed the RT_{PTS} screening criterion of 10 CFR 50.61 prior to the expiration of the operating licenses.

The RT_{PTS} assessment will be updated whenever core loadings, surveillance measurements, or other information (i.e. extension of operating license) indicates a significant change in the projected values. As such, all future core loading design changes and future surveillance capsule dosimetry data will be evaluated for possible effects on the RT_{PTS} assessment.

4.4.3 Reactor Coolant Activity Limits

Release of activity into the reactor coolant in itself does not constitute a hazard to the public. Activity in the coolant could constitute a hazard to the public only if the Reactor Coolant System barrier is breached, and then only if the coolant contains excessive amounts of activity which could be released to the environment. The plant systems are designed for operation with activity in the Reactor Coolant System corresponding to 1 percent fuel defects. In the event of steam generator tube leakage, high activity level at the condenser air ejector exhaust will initiate an alarm to warn the operator to take corrective action. The Reactor Coolant System activity limit during operation is defined in the Technical Specifications.

4.4.4 Maximum Pressure

The Reactor Coolant System serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. Thus, the safety limit of 2735 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. Reactor Coolant System pressure settings are given in Table 4.1-2.

4.4.5 System Minimum Operating Conditions

Minimum Operating Conditions for the Reactor Coolant System for all phases of operation are given in the Technical Specifications.

4.5 TESTS AND INSPECTIONS

4.5.1 Reactor Coolant System Inspection

4.5.1.1 Non-Destructive Inspection of Material and Components

Table 4.5-1 summarizes the quality control program for Reactor Coolant System components. In this table, the non-destructive tests and inspections which are required by Westinghouse specifications on Reactor Coolant System components and materials are specified for each component. The tests required by the applicable codes are included in this table. Westinghouse requirements, which are more stringent in some areas than those requirements specified in the applicable codes, are also included. The fabrication and quality control techniques used in the fabrication of the Reactor Coolant System were equivalent to those used for the reactor vessel.

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

1. Ultrasonic Testing - A 100% volumetric ultrasonic test of reactor vessel plate for shear wave was performed in addition to code requirements. This 100% volumetric ultrasonic test is a severe requirement, but it assures that the plate is of the highest quality.
2. Radiation Surveillance Program - In the surveillance program, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading (WOL) fracture mechanics test specimens. This program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, in accordance with ASTM-E-185, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels". The surveillance program does not include thermal control specimens. These specimens are not required since the surveillance specimens will be exposed to the combined neutron irradiation and temperature effects and the test results will provide the maximum transition temperature shift. Thermal control specimens as considered in ASTM-E-185 would not provide any additional information on which the operational limits for the reactor vessel are set.

4.5.1.1.1 Radiation Surveillance Program

The reactor vessel surveillance program uses eight specimen capsules which is more than the minimum number recommended by ASTM-E-185. The capsules are located about 3 inches from the vessel wall directly opposite the center portion of the core. Sketches of an elevation and plan view showing the original location and dimensional spacing of the capsules with relation to the core, thermal shield and vessel and weld seams is shown in Figures 4.5-1 and 4.5-2, respectively. The capsules can be removed when the vessel head is removed, and can be replaced when the internals are removed. The capsules contain reactor vessel steel specimens from the limiting shell plate located in the core region of the reactor and associated weld metal and heat affected zone metal. (As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01% will be made for surveillance material base metals and as deposited weld metal.) In addition, 64 correlation monitors made from fully documented specimens of SA-533 Grade B Class 1 material obtained through Subcommittee II of ASTM Committee E10, Radioisotopes and Radiation Effects, are inserted in the capsules for Unit No. 1 only. The eight capsules contain approximately 32 tensile specimens, 352 Charpy V-notch specimens (which include weld metal and heat affected zone material) and 32 WOL specimens. Dosimeters including Ni, Cu, Fe, Co-A1, Cd shielded Co-A1, Cd shielded Np-237 and Cd shielded U-238 are placed in filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity. The complete capsule is helium leak tested. Vessel material sufficient for at least 2 capsules will be kept in storage should the need arise for additional replacement test capsules in the program.

The eight capsules for Unit No. 1 contain the following types and number of specimens.

Material	**Capsule S, V, W, and X**			**Capsule T and U**			**Capsule Y and Z**		
	Charpy	Tensile	WOL	Charpy	Tensile	WOL	Charpy	Tensile	WOL
Limiting Plate*	10	--	--	10	2	4	10	2	--
Limiting Plate**	10	2	4	10	--	--	10	--	--
Weld Metal	8	2	--	8	2	--	8	2	4
HAZ	8	--	--	8	--	--	8	--	--
Correlation Monitors	8	--	--	8	--	--	8	--	--

* Specimens machined in the major rolling direction of the plate.

** Specimens machined normal to the major rolling direction of the plate.

The eight capsules for Unit No. 2 contain the following types and number of specimens.

Material	Capsule S, V, W, and X			Capsule T, U, Y and Z		
	Charpy	Tensile	WOL	Charpy	Tensile	WOL
Limiting Plate*	8	--	--	8	--	--
Limiting Plate**	12	2	4	12	2	--
Weld Metal	12	2	--	12	2	4
HAZ	12	--	--	12	--	--

Each capsule contains the following dosimeters and thermal monitors.

4.5.1.1.2 Dosimeters

Pure Cu
Pure Fe
Pure Ni
CoA1 (0.15% Co)
CoA1 (Cadmium shielded)
U238 (Cadmium shielded)
NP237 (Cadmium shielded)

4.5.1.1.3 Thermal Monitors

97.5% Pb, 2.5% Ag (579°F MP)
97.5% Pb, 1.75% Ag, 0.75% Sn (590°F MP)

* Specimens machined in the major rolling direction of the plate.
** Specimens machined normal to the major rolling direction of the plate.

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the adjacent vessel wall because the specimens are located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the RT_{NDT} measurements are representative of the vessel at a later time in life.

The calculated maximum fast neutron (E > 1 MeV) exposure of the reactor vessel at the clad/base metal interface after 32 EFPY of cumulative operating time is 1.802 x 10^{19} n/cm^2 for Unit 1 and 1.625 x 10^{19} n/cm^2 for Unit 2.

The reactor vessel surveillance capsules were originally located at orientations shown in Figure 4.5-2. The capsule lead factors (ratio of fast fluence at the capsule location versus that at the vessel inner wall) for Unit 1 and 2 are listed below:

	Unit 1	**Unit 2**
Capsule Identification	**Lead Factor**	**Lead Factor**
T	3.51	3.48
X	3.51	3.46
Y	3.51	3.47
U	3.50	3.44
W, V, Z	1.23	1.22
S	1.82*	1.22

*The projected Unit 1 lead factor for Capsule S above correlates to 32 EFPY. The cumulative lead factor for Capsule S will continue to decrease with increased EFPY based on relocations of Capsule S in the reactor vessel in 1995 and 2010. The projected lead factors as a function of EFPY are shown in LTR-REA-10-79.

The Unit 1 capsule at the 4° position was originally known as capsules S while the Unit 1 capsule at the 184° position was originally known as capsule W. In 1995 the capsule at the 4° position was moved to the 40° position and was re-designated as capsule W while the capsule at the 184° position was re-designated as capsule S. These changes were documented in AEP Safety Review Screening Checklist CE-95-0309, dated 9/19/95.

During Unit 1 refueling outage in March 2010 (U1C23), Capsule "W" was formally relocated back to its 4 degree location and re-named as capsule "S", (which was its original designation). The designation of Capsule "S" at 184 degrees was changed back to "W" which was its original designation.

Correlations between the calculations and the measurements on the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Sub-Section 4.5.1.3 and indicate good agreement.

The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution will be considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure will be made by use of data on withdrawn capsules. Capsules T, U, X, and Y have been removed and tested for both units.

The schedule for removal of capsules is as follows:

Removal Time Effective Full Power Years (EFPY)			
Capsule	**Unit 1**	**Capsule**	**Unit 2**
T	1.27	T	1.09
X	3.48	X	5.27
Y	4.95	Y	3.24
U	9.17	U	8.65
W	32		
S	Standby	S	32
V, Z	Standby	V, W, Z	Standby

CNP has committed to pulling and testing one additional standby capsule at each unit between 32 and 48 EFPY to cover the peak fluence expected at 60 years. A fluence update will be performed at 32 EFPY when Capsules W (Unit 1) and S (Unit 2) are pulled and tested. A subsequent fluence update will be performed when the standby capsules at each unit are pulled and tested between 32 EFPY and 48 EFPY.

4.5.1.1.4 Surveillance Capsule Program Update - Unit One Only

In accordance with the Radiation Surveillance Program described previously, capsules T, X, Y and U were removed from Unit One reactor vessel at the defined removal time. Capsules S, V, W, and Z remain in the vessel.

Subsequent to the removal of capsule "U", Unit One began operating at a reduced reactor coolant system temperature and pressure (RTP) to improve original steam generator longevity. As a result of this change, it was decided to relocate one of the remaining radiation surveillance capsules to a higher lead factor area to determine the potential embrittlement affects of operating at reduced temperature and pressure.

To address these changes from an administrative standpoint, the following changes have been made to the capsule designations. It should be noted that the specimens located within the capsules are physically stamped with their original letter designations.

The Unit 1 capsule at the 4° position was originally known as capsules S while the Unit 1 capsule at the 184° position was originally known as capsule W. In 1995 the capsule at the 4° position was moved to the 40° position and was re-designated as capsule W while the capsule at the 184° position was re-designated as capsule S. These changes were documented in AEP Safety Review Screening Checklist CE-95-0309, dated 9/19/95.

Figure 4.5.2a shown in revision 21.2 of the UFSAR provides a surveillance capsule location view for Unit One based on the capsule movements and re-designations described above.

During Unit 1 refueling outage in March 2010 (U1C23), Capsule “W” was formally relocated back to its 4 degree location and re-named as capsule “S”, which was its original designation. The designation of Capsule “S” at 184 degrees was changed back to “W”, which was its original designation.

At the conclusion of Fuel Cycle 14, the original Capsule “S” from location of 4 degree was relocated to 40 degree location in 1995 and was relocated back to 4 degree location in 2010. The 40 degree location is a higher lead factor location and Capsule “S” accumulated a higher dose for about 15 years. Therefore, the total irradiation history of Capsule S encompassed Fuel Cycles 1 through 14 at the 4º location, Fuel Cycles 15 through 22 at the 40º location, and Fuel Cycles 23 and beyond at the 4º location. At the time of capsule relocation, D. C. Cook Unit 1 had operated for a total of 22.381 Effective Full Power Years (EFPY).

The impact of the relocation of D. C. Cook Unit 1 Reactor Vessel Materials Surveillance Capsule “S” from a quadrant equivalent azimuthal location of 40º to an azimuthal location of 4º relative to the core cardinal axes at the conclusion of Fuel Cycle 22 has been documented in a Westinghouse report noted in Reference (2) in Section 4.5.1.4.

By this back relocation of the capsule, Figure 4.5-2a becomes superseded and Figure 4.5-2 provides the location of remaining four capsules going forward from U1C23.

4.5.1.1.5 Quality Control Program

Table 4.5-1 summarizes the quality control program with regard to inspections performed on reactor coolant system components. In addition to the inspections shown in Table 4.5-1, there were those performed by the equipment supplier to confirm the adequacy of material he received, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME code requirements. The inspection procedures and acceptance standards required on piping materials and piping fabrication were governed by USAS B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were reviewed by qualified engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the accessible external surfaces of the primary reactor coolant system pressure containing segments received a 100% surface inspection by Magnetic Particle or Liquid Penetrant Testing after hydrostatic test (See Table 4.5-1). Reactor vessel plate material was subjected to angle beam as well as straight beam ultrasonic testing to give maximum assurance of quality. Reactor vessel forgings received the same inspection. In addition, 100% of the material volume was covered in these tests as an added assurance over the grid basis required in the code.

Quality Control engineers monitored the supplier's work, witnessing key inspections not only in the supplier's shop but in the shops of subvendors of the major forgings and plate material. Normal surveillance included verification of records of material, physical and chemical properties, review of radiographs, performance of required tests, and qualification of supplier personnel.

Section III of the ASME Code required that nozzles carrying significant external loads are attached to the shell by full penetration welds. This requirement was carried out in the reactor coolant piping, where auxiliary pipe connections to the reactor coolant loop were made using full penetration welds.

The Reactor Coolant System components were welded under procedures, which required the use of both preheat and post-heat.

Preheat requirements, not mandatory under Code rules, were performed on weldments including P1 and P3 materials which are the materials of construction in the reactor vessel, pressurizer and steam generators. The purpose of using both preheat and post-heat of weldments was to produce tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones. Post-heating achieves this by tempering any hard zones, which may have formed due to rapid cooling.

4.5.1.1.6 Electroslag Weld Quality Assurance

The 90° elbows used in the reactor coolant loop piping are electroslag welded. The following efforts were performed for quality control of these components.

1. The electroslag welding procedure employing one wire technique was qualified in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section IX and Code Case 1355 plus supplementary evaluations as requested by WNES-PWRSD. The following test specimens were removed from a 5-inch thick weldment and successfully tested. They are:
   a. 6 Transverse Tensile Bars - as welded
   b. 6 Transverse Tensile Bars - 2050°F, H20 Quench
   c. 6 Transverse Tensile Bars - 2050°F, H20 Quench + 750°F stress relief heat treatment
   d. 6 Transverse Tensile Bars - 2050°F, H20 Quench, tested at 650°F
   e. 12 Guided Side Bend Test Bars
2. The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The acceptance standards were ASTM E-186 severity level 2 except no category D or E defectiveness was permitted and USAS Code Case N-10, respectively.
3. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were USAS Code Case N-10.
4. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with USAS Code Case N-10.
5. Weld metal and base metal chemical and physical analyses were determined and certified.
6. Heat treatment furnace charts were recorded and certified.

Reactor coolant pump casings fabricated by electroslag welding were qualified as follows:

1. The electroslag welding procedure employing two and three wire technique was qualified in accordance with the requirements of the ASME Code Section IX and Code Case 1355 plus supplementary evaluations as requested by WNES-PWRSD. The following test specimens were removed from an 8 inch thick and from a 12 inch thick weldment and successfully tested for both the 2 wire and the 3 wire techniques, respectfully. They are:
   a. Two wire electroslag process - 8” thick weldment.
      1. 6 Transverse Tensile Bars - 750°F post weld stress relief
      2. 12 Guided Side Bend Test Bars
   b. Three wire electroslag process - 12” thick weldment
      1. 6 Transverse Tensile Bars - 750°F post weld stress relief
      2. 17 Guided Side Bend Test Bars
      3. 21 Charpy-V Notch Specimens
      4. Full section macro examination of weld and heat affected zone
      5. Numerous microscopic examinations of specimens removed from the weld and heat affected zone regions
      6. Hardness survey across weld and heat affected zone
   c. A separate weld test was made using the 2-wire electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an 8 inch thick weldment in the stop-re-start-repaired region and successfully tested. They are:
      1. 2 Transverse Tensile Bars - as welded
      2. 4 Guided Side Bend Test Bars
      3. Full section macro examination of weld and heat affected zone.
   d. The weld test blocks in a., b., and c. above were radiographed using a 24 MeV Betatron. The radiographic quality level as defined by ASTM E-94 obtained was between one-half of 1% to 1%. There were no discontinuities evident in any of the electroslag welds.
2. The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 except no category D or E defectiveness was permitted for section thickness up to 4½ inches and ASTM E-280 severity level 2 for section

thicknesses greater than 4½ inches. The penetrant acceptance standards were ASME Code Section III, paragraph N-627.

3. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Code Section III, paragraph N-627.
4. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Code Section III, paragraph N-627.
5. Weld metal and base metal chemical and physical analyses were determined and certified.
6. Heat treatment furnace charts were recorded and certified.

4.5.1.1.7 In-Process Control of Variables

There are many variables that must be controlled in order to maintain desired quality welds. These, together with an explanation of their relative importance are as follows:

1. Heat Input vs Output

   The heat input is determined by the product of volts times current and is measured by voltmeters and ammeters, which are considered accurate, as they are calibrated every 30 days. During any specific weld these meters are constantly monitored by the operators.

   The ranges specified are 500-620 amperes and 44-50 volts. The amperage variation, even though it is less than ASME allows by Code Case 1355, is necessary for several reasons:

   a. The thickness of the weld is in most cases the reason for changes.
   b. The weld gap variation during the weld cycle will also require changes. For example, the procedure qualifications provide for welding thicknesses from 5” to 11” with two wires. The current and voltage are varied to accommodate this range.
   c. Also, the weld gap is controlled by spacer blocks. These blocks must be removed as the weld progresses. Each time a spacer block is removed there is the chance of the weld pinching down to as much as 1” or opening to perhaps as much as 1½”. In either case, a change in current may be necessary.
   d. The heat output is controlled by the heat sink of the section thickness and metered water flow through the watercooled shoes. The nominal temperature of the discharged water is 100°F.

2. Weld Gap Configuration

   As previously mentioned, the weld gap configuration is controlled by 1¼" spacer blocks. As these blocks are removed there is the possibility of gap variation. It has been found that a variation from 1" to 1-3/4" is not detrimental to weld quality as long as the current is adjusted accordingly.

1. Flux Chemistry

   The flux used for welding is Arcos BV-1 Vertomax. This is a neutral flux whose chemistry is specified by Arcos Corporation. The molten slag is kept at a nominal depth of 1-3/4" and may vary in depth by plus or minus 3/8" without affecting the weld. This is measured by a stainless steel dipstick.

2. Weld Cross Section Configuration

   It is noted that the higher the current or heat input and the lower the heat output the greater the dilution of weld metal with base metal. This causes a more round barrel-shaped configuration as compared to welding with less heat input and higher heat output. This would cut the amount of dilution to provide a more narrow barrel-shaped configuration. This is also a function of section thicknesses; the thinner the section, the more round the pattern that is produced.

3. Welder Qualification

   Welder qualification is in accordance with ASME Code, Section IX rules, using transverse side bend test specimens per Table Q.24.1.

4.5.1.2 Reactor Coolant System In-Service Inspection Program

1. Introduction

   a. Basis

      The in-service inspection program is based, as far as is practicable, on the applicable edition of Section XI of the ASME Code, In-Service Inspection of Nuclear Power Plant Components. Since this code was not available during the early design stages, 100-percent compliance may not be feasible. Access for in-service examination was considered during the final design, and modifications were made where practical to make provisions for maximum accessibility within the limits of the basic plant design. For details of these provisions, see item 4, Access For Examination.

   b. Methods of Examination

      Although ultrasonic techniques will be used for most of the volumetric examination, radiography may be used on small-diameter piping as well as for cast structures and other areas where material characteristics do not allow the use of ultrasonic techniques.

      The method of examination planned for each area - volumetric, surface, or visual - is detailed as part of the in-service inspection program.

2. Inspection Program

   The ISI Program Plan was developed according to the requirements of 10 CFR 50.55(a), paragraph g(4). Every ten years the program will be updated in accordance with 10 CFR 50.55(a), paragraph g(4).

3. Examination Techniques and Personnel Qualifications

   The in-service examinations are scheduled to be performed by several techniques. These techniques are listed below with some comments as to their applicability.

   a. Visual Examination - A visual examination can readily be made of the exterior of the piping and its supports, valves and their supports, pumps and their supports, and the pressurizer and the steam generators when the insulation has been removed. With the insulation in place, a visual examination can determine relatively gross changes, leaks, support hanger settings, etc. A visual examination can be made of the exterior of the upper closure head outside the control rod drive structure, but a visual examination of the interior surface of the head will require either decontamination or remote techniques. A gross visual examination of the interior of the reactor can be made through the use of monoculars or binoculars of 5X to 10X magnification. A critical visual examination can be made with remote viewing equipment such as borescopes or television systems. Direct or remote visual examination of the internal surfaces of the vessel can indicate surface scratches or evidence of corrosion, erosion, misalignment, or movement, but it is not believed to be capable of determining cracking in the cladding. Mechanical removal of the oxide layer can make cracks visible by direct or remote visual observation, but this technique is only feasible on selected areas.

In general, visual examination shall be conducted in conformance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. Surface Examination

1. Magnetic Particle Examination - This examination is applicable only to ferromagnetic materials and, in general, shall be conducted in conformance with Section XI of the ASME Boiler and Pressure Vessel Code.

2. Liquid Penetrant Examination - This examination is applicable to any nonporous surface. As with visual techniques, liquid penetrant techniques usually cannot detect cracks in the cladding after the system has been in service unless the surface oxide is mechanically removed. Any liquid penetrant examination conducted shall be in conformance with Section XI of the ASME Boiler and Pressure Vessel Code.

c. Volumetric Examination

Radiography - This technique was used to examine almost all of the pressure boundary welds in the system covered by Section III of the ASME Boiler and Pressure Vessel Code. These techniques may be duplicated in-service on most of these welds with the exception of those in the reactor pressure vessel. However, as pointed out for the pump casing welds, such radiography may be very difficult during in-service inspection due to the background radiation and the geometry involved. Thus, where radiography must be employed during in-service inspection, techniques and results that are not in conformance with Section XI of the ASME Boiler and Pressure Vessel Code may be required. However, any radiography to be performed, will be in accordance with the Code as far as is practical.

Ultrasonic Examination - This will be performed using both manual and automated techniques, depending upon the radiation level and the geometry involved. The procedures used for the in-service inspection program, in general, conform with Section XI of the ASME Boiler and Pressure Vessel Code.

d. Personnel Qualification

Personnel performing the nondestructive examination operation are qualified in accordance with SNT-TC-1A as applicable to the examination techniques and methods used. All in-service nondestructive examinations will be performed by or under the direct supervision of Level II personnel. Personnel performing examinations not covered by SNT-TC-1A, shall be qualified to comparable levels of competence by subjection to comparable examinations.

Detailed procedures have been written for all examination methods.

4. Access for Examination

a. Plant Accessibility

The following provisions and modifications were performed within the limits of the basic plant design to maximize the plant accessibility for examination purposes:

1. The reactor vessel closure head is stored dry on the reactor operating deck during refueling to facilitate visual examination.
2. Reactor vessel studs, nuts and washers are removed todry storage during refueling.
3. Removable plugs are provided in the primary shield just above the nozzle welds, and the metallic insulation covering the nozzle welds is readily removable.
4. Those components and piping requiring insulation and subject to in-service inspection according to Section XI are provided with readily removable metallic insulation over all welds (except for the reactor vessel shell and those piping welds in the primary shield and crane wall penetrations).
5. The reactor coolant pump design was modified to include a removable coupling (spool piece) between the motor and pump to facilitate reactor coolant pump seal inspection and/or replacement.
6. Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the lower internals.
7. A removable manway is provided in the lower internals lower core support plate to allow remote visual examination of the reactor vessel bottom head without removal of the lower internals.
8. Manways are provided in the steam generator channel head and pressurizer top head to allow access for internal examination.
9. Subsequent to the formal issue of the Section XI code, requirements were established to locate, where possible, piping

deadweight and seismic supports and rupture restraints to reduce to a minimum the interference with examination.

b. Reactor Vessel

The reactor vessel presents special problems in access because of the radiation levels and the need for remote underwater accessibility. Because of these limitations, several steps were incorporated into the design and fabrication of the vessel to facilitate the preparation for in-service examination. These are:

1. The design of the reactor vessel in the core area is a clean, uncluttered cylindrical surface to permit positioning of in-service examination tools without obstruction.
2. Shop ultrasonic examinations were performed on internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow ultrasonic testing of the base metal. Indications of cladding separation whose amplitude equals or exceeds that of a 3/4T reference hole were not permitted.
3. Subsequent to the reactor vessel shop hydrostatic test, important areas of the reactor vessel were ultrasonically tested and mapped.

4.5.1.3 Determination of Reactor Vessel RTNDT

A. Measurement of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

The current plans for the Unit 1 and Unit 2 Reactor Vessel Irradiation Surveillance Program, as developed by Southwest Research Institute, contain the detailed procedures that describe the methods used to test the irradiation samples and to determine their specific activity and analyze and compute the neutron fluence exposure. The procedure is generally applicable for testing and analysis of any gamma-emitting specimens but is used specifically for:

1. 59Co (n,γ) 60Co
2. 58Ni (n, p) 58Co
3. 54Fe (n, p) 54Mn
4. 63Cu (n,α) 60Co

The procedures are based on ASTM Method E181, “Standard General Methods for Analysis of Radioisotopes”, ASTM Method E261,”Standard Method for Measuring Neutron Flux by Radioactivation Techniques”, and ASTM E185, “Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessel”. The procedures are used to correlate the neutron fluence with the changes in reactor vessel material fracture toughness properties so that the heat-up and cool-down limitations can be determined in accordance with requirements of the Technical Specifications. See Section 4.4 for further details.

B. Calculation of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples (Reference 1)

As part of the surveillance testing and evaluation program, the neutron transport and dosimetry analysis serves two purposes: (1) to determine the neutron fluence (E > 1.0 MeV) in the surveillance capsule where the metallurgical test specimens are located and (2) to determine the neutron fluence (E > 1.0 MeV) incident on and within the reactor pressure vessel (RPV).

The methodology for RPV fluence determination is based on combining results of transport calculations with measured dosimeter activities. The transport calculations provide three important sets of data in the overall analysis: (1) spectrum-weighted, effective dosimeter cross sections, (2) lead factors for various locations in the RPV, and (3) fluence rates at locations of interest.

The calculated effective cross sections for different dosimeters are divided into the measured reaction rates in order to obtain the fluence rate (E > 1.0 MeV) at the capsule location. The corresponding fluence rates at various depths into the RPV are obtained by dividing the capsule fluence rate by the appropriate lead factors. Both the effective cross sections and the lead factors depend only on ratios of computed results so that absolute calculations are not required. The measured dosimeter activities provide the fluence rate normalization. However, absolute fluence rates are calculated to compare with measurements to provide a measure of the uncertainty involved in the RPV fluence determination procedure.

Industry developments have improved upon neutron flux calculation techniques used in earlier Southwest Research capsule analysis reports for Units 1 & 2. The capsule analysis reports should be consulted for the specific technique employed to compute the neutron flux. The current technique utilizes a discrete ordinates calculation using an updated version of the Dot 4 one and two-dimensional neutron/photon transport code to obtain the radial ® and azimuthal (θ) fluence-rate distribution for the vessel geometry and capsule location. The inclusion of the surveillance capsules in the R-θ model accounts for the significant perturbation effects from the physical presence of the capsules. A 47-group energy structure for the SAILOR cross-section, and S8 angular structure and a P3 Legrendre cross-section are used in the computations.

C. Measurement of the Initial RTNDT of the Reactor Pressure Vessel Base Plate and Forgings Material

The unirradiated or initial RTNDT temperature of pressure vessel base plate and forgings material is 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 RTNDT temperature is defined in ASTM E208 as “the temperature at which a specimen is broken in a series of tests in which duplicate no break performance occurs at 10°F higher temperature”. Using the Charpy V-notch test, the RTNDT is defined as the temperature at which the energy required to break the specimen is a certain “fixed” value.

For ASTM A533B Class 1, A508 Class 2 and A508 Class 3 steel the ASME III Table N-421 specifies an energy value of 30 ft-lb. This value is based on a correlation with the drop weight test and is 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 RTNDT.

As part of the surveillance program, Charpy V-notch impact tests, tensile tests, and fracture mechanics specimens are taken from the core region plates and forgings, and core region weldments including heat-affected zone material. The test locations are similar to those used in the tests by the fabricator at the plate mill.

The uncertainties of measurement of the RTNDT of base plate are:

1. Differences in Charpy V-notch foot pound values at a given temperature between specimens.
2. Variation of impact properties through plate thickness.

The fracture toughness technology for pressure vessels and correlation with service failures based on Charpy V-notch impact data are based on the averaging of data. The Charpy V-notch 30 ft-lb “fix” temperature is based on multiple tests by the material supplier, the fabricator, and by Westinghouse as part of the surveillance program. In the review of available data, differences of 0°F to approximately 40°F are observed in comparing curves plotted through the minimum and average values respectively. The value of RTNDT derived from the average curve is judged to be representative of the material because of the averaging of at least 15 data points, consistent with the specified procedures of ASTM E23. In the case of the assessment of RTNDT shift due to fast neutron flux, the displacement of transition curves is measured. The selection of maximum, minimum or average curves for this assessment is not significant since like curves are used.

There are quantitative differences between the RTNDT temperature measurements at the surface, ¼ thickness or the center of a plate. Differences in RTNDT temperature between ¼ T and the center in heavy plates had been observed to vary from improvement in the RTNDT temperature to increases up to

85°F. The RTNDT temperature at the surface had been measured to be as much as 85°F lower than at ¼ T.

The ¼ T location is considered conservative since the enhanced metallurgical properties of the surface are not used for the determination of RTNDT temperature. In addition, the limiting RTNDT temperature for the reactor vessel after operation is based on the RTNDT temperature shift due to irradiation. Since the fast neutron dose is highest at the inner surface, usage of the ¼ T RTNDT temperature criterion is conservative.

Data are being accumulated on the variation of RTNDT across heavy section steels at Westinghouse Nuclear Energy Systems. Similarly, the Pressure Vessel Research Committee sponsors an evaluation of properties of pressure vessel steels in plates and forgings greater than 6 inches thick. Preliminary data show RTNDT temperature differences between ¼ T and center of less than 20°F. The present criteria of using RTNDT temperature + 60°F at the ¼ T location without taking advantage of the enhanced properties at the surface of reactor vessel plates is conservative.

To assess any possible uncertainties in the consideration of RTNDT temperature shift for welds, heat affected zone, and base metal, test specimens of these three "material types" are included in the reactor vessel surveillance program.

4.5.1.4 References for Section 4.5

1. "Reactor Vessel Material Surveillance Program for Cook Nuclear Plant Unit 2 Analysis of Capsule X," P. K. Nair, et. al. (May 1987)
2. Westinghouse letter No. AEP-10-89 dated May 20, 2010 and its attachment Engineering Report No. LTR-REA-10-79

System Design and Operating Parameters

	Unit 1	Unit 2
Original plant design life, years [1]	40	40
Number of heat transfer loops	4	4
Design pressure, psig	2485	2485
Nominal operating pressure, psig	2235	2235
Approximate total RCS volume (including pressurizer and surge line) with 0% steam generator tube plugging [2]	12,540 $ft.^3$	12,470 $ft.^3$
Approximate system liquid volume (including pressurizer water) with 0% steam generator tube plugging [2]	11,640 $ft.^3$	11,570 $ft.^3$
Approximate system liquid volume (including pressurizer water) at maximum guaranteed power with 0% steam generator tube plugging [3]	11,990 $ft.^3$	12,019 $ft.^3$
Total Reactor heat output (100% power) Btu/hr	$12{,}283 \times 10^6$ (3600 MWt)	$12{,}283 \times 10^6$ (3600 MWt)

[1] Licensed life is 60 years in accordance with Chapter 15 of the UFSAR.

[2] This value is a best estimate based on ambient (70° F) conditions with 0% steam generator tube plugging. Refer to Westinghouse letter AEP-98-161 and IMP database SEC-SAI-4824-CO.

[3] This includes a 3% volume increase (1.3% for thermal expansion and 1.7% for pipe connections to the reactor coolant loops, volume in the rod drive mechanisms and calculation inaccuracies).

System Design and Operating Parameters

	Unit 1	Unit 2
	Bounding Conditions for Rerating Lower/Upper [4]	Bounding Conditions for Rerating Lower/Upper [4]
Reactor vessel coolant temperature at full power:		
Inlet, nominal, ºF	511.7/549.3	511.7/549.3
Outlet, nominal, ºF	582.3/616.9	582.3/616.9
Coolant temperature rise in vessel at full power, avg., ºF	70.6/67.6	70.6/67.6
Total coolant flow rate, lb/hr x 10^6	139.5/133.2	139.5/133.2
Steam pressure at full power, psia	576/820	576/820
Steam Temp. @ full power, ºF	481.8/521.0	481.8/521.0
Approximate total RCS volume (including pressurizer and surge line) with 0% steam generator tube plugging. [2]	12,540 $ft.^3$	12,470 $ft.^3$

[4] Limiting values based upon 3600 MWt rerating condition in WCAP-12135.

Reactor Coolant System Design Pressure Settings

	Pressure (psig)	
	Unit 1	Unit 2
Design Pressure	2485	2485
Operating Pressure	2235	2235
Safety Valves	2485	2485
Power Relief Valves[1]	2335	2335
Pressurizer Spray Valves (Begin to Open)	2260	2260
Pressurizer Spray Valves (Full Open)	2310	2310
Pressurizer Pressure High - Reactor Trip	≤2385	≤2385
High Pressure Alarm	2310	2310
Pressurizer Pressure Low - Reactor Trip	≥1950	≥1950
Low Pressure Alarm	2210	2210
Pressurizer Pressure Low - Safety Injection	≥1815	≥1815
Hydrostatic Test Pressure	3106	3107[2]
Backup Heaters On	2210	2210
Proportional Heaters (Begin to Operate)	2250	2250
Proportional Heaters (Full Operation)	2220	2220

[1] During startup and shutdown, a manually energized safeguard circuit is in service while the reactor coolant system temperature is below 266°F for Unit 1 and 299°F for Unit 2. This allows automatic opening of that Unit's two power relief valves at ≤435 psig for low temperature overpressure protection (LTOP) of the reactor vessel. This safeguard circuit ensures that the reactor pressure remains below the ASME Section III, Appendix G "Protection against Non-ductile Failure" limits in the case of an LTOP event.

[2] Original design

Reactor Vessel Design Data

Design Pressure, psig	2485
Operating Pressure, psig	2235
Hydrostatic Test Pressure, psig	3107[1] Unit 2 and 3106[2] psig Unit 1
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in.	43-9 $^{11}/_{16}$ (Unit 1)
(Bottom Head O.D. to top of Control Rod Mechanism Adapter)	43-10 (Unit 2)
Thickness of Insulation, min., in	3
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs, in	7
ID of Flange, in	172½
OD of Flange, in	205
ID at Shell, in	173
Inlet Nozzle ID, in	27½
Outlet Nozzle ID, in	29
Clad Thickness, min., in	$^{5}/_{32}$
Lower Head Thickness, min., in (base metal)	5⅜
Vessel Belt-Line Thickness, min., in (base metal)	8½
Closure Head Thickness, in	6½
Total Water Volume Below Core, ft^3	1050
Water Volume in Active Core Region, ft^3	665

[1] Original design

[2] Steam Generator Replacement Pressure

Reactor Vessel Design Data

Total Water Volume to Top of Core, ft^3	2352	
Total Water Volume to Coolant Piping Nozzles Centerline, ft^3	2959	
Total Reactor Vessel Water Volume, ft^3 (estimated) (with core and internals in place)	4660[3]	
	Unit 1 Bounding Conditions for Rerating Lower / Upper	Unit 2
Reactor Coolant Inlet Temperature, °F	514.9 / 545.2	541.27
Reactor Coolant Outlet Temperature, °F	579.1 / 607.5	606.35
Reactor Coolant Flow, lb/hr x 10^6	139.0 / 133.9	134.6

[3] This volume is a general number that approximates either Unit's Vessel. The actual volume of either vessel is cycle dependent and can be obtained from the Westinghouse IMP database for the specific cycle.

Pressurizer and Pressurizer Relief Tank Design Data

Pressurizer	
Design Pressure, psig	2485
Operating Pressure, psig	2235
Hydrostatic Test Pressure (cold), psig	3106 Unit 1, 3107[1] Unit 2
Design/Operating Temperature, °F	680/653
Pressurizer Water Level, Full Power[2]	49.9% (Unit 1) / 55% (Unit 2)
Total Internal Volume[3], ft3	1800
Surge Line Nozzle Diameter, in.	14
Shell ID, in.	84
Electric Heater Capacity, kW [4]	1800 kW
Heatup rate of Pressurizer, °F/hr	55 (approx.)
Start-up Water Solid, °F/hr	40
Hot Standby Condition, °F/hr	70
Design Spray Rate for Valves Full Open, gpm	800
Continuous Spray Rate, gpm	1

1 Original design

2 Estimated values from operating data, actual values determined by T_{ave} full-load.

3 This volume is a general number that approximates either Unit's Pressurizer. The actual volume of either pressurizer can be obtained from the current Westinghouse IMP database for the Unit.

4 Some heaters may be removed from service to a practical limit, greater than the TS minimum capacity, which supports plant evolutions.

Pressurizer and Pressurizer Relief Tank Design Data

Pressurizer Relief Tank	
Design Pressure, psig	100
Rupture Disc Release Pressure, psig	100
Design Temperature, °F	340
Normal Water Temperature, °F	Containment Ambient (120°F Max.)
Normal Operating Pressure, psig	3
Normal Water Volume, ft3	1430
Normal Gas Volume, ft3	370
Cooling time required following design maximum discharge, hr.	Approx. 1
Number of spray nozzles	5
Total Spray Flow, gpm	150
Total Volume, ft3	1800
Total Rupture Disc Relief Capacity, saturated steam, lb/hr	1.6×10^6

STEAM GENERATOR DESIGN DATA*

	Unit 1	Unit 2
Number of Steam Generators	4	4
Design Pressure, Reactor Coolant/Steam, psig	2485/1085	2485/1085
Reactor Coolant Hydrostatic Test Pressure(tube side-cold), psig	3106	3107
Design temperature, Reactor Coolant/Steam, °F	650/600	650/600
Reactor Coolant Flow, lb/hr x 10^6	33.9[1] 38.2[2]	33.7
Total Heat Transfer Surface Area, ft^2	54,927	54,500
Rated Thermal Output/MWt)	816	852.75
Operating Parameters at 100% Load		
Primary Side:		
Heat Transfer Rate (per unit), Btu/hr x 10^6	2773[3] 3070[4]	2910
Coolant Inlet Temperature, °F	582.3 - 616.9	606.4
Coolant Outlet Temperature, °F	511.7 - 549.3	541.3
Flow Rate, (per unit), lb/hr x 10^6	33.9 (1) 3.82 (2)	33.7
Pressure loss, psi.	32.05	26.1
Secondary Side:		
Steam Temperature at full power, °F	481.8 - 521	521.1
Steam Flow, lb/hr x 10^6	3.53 (3) 3.9 (4)	3.685
Steam Pressure at full power, psia	575.8 - 819.7	820
Maximum moisture carryover, wt %	0.045	0.15
Feedwater Temperature	440 @ nozzle	431.3 @ #6 Heater Outlet
Fouling Factor, hr-ft^2 °F/Btu	0.00005	0.00005
Overall Height, ft-in	67 - 7.25	67-8
Shell OD, upper/lower, in	175.75/135	175.9/135
Number of U-tubes	3496	3592

*Quantities are for each steam generator.
[1] RCS flow rate based on Thermal Design Flow of 88500 gpm at 536°F.
[2] RCS flow rate based upon Mechanical Design Flow of 99700 gpm at 535°F.
[3] Heat transfer rate and steam flow for 816 MWt per steam generator.
[4] Heat transfer rate and steam flow for 900 MWt per steam generator.

STEAM GENERATOR DESIGN DATA*

U-tube outer Diameter, in	0.875	0.875
Tube Wall Thickness, (minimum), in	0.044	0.050
Number of manways/ID, in	2/18 2/16	4/16
Number of handholes/ID, in	6/6	6/6
Number of inspection ports/ID, in	2/6	2/4
Unit 1	**Rated Load**	**No Load**
Reactor Coolant Water Volume,** ft^3	1141.1[5]	1141.1 [(5)]
Primary Side Fluid Heat Content, Btu x 10^6	29.8[6]	29.2[7]
Secondary Side Water Volume, ft^3	2035[8]	3235
Secondary Side Steam Volume, ft^3	3583 [(8)]	2315
Secondary Side Fluid Heat Content, Btu x 10^7	5.69 [(8)]	8.78[9]
Unit 2	**Rated Load**	**No Load**
Reactor Coolant Water Volume,** ft^3	1112	1112
Primary Side Fluid Heat Content, Btu	29.0 x 10^6	28.46 x 10^6
Secondary Side Water Volume, ft^3	2077	3351
Secondary Side Steam Volume, ft^3	3589	2315
Secondary Side Fluid Heat Content, Btu	5.18 x 10^7	8.44 x 10^7

*Quantities are for each steam generator.
** Values may change subject to steam generator tube plugging.
[5] Hot condition @ power. Volume at ambient temperature is approximately 1130 $ft.^3$.
[6] Based upon hot volume and primary fluid @ 567.8°F and pressure of 2250 psia.
[7] Based upon hot volume and primary fluid @547°F and pressure of 2250 psia.
[8] Based upon secondary side fluid @ 515.2°F (saturated conditions).
[9] Based upon secondary side fluid @ 547°F (saturated conditions).

REACTOR COOLANT PUMPS DESIGN DATA[1]

Number of Pumps	4 Design
Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3106 Unit 1, 3107[2] Unit 2
Design Temperature (casing), °F	650
RPM at Nameplate Rating	1189
Suction Temperature, °F	536.3 (Unit 1)/541.00[3] (Unit 2)
Required net positive suction head, ft	170
Developed Head, ft	277
Capacity, gpm	88,500
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump Discharge Nozzle ID, in	27.5
Pump Suction Nozzle ID, in	31
Overall Unit Height, ft-in.	27-0
Water Volume, ft^3	81
Pump-Motor Moment of Inertia, lb-ft^2	82,000

MOTOR DATA:	
Type	**AC Squirrel Cage Induction, Single Speed, Air Cooled**
Voltage	4000
Insulation Class	F
Phase	3

[1] Quantities are for each pump.
[2] Original design.
[3] Original design power capability parameter.

MOTOR DATA:	
Type	**AC Squirrel Cage Induction, Single Speed, Air Cooled**
Frequency, Hz	60
Starting	
Current, amp	4800
Input (hot reactor coolant), kw	4337
Input (cold reactor coolant), kw	5663
Power, HP (nameplates)	6000
Pump Weight, lb. (dry)	175,200

Reactor Coolant Piping Design Parameters

Reactor inlet piping, ID, in	27.5
Reactor inlet piping, minimum[1] thickness, in	2.56
Reactor outlet piping, ID, in	29
Reactor outlet piping, minimum[1] thickness, in	2.69
Coolant pump suction piping, ID, in	31
Coolant pump suction piping, minimum[1] thickness, in	2.88
Pressurizer surge line piping, ID, in	11.188
Pressurizer surge line piping, nominal thickness, in	1.406
Design pressure, psig	2485
Operating Pressure, psig	2235
Hydrostatic test pressure (cold), psig	3106 (Unit 1) / 3107[2] (Unit 2)
Design temperature, ºF	650
Design temperature (pressurizer surge line)ºF	680
Design pressure, pressurizer relief line, psig	[3]
Design temperature, pressurizer relief lines, ºF	[1]
Water volume (all 4 loops without surge line), ft^3	1185
Surge line volume, ft^3	43

[1] Original procurement minimums
[2] Original design
[3] From pressurizer to safety valve: 2485 psig, 650°F; From safety valve to pressurizer relief tank: 500 psig, 470°F

Pressurizer Valves Design Parameters

Pressurizer Spray Control Valves	
Number	2
Design pressure, psig	2485
Design temperature, ºF	650
Design flow for valves full open, each, gpm	400
Fluid temperature, ºF	530-550
Position after failure of actuating force	Closed
Pressurizer Safety Valves	
Number	3
Relieving capacity, lb/hr	420,000
Set pressure, psig	2485
Fluid	Saturated steam
Constant backpressure:	
Normal, psig	3
Expected during discharge, psig	350
Pressurizer Power Relief Valves	
Number	3[1]
Design pressure, psig	2485
Design temperature ºF	680
Design capacity at nominal set pressure 2350 psia, (each) lbm/hr	210,000
Fluid	Saturated steam or water

[1] Only two required. Third valve is considered an installed spare.

Reactor Coolant System Design Pressure Drop

	Pressure Drop, psi (estimated)	
	Unit 1[1]	**Unit 2[2]**
Across Pump Discharge Leg	1.3/1.1	1.4 /1.2
Across Reactor Vessel, Including Nozzles	52.0/44.6	50.4 / 48.5
Across Hot Leg	1.2/1.0	1.2 / 1.1
Across Steam Generator, Including Nozzles	33.4/50.9	33.1 / 36.1
Across Pump Suction Leg	3.1/2.6	3.2 / 2.9
Total Pressure Drop	91.0/100.2	89.3 / 89.9

Note that the first value provided coincides with the maximum Best Estimate Flow (minimum steam generator tube plugging, minimum reactor vessel average temperature, TPR) and that the second value provided coincides with the minimum Best Estimate Flow (maximum steam generator tube plugging, maximum reactor vessel average temperature, TPI).

(1) Data updated as a result of new Best Estimate Flows calculated in 2018.
(2) Data updated as a result of new Best Estimate Flows calculated in 2017.

Design Thermal and Loading Cycles

Item	Transient	Cycles [1]
Level A Limits (Normal)		
1	Heatup at 100 °F/hr.	200
2	Cooldown at 100°F/hr. (Pressurizer @ 200 °F/hr.)	200
3	Unit Loading at 5% of full power/min.	18,300/11,680 [2 3]
4	Unit Unloading at 5% of full power/min.	18,300/11,680 [2 3]
5	Step Load Increase of 10% of full power	2,000 [4]
6	Step Load Decrease of 10% of full power	2,000 [4]
7	Large Step Decrease in load (with steam dump)	200

[1] For Unit 1 Model 51R replacement steam generator manway and handhole stud preloads, the design considers 100 cycles each of tensioning and detensioning or torquing and detorquing, as appropriate.

[2] Unit 1 rerating to 3600 MWt.

[3] The Unit 1 Model 51R replacement steam generators have been structurally designed for the lower cycle limit for both 3264 MWt and the 3600 MWt power uprate condition. The RCS average temperature and steam temperature will deviate ± 3 °F in one minute. The corresponding RCS pressure variation will be ± 100 psi.

[4] WCAP-17588-P, D. C. Cook Unit 1 Lower Radial Support Clevis Insert Acceptable Minimum Bolting Pattern Analysis, used 200 Step Load Increase of 10% of full power and 200 Step Load Decrease of 10% of full power transients to qualify the minimum bolting pattern. A new procedural limit was set to account for the lower number of transients allowed for the Unit 1 Clevis Insert Bolts. WCAP-17588-P does not impact any other analyses performed using the transients described in Table 4.1-10.

Design Thermal and Loading Cycles

Item	Transient	Cycles [1]
8	Hot Standby Operation	18,300 [5]
9	Turbine Roll Test	10
10	Steady State Fluctuations	Infinite [6]
	Level B Limits (Upset)	
11	Loss of Load (without immediate turbine or Reactor trip)	80
12	Loss of Power (blackout with natural circulation in Reactor Coolant System)	40
13	Loss of Flow (partial loss of flow one pump only)	80
14	Reactor Trip From Full Power	400
15	a) Operational Basis Earthquake (20 events of 20 cycles each event), except Reactor Vessel	400
	b) Operational Basis Earthquake, Reactor Vessel only (10 events of 20 cycles each event)	200
	Level C Limits (Emergency)	

[5] Applies to steam generator only. Reflects cyclic limit for the feed ring of a rapid injection of cold feedwater.

[6] Reactor coolant system average temperature is assumed to increase and decrease a max. of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psi.

Design Thermal and Loading Cycles

Item	Transient	Cycles [1]
	None	
	Level D Limits (Faulted)	
16	Reactor Coolant Pipe Break (LOCA)	1
17	SSE	1
18	Steam Pipe Break	1
	Test Conditions	
19	Primary Side Hydrostatic Tests Before Initial Startup @ 3107 psig	5 [7]
20	Primary Side ASME Section XI/Field Tests	10 [8]
21	Secondary Side Hydrostatic Test Before Initial Startup at 1356 psig	5 / 20 [8] [9]
22	Primary to Secondary Leak Test	50 / 90 [8]
23	Secondary to Primary Leak Test	120 [8]

[7] Unit 1 Model 51R replacement steam generator shop hydro was 3106 psig.
[8] Unit 1 Model 51R replacement steam generator.
[9] Unit 1 Model 51R replacement steam generator not subjected to secondary side shop hydro. Leakage test performed after installation in accordance with Code Case N-416-1.

SUMMARY OF PLANT OUTAGE FOR YANKEE-ROWE (1964 - 1969)

Starting Date	Duration Days/Hours		Outage Type	Case Equipment/System
1/17/64	-	3.1	Forced	Turbine Trip
2/12/64	-	21.8	Scheduled	Control Rod Drop Testing
3/11/64	-	4.5	Forced	Moisture separator level switch tripped due to vibration
3/26/64	-	4	Forced	Control Valves Sticking
5/18/64	-	5.4	Forced	Low condensate pump discharge pressure
8/2/64	35	-	Scheduled	Refueling and general maintenance
9/9/64	-	2.4	Scheduled	Check of Overspeed Trip
9/11/64	-	14.7	Forced	Spurious Reactor Trip
10/18/64	-	12.2	Forced	Condenser Noise
10/22/64	-	22.4	Forced	Neutron Counter Gain Control
2/12/65	-	15.2	Forced	Switch yard Electric
3/5/65	-		Scheduled	Switch yard Electric
8/9/65	93	6	Scheduled	Refueling
11/26/65	2	20	Scheduled	Turbine Repair-Physics Testing
2/4/66	-	3.12	Forced	Reactor Scram
4/4/66	-	89.5	Scheduled	Leaking Pressurizer Safety Valves

SUMMARY OF PLANT OUTAGE FOR YANKEE-ROWE (1964 - 1969)

Starting Date	Duration Days/Hours		Outage Type	Case Equipment/System
7/10/66	-	3.68	Forced	Reactor Scram
8/25/66	-	2.40	Forced	Reactor Scram
10/4/66	34	10.23	Scheduled	Refueling
2/24/66	-	2.88	Forced	Reactor Scram
12/28/66	-	2.12	Forced	Reactor Scram
3/8/67	11	21	Scheduled	Steam Generator Leak Repair
5/12/67	-	16.87	Scheduled	Condenser Cleaning
7/9/67	17	1.5	Scheduled	Steam Generator Leak Repairs
10/28/67	-	9	Scheduled	AEC Operator Examinations
10/13/67	-	2.6	Forced	Reactor Scram
3/23/68	38	-	Scheduled	Core VI-VII Refueling and maintenance
7/20/68	1	10	Scheduled	Repair Leak from No. 1 M.C. Pump Stator Cap
11/8/68	6	16.42	Scheduled	Repair No. 4 Main Coolant Pump Thermal Barrier Leak and other Maintenance
1/18/69	1	2.1	Scheduled	Operator Training
2/15/69	1	1.8	Scheduled	Operator Training
3/1/69	-	11	Scheduled	AEC Operator Examination

SUMMARY OF PLANT OUTAGE FOR YANKEE-ROWE (1964 - 1969)

Starting Date	Duration Days/Hours		Outage Type	Case Equipment/System
4/11/69	4	18	Forced	Repair Reactor Instrument Leak
7/17/69	-	4.8	Forced	Reactor Scram
8/2/69	53	18.5	Scheduled	Refueling Maintenance
10/16/69	-	6.1	Forced	Reactor Scram
10/29/69	-	12	Scheduled	Turbine Valve Flange Steam Leak Repair

UNIT 1 AND UNIT 2 - REACTOR COOLANT SYSTEM CODES[1]

Component	Code	UNIT 1 Addenda and Code Cases
Reactor Vessel	ASME III [2] Class A	1965 Edition through 1966 Winter Addenda, Code Cases 1332-2, 1358, 1339-2, 1335, 1359-1, 1338-3, 1336
Reactor Vessel Closure Head (RVCH)	Class 1 (RVCH)	1995 Edition through 1996 Addenda (RVCH)
Full Length Control Rod Drive Mechanisms	ASME III [2] Class 1	1965 Edition through 1966 Winter Addenda 1995 Edition through 1996 Addenda (RVCH)
Steam Generators (OSG Model 51 Steam Dome Shell)	ASME III [2] Class A	1965 Edition through 1966 Winter Addenda Code Cases 1401 and 1498
Steam Generators (RSG Model 51R)	ASME III [2] Class 1	1989 Edition (No Addenda), Code Cases N-20-3, N-71-15, N-411-1, N-474-1, 2142-1, 2143-1, N-401-1, and N-416-1
Reactor Coolant Pump Casings	No Code (Designed with ASME III [2] Article 4 as a Guide)	1968 Edition
Pressurizer	ASME III [2] Class A	1965 Edition through Winter 1966 Addenda, Code Cases 1401, 1459
Pressurizer Safety Valves	ASME III [2]	1968 Edition
Power Operated Relief Valves	B-16.5	
Main Reactor Coolant System Piping	B31.1 [1]	1967 Edition
Reactor Coolant System Valves	B-16.5 or MSS-SP-66, and ASME III 1968 Edition [2]	

[1] Repairs and replacement for pressure retaining components within the code boundary, and their supports, are conducted in accordance with ASME Section XI.

[2] ASME Boiler and Pressure Vessel Code, Section III-Nuclear Vessels

UNIT 1 AND UNIT 2 - REACTOR COOLANT SYSTEM CODES

Component	Code	UNIT 2 Addenda and Code Cases
Reactor Vessel	ASME III [2] Class A ASME III, Class 1 (RVCH)	1968 Edition (1968 Summer Addenda), Code Cases 1335-4 and N-60-5 1995 Edition through 1996 Addenda
Full Length Control Rod Drive Mechanisms	ASME [2] Class 1	1995 Edition through 1996 Addenda
Steam Generators	ASME III [2] Class A	1968 Edition through Winter 1968 Addenda, Code Cases 1401, 1498 for upper assemblies and 1983 Edition through Summer 1984 for replacement lower assemblies
Reactor Coolant Pump Casings	No Code (Designed with ASME III [2] Article 4 as a Guide)	1968 Edition through Summer 1969 Addenda
Pressurizer	ASME III [2] Class A	1965 Edition through Winter 1966 Addenda
Pressurizer Safety Valves	ASME III [2]	1968 Edition
Power Operated Relief Valves	B16.5	
Main Reactor Coolant System Piping	B31.1 [1]	1967 Edition
Reactor Coolant System Valve	B-16.5 or MSS-SP-66, and ASME III, 1968 Edition [2]	

COMPONENT TRANSIENT LIMITS[1]

Component	Cyclic Or Transient Limit	Design Cycle Or Transient
Reactor Coolant System	200 heatup cycles @ ≤100ºF/hr and 200 cooldown cycles @ ≤100ºF/hr. (pressurizer cooldown @ ≤200ºF/hr)	Heatup cycle - T_{avg} from ≤200ºF to ≥547ºF. Cooldown cycle - T_{avg} from ≥547ºF to ≤200ºF.
	80 loss of load cycles	Without immediate turbine or reactor trip
	40 cycles of loss of offsite AC electrical power	Loss of offsite AC electrical power source supplying the onsite Class 1E distribution system
	80 cycles of loss of flow in 1 reactor coolant loop	Loss of only 1 reactor coolant pump
	400 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	200 large step decreases in load	100% to 5% of RATED THERMAL POWER with steam dump
	Operating basis earthquake	400 cycles - 20 earthquakes of 20 cycles each (except Reactor Vessel)
		200 cycles – 10 earthquakes of 20 cycles each (Reactor Vessel only)
	50 leak tests	Pressurized to 2500 psia
	5 hydrostatic pressure tests	Pressurized to 3107 psig (3106 psig for Unit 1 Model 51R)
Secondary System	1 steam line break	Break in a steam line 5.5" equivalent diameter
	5 hydrostatic pressure tests	Pressurized to 1356 psig

[1] A log of the actual number of transients is maintained.

MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Material	
		Unit 1	Unit 2
Reactor Vessel	Pressure Plate	ASTM A-533 Grade B Class 1	ASTM A-533 Grade B Class 1
	Pressure Forgings (excl. RVCH)	ASTM A508, Class 2	ASTM A508, Class 2
	RVCH	SA-508 Grade 3, Class 1	SA-508 Grade 3, Class 1
	Primary Nozzle Safe Ends	Type 316 forging overlaid on I.D. and O.D. with Type 308L and Inconel weld metal after final post-weld heat treatment	Type 316 forging overlaid on I.D. and O.D. with Type 316 weld metal prior to final post-weld heat treatment
	Cladding, Stainless	Combination of Type 308, Type 309 and Type 312	Type 308L, Type, 309L
	Stainless Weld Rod	Type 308, Type 309	Type 308L, Type 309, Type 309L, Type 316
	O-Ring Head Seals	Inconel - 718	Inconel - 718
	CRDM's	Inconel and Stainless Type 304	Inconel and Stainless Type 304
	Studs	SA - 540 Grade B - 24	SA - 540 Grade B - 24
	Instrumentation Nozzles	Inconel and Stainless End Type 304	Inconel and Stainless End Type 304
	Insulation	Stainless Steel	Stainless Steel
Steam Generator	Pressure Plate	ASTM A 533 Grade A Class 1	ASTM A 533 Grade A Class 1 for upper assembly (steam dome), ASTM
	Pressure Forgings Tubesheets	SA-508 Class 3a	ASTM A - 508 Class 2 A
	Transition Cone & Stub Barrels	SA-508 3a	ASTM A - 508 Class 3
	Primary Nozzle Safe Ends	SA-336 Class F316N/F316LN	Stainless steel weld metal - carbon steel to stainless steel juncture on O.D. overlaid with Type 309 and 308L weld metal
	Cladding, Stainless	ER 308L, ER309L	Type ER 309L
	Stainless Weld Rod	Type 308L, Type 309	Type 308L, Type 309L
	Cladding for Tube Sheets	UNS NO6082	Inconel
	Tubes	SB-163 Alloy 690 TT	Inconel - 690 (TT)

MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Material	
		Unit 1	Unit 2
	Channel Head Castings (Unit 1 Forging)	SA-508 Class 3A	ASTM A – 216 Grade WCC
Pressurizer	Shell	SA - 533 Grade A (Class 1)	SA - 533 Grade A (Class 2)
	Heads	SA – 216 Grade WCC	SA - 533 Grade A (Class 2)
	Support Skirt	SA - 516 Grade 70	SA - 516 Grade 70
	Nozzle Weld Ends	SA – 182 F316	SA – 182 F316
	Inst. Tube Coupling	SA – 182 F316	SA – 182 F316
	Cladding, Stainless	Type 308, Type 309 (modified)	Type 308 Type 309 (modified)
	Nozzle Forgings	Integrally cast with head	SA - 508 Class 2 Mn - Mo
	Heater Support Baffle Plate	SA - 240 Type 304	SA - 240 Type 304
	Inst. Tubing	SA - 213 Type 316	SA - 213 Type 316
	Heater Well Tubing	SA – 213 Type 316 Seamless	SA - 213 Type 316 Seamless
	Heater Well Adaptor	SA - 182 F316	
Pressurizer Relief Tank	Shell	ASTM A- 285 Grade C	ASTM A- 285 Grade C
	Heads	ASTM A-285 Grade C	ASTM A -285 Grade C
	Internal Coating	Amercoat 55	Amercoat 55
Pipe	Pipes	ASTM A-351 Grade CF8M ASTM A - 376 Grade TP 304 or TP 316	ASTM A-351 Grade CF8M ASTM A- 376 Grade TP 304 or TP 316
	Fittings	ASTM A-351 Grade CF8M	ASTM A-351 Grade CF8M
	Nozzles	ASTM A- 182 Grade F316	ASTM A- 182 Grade F316
Pump	Shaft	ASTM A-182 Grade F347	ASTM A- 182 Grade F347
	Impeller	ASTM A-351 Grade CF8M	ASTM A- 351 Grade CF8M
	Casing	ASTM A-351 Grade CF8M	ASTM A- 351 Grade CF8M
Valves	Pressure Containing Parts	ASTM A-351 Grade CF8M and ASTM A-182 Grade F316	ASTM A- 351 Grade CF8M and ASTM A- 182 Grade F316

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical Conductivity	Determined by the concentration of boric acid and alkali present. Expected range is 1 to 40 µMhos/cm at 25ºC.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25ºC.
Oxygen, ppm, max.	0.10
Chloride, ppm, max	0.15
Fluoride, ppm, max.	0.15
Hydrogen, cc (STP)/kg H_20	25 – 50
Total Suspended Solids, ppm, max.	1.0
pH Control Agent (Li^7OH)	Reactor coolant pH is controlled during power operation by adjusting lithium as a function of the coolant boron concentration.
Boric Acid as ppm B	Variable from 0 to 4000

SUMMARY OF ESTIMATED PRIMARY PLUS SECONDARY STRESS INTENSITY FOR COMPONENTS OF THE REACTOR VESSEL (UNIT 1)[1]

Item	Stress Intensity (psi)		Allowable Stress (psi) (at Operating Temperature)
	Original	Rerated	
Control Rod Housing	55,300	66,050	69,900
Head Flange	50,400	54,380	80,100
Vessel Flange	45,350	65,850	80,100
Primary Nozzles Inlet	48,400	49,860	80,000
Primary Nozzles Outlet	54,060	59,580	80,000
Stud Bolts	95,870	83,320	104,400
Core Support Pad	40,800	69,700	69,900
Bottom Head to Shell	34,100	34,530	80,000
Bottom Instrumentation	53,400	51,490	69,900
Vessel Wall Transition	37,900	33,570	80,000

[1] The vessel stress intensities for Unit 2 are available in the Unit 2 Stress Report.

SUMMARY OF ESTIMATED CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL (UNIT 1)[1]

Item	Usage Factor [2 and 3]	
	Original	Rerated
Control Rod Housing	.06	0.81
Head Flange	.015	0.185
Vessel Flange	.005	0.092
Stud Bolts	.310	0.449
Primary Nozzles Inlet	0.020	.098
Primary Nozzles Outlet	0.028	.063
Core Support Pad (lateral)	0.015	.693
Bottom Head to Shell	0.003	.018
Bottom Instrumentation	0.142	.122
Vessel Wall Transition	0.002	.007

[1] The usage factors for Unit 2 are available in the Unit 2 Stress Report.
[2] Covers all transients.
[3] As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

REACTOR COOLANT SYSTEM QUALITY CONTROL PROGRAM

Component		RT*	UT**	PT***	MT****	ET*****
1.	Steam Generator					
	1.1 Tube Sheet					
	1.1.1 Forging		yes		yes	
	1.1.2 Cladding		yes(+)	yes(++)		
	1.2 Channel Head					
	1.2.1 Casting	yes			yes	
	1.2.2 Forging (Unit 1 Model 51R)		yes		yes	
	1.2.3 Cladding			yes		
	1.3 Secondary Shell & Head					
	1.3.1 Plates		yes			
	1.3.2 Forgings (Unit 1 Model 51R)		yes		yes	
	1.4 Tubes		yes			yes
	1.5 Nozzles (forgings)		yes		yes	
	1.6 Weldments					
	1.6.1 Shell, longitudinal	yes			yes	
	1.6.2 Shell, circumferential	yes			yes	
	1.6.3 Cladding (Channel Head-Tube Sheet joint cladding restoration)			yes		
	1.6.4 Steam and Feedwater Nozzle to shell	yes			yes	
	1.6.5 Support brackets				yes	
	1.6.6 Tube to tube sheet			yes		
	1.6.7 Instrument connections (primary and secondary)				yes	

* Radiographic
** Ultrasonic
*** Dye Penetrant
**** Magnetic Particle
***** Eddy Current
(+) Flat Surfaces Only
(++) Weld Deposit Areas Only

REACTOR COOLANT SYSTEM QUALITY CONTROL PROGRAM

Component		RT*	UT**	PT***	MT****	ET*****
	1.6.8 Temporary attachments after removal				yes	
	1.6.9 After hydrostatic test (after welds and complete channel head - where accessible)				yes	
	1.6.10 Nozzle safe ends (if forgings)	yes		yes		
	1.6.11 Nozzle safe ends (if weld deposit)			yes		
2.	Pressurizer					
	2.1 Heads					
	2.1.1 Casting	yes			yes	
	2.1.2 Cladding			yes		
	2.2 Shell					
	2.2.1 Plates		yes		yes	
	2.2.2 Cladding			yes		
	2.3 Heaters					
	2.3.1 Tubing(+++)		yes	yes		
	2.3.2 Centering of element				yes	
	2.4 Nozzle		yes	yes		
	2.5 Weldments					
	2.5.1 Shell, longitudinal	yes			yes	
	2.5.2 Shell, circumferential	yes			yes	
	2.5.3 Cladding			yes		
	2.5.4 Nozzle Safe End (if forging)	yes		yes		
	2.5.5 Nozzle Safe End (if weld deposit)			yes		
	2.5.6 Instrument Connections			yes		
	2.5.7 Support Skirt				yes	
	2.5.8 Temporary Attachments after removal				yes	

(+++) Or a UT and ET

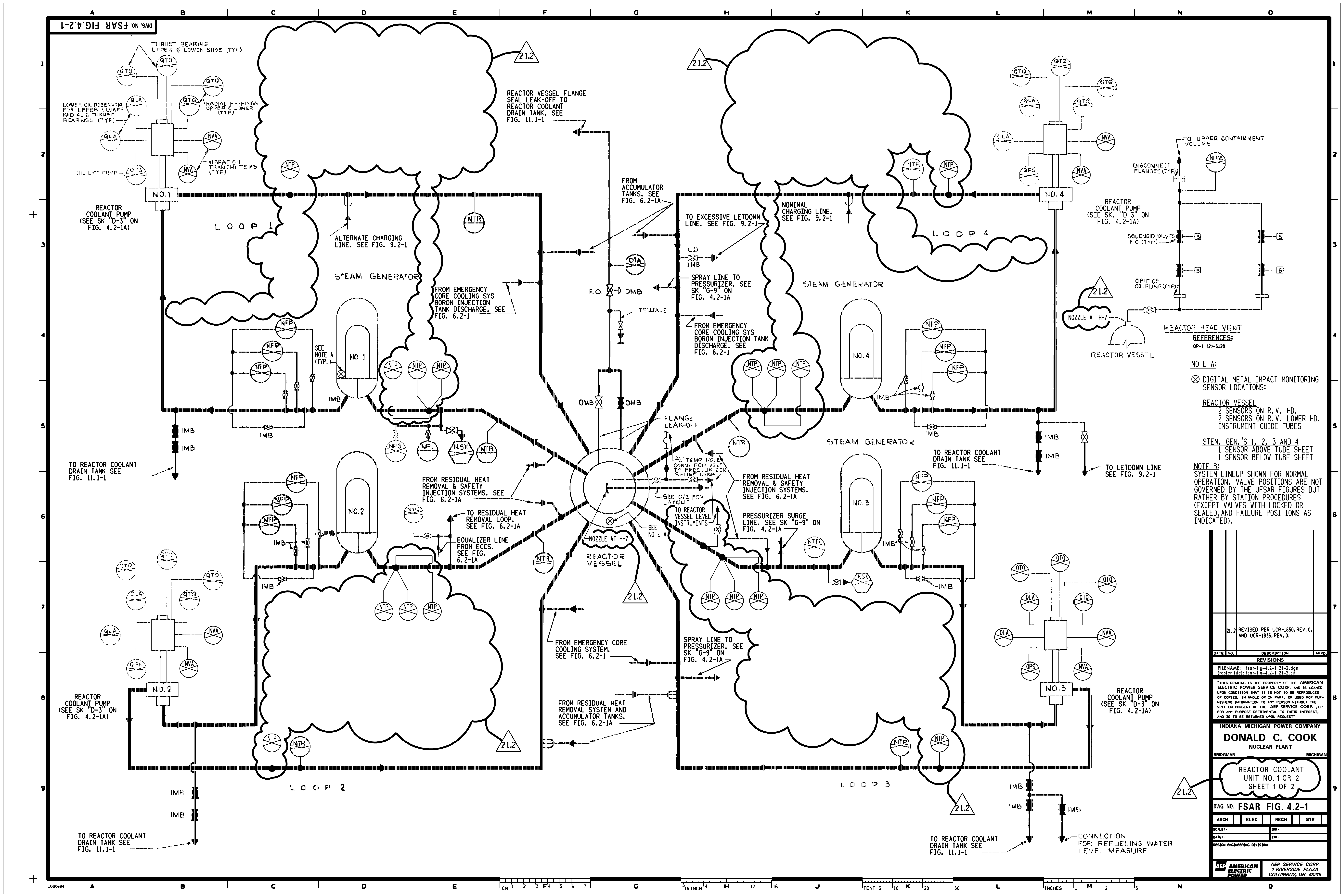
REACTOR COOLANT SYSTEM QUALITY CONTROL PROGRAM

	Component	RT*	UT**	PT***	MT****	ET*****
	2.5.9 All welds and cast heads after hydrostatic test				yes	
	2.6 Final Assembly					
	2.6.1 All accessible surfaces after hydrostatic test				yes	
3.	Piping repairs and replacements are conducted in accordance with ASME Section XI					
	3.1 Fittings and Pipe (Castings)	yes		yes		
	3.2 Fittings and Pipe (Forgings)		yes	yes		
	3.3 Weldments					
	3.3.1 Circumferential	yes		yes		
	3.3.2 Nozzle to runpipe (except no RT for nozzles less than 4 inches)	yes		yes		
	3.3.3 Instrument connections			yes		
4.	Pumps					
	4.1 Castings	yes		yes		
	4.2 Forgings					
	4.2.1 Main Shaft		yes	yes		
	4.2.2 Main Studs		yes	yes		
	4.2.3 Flywheel (Rolled Plate)		yes			
	4.3 Weldments					
	4.3.1 Circumferential	yes		yes		
	4.3.2 Instrument connections			yes		
5.	Reactor Vessel					
	5.1 Forgeries					
	5.1.1 Flanges		yes		yes	
	5.1.2 Studs		yes		yes	
	5.1.3 Head Adapters		yes	yes		
	5.1.4 Head Adapter Tube		yes	yes		

REACTOR COOLANT SYSTEM QUALITY CONTROL PROGRAM

	Component	RT*	UT**	PT***	MT****	ET*****
	5.1.5 Instrumentation Tube		yes	yes		
	5.1.6 Main Nozzles		yes		yes	
	5.1.7 Nozzle safe ends (if forging is employed)		yes	yes		
	5.2 Plates		yes		yes	
	5.3 Weldments					
	5.3.1 Main Steam	yes			yes	
	5.3.2 CRD Head Adapter Connection			yes		
	5.3.3 Instrumentation tube connection			yes		
	5.3.4 Main nozzles	yes			yes	
	5.3.5 Cladding		Yes (++++)	yes		
	5.3.6 Nozzle-safe ends (if forging)	yes		yes		
	5.3.7 Nozzle safe ends (if weld deposit)	yes		yes		
	5.3.8 Head adaptor forging to head adaptor tube	yes		yes		
	5.3.9 All welds after hydrotest				yes	
6.	Valves					
	6.1 Castings	yes		yes		
	6.2 Forgings (No NDE for valves two inches and smaller)		yes	yes		

(++++) UT of Clad Bond-to-Base Metal



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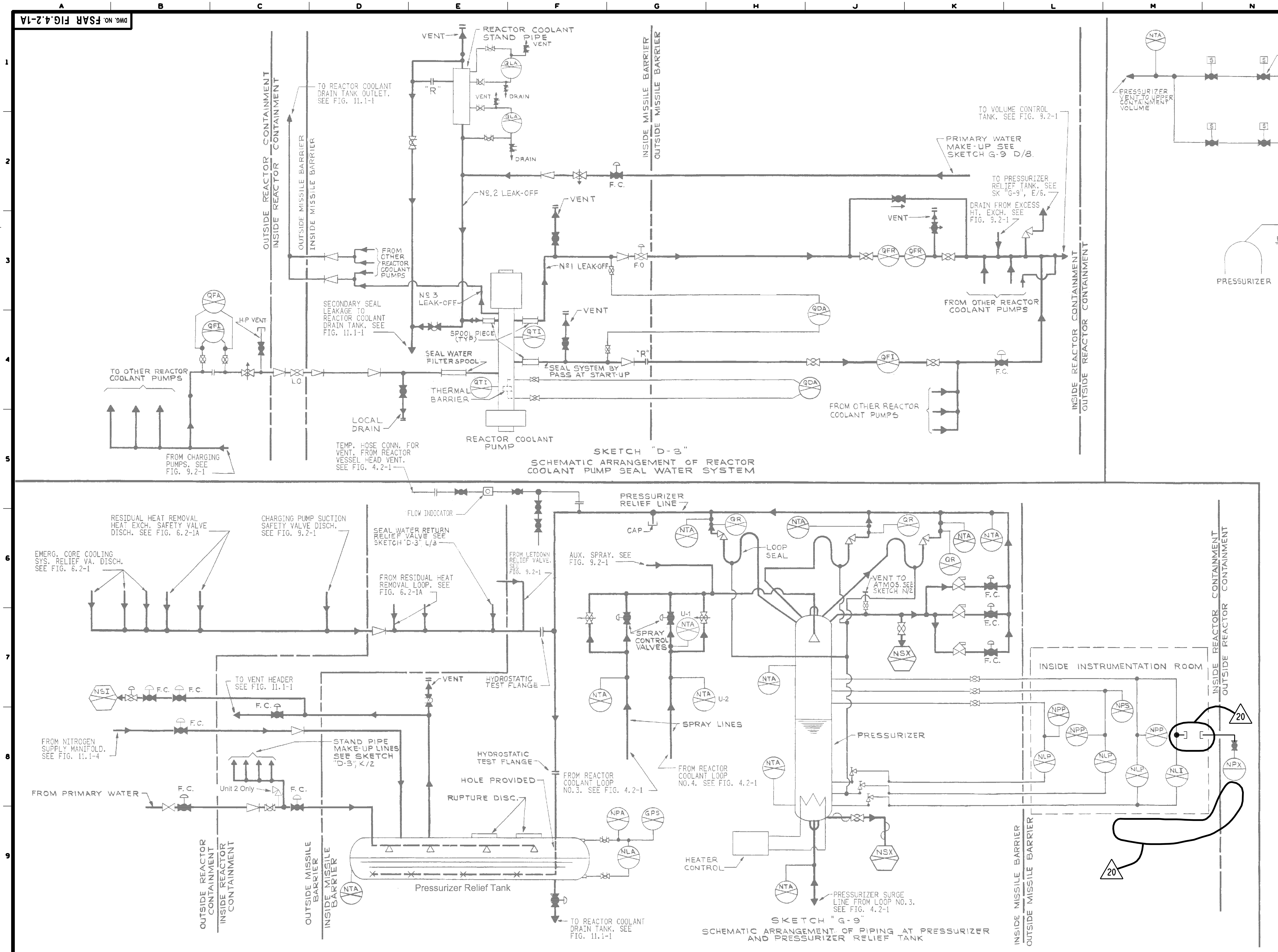


NOTE
SYSTEM LINEUP SHOWN FOR NORMAL OPERATION.
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).

DATE	NO.	DESCRIPTION	APPD.
	20	REVISED PER UCR-1712.	

REVISIONS

FILENAME: fsar-fig-4-2-1a 20.dgn
(raster file): fsar-fig-4-2-1a 20.tif

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK
NUCLEAR PLANT

BRIDGMAN MICHIGAN

REACTOR COOLANT
UNIT NO. 1 OR 2
SHEET 2 OF 2

DWG. NO. FSAR FIG. 4.2-1A

ARCH | ELEC | MECH | STR

SCALE:
DATE:
DR:
CH:
DESIGN ENGINEERING DIVISION:

AEP AMERICAN ELECTRIC POWER — AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215

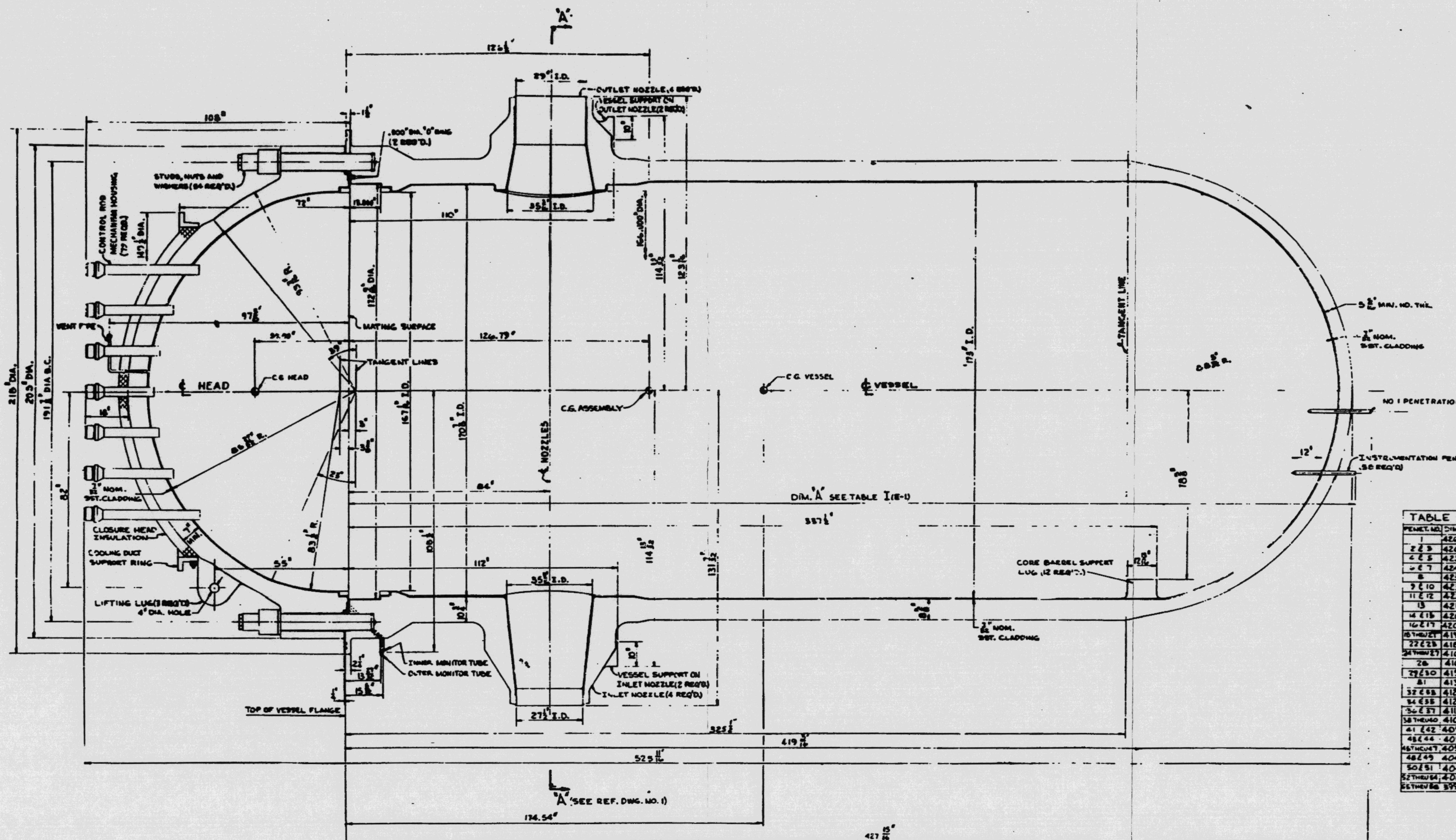


TABLE I

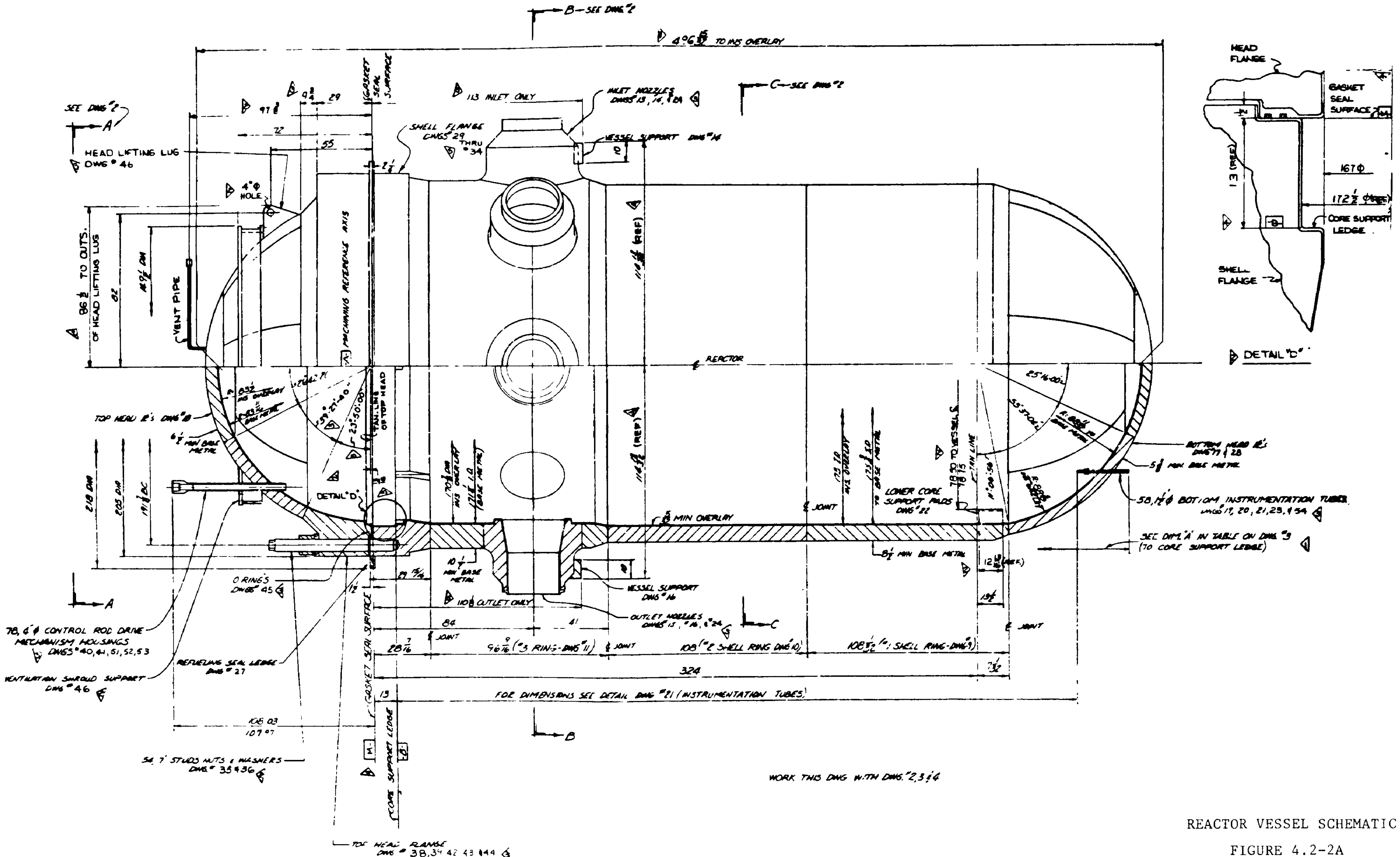
PENET. NO.	DIM "A"
1	426 1/16"
2 & 3	426 1/16"
4 & 5	425 9/32"
6 & 7	424 25/32"
8	423 3/4"
9 & 10	423 11/32"
11 & 12	422 31/32"
13	421 1/4"
14 & 15	420 9/16"
16 & 17	420 1/8"
18 THRU 21	419 21/32"
22 & 23	418 15/16"
24 THRU 27	416 13/16"
28	416 13/32"
29 & 30	415 5/32"
31	413 13/16"
32 & 33	412 27/32"
34 & 35	412 1/16"
36 & 37	411 11/32"
38 THRU 40	410 7/32"
41 & 42	409 25/32"
43 & 44	405 31/32"
45 THRU 47	405 1/2"
48 & 49	404 1/2"
50 & 51	404"
52 THRU 54	401 19/32"
55 THRU 56	399 21/32"

COMPONENT	WEIGHT POUNDS
VESSEL	678,587 #
CLOSURE HEAD	153,905 #
STUD, NUT & WASHER	33,420 #
TOTAL	865,812 #

REACTOR VESSEL SCHEMATIC

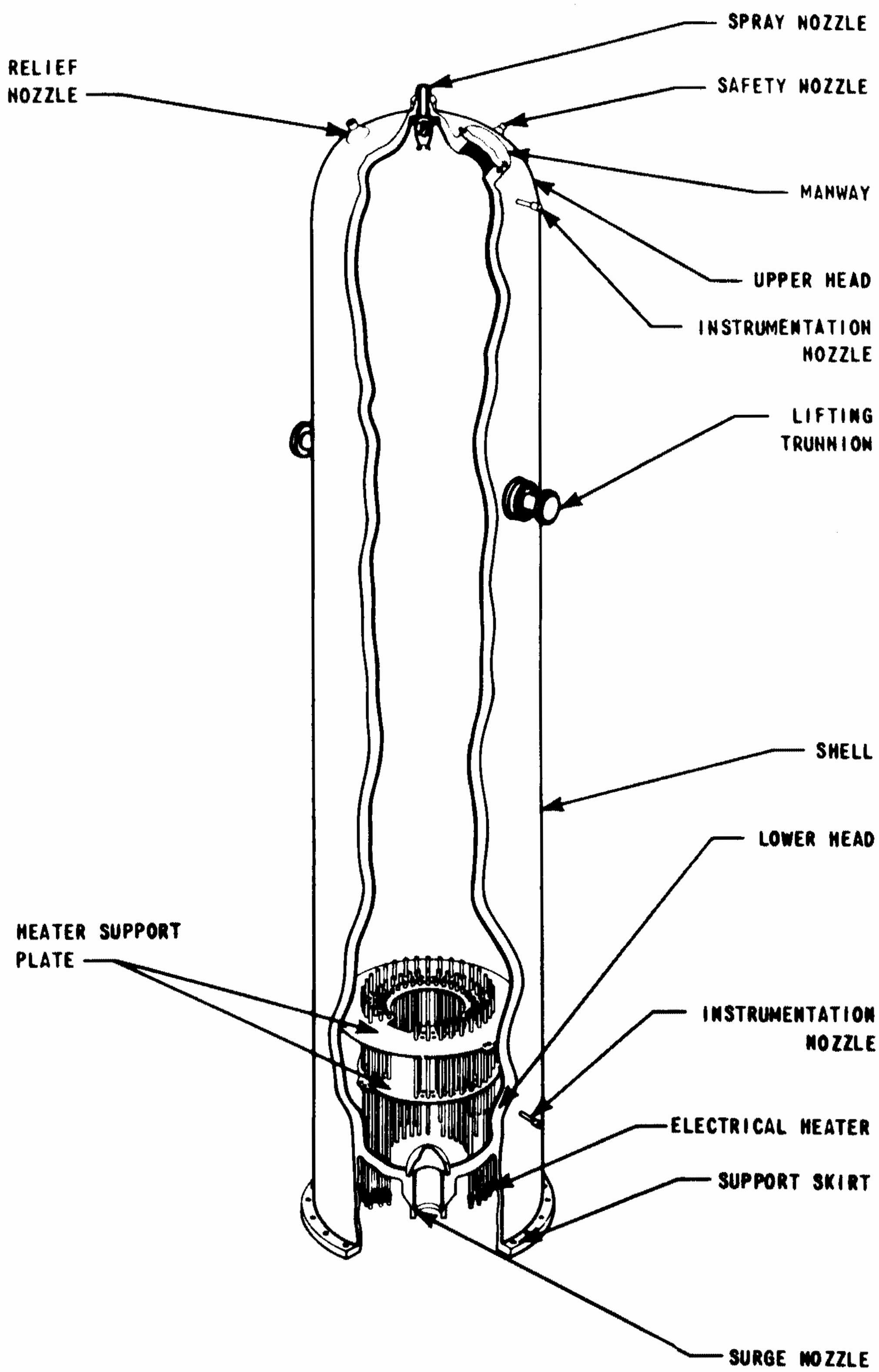
FIGURE 4.2-2

July, 1982



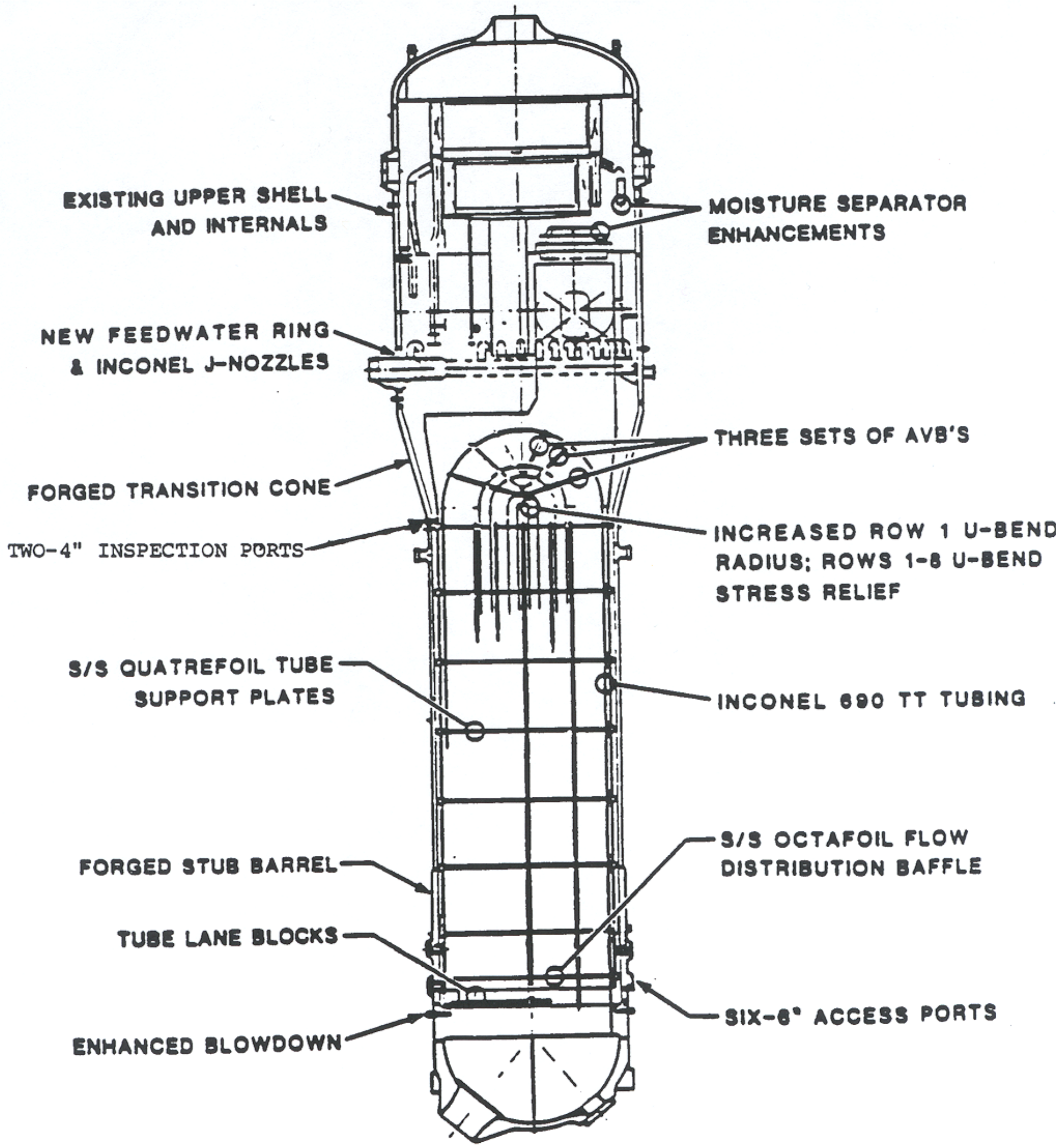
REACTOR VESSEL SCHEMATIC

FIGURE 4.2-2A
July, 1982



PRESSURIZER
FIGURE 4.2-3

July, 1982

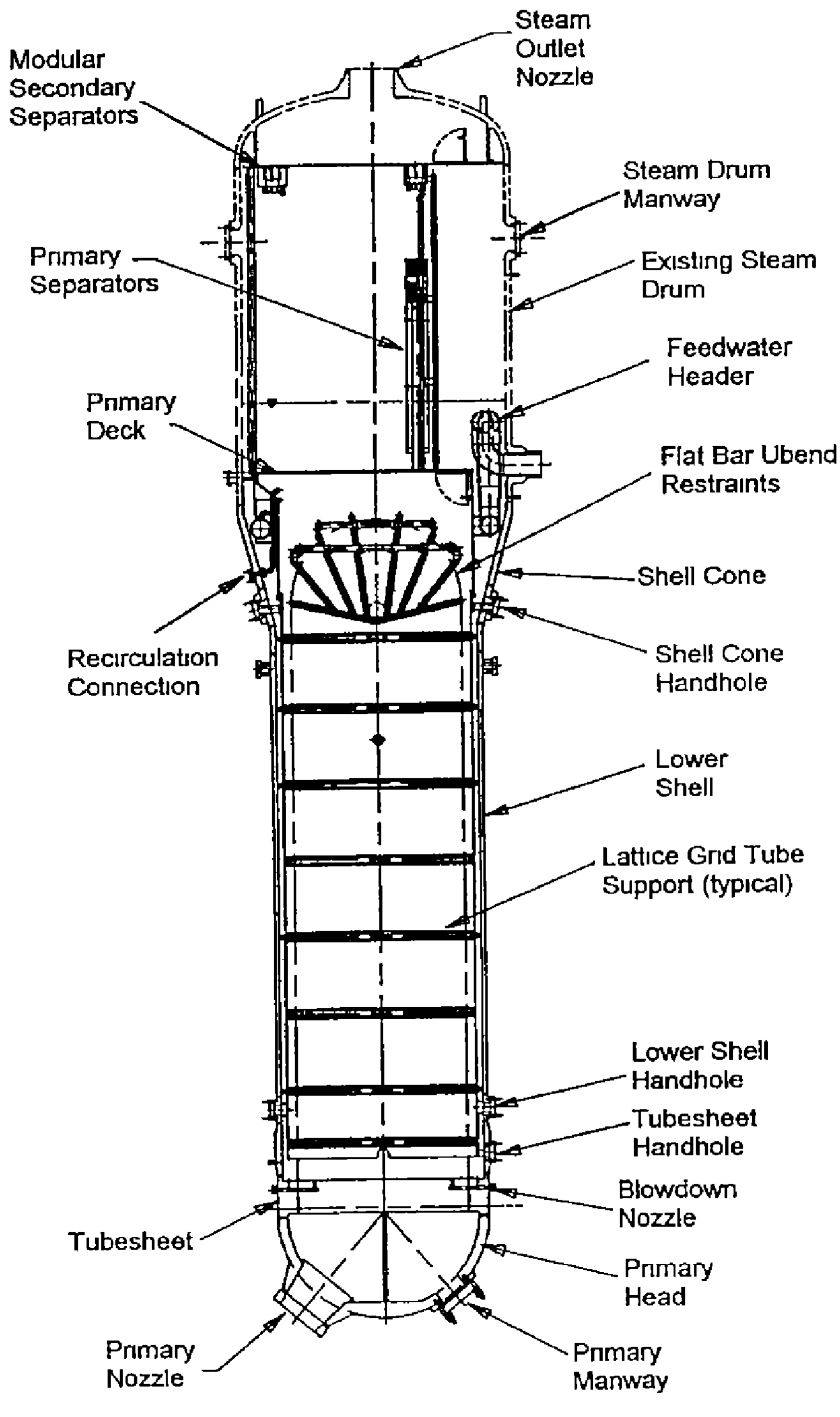


MODEL 51F STEAM GENERATOR

43

REVISION 0

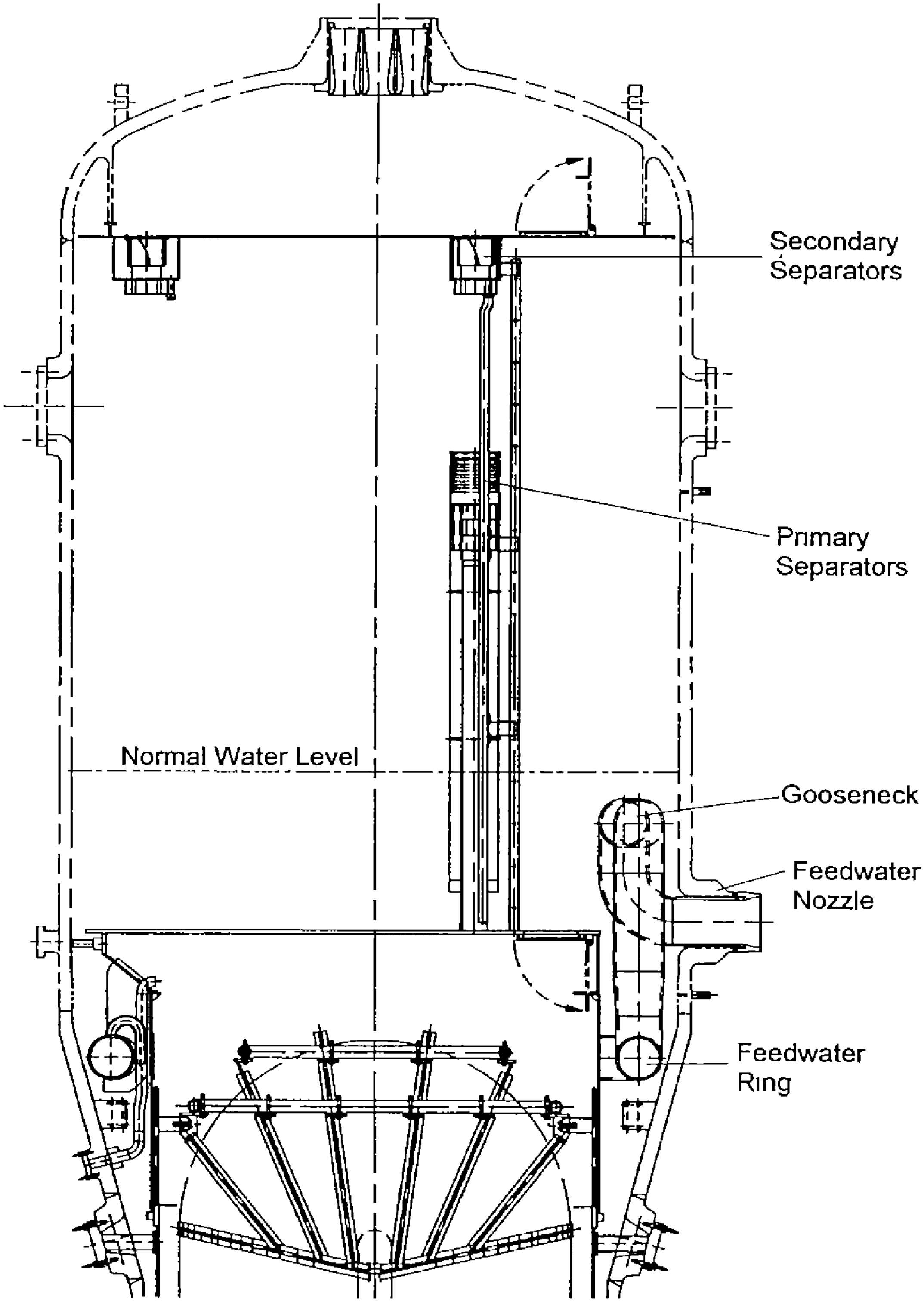
Revision: **18**	Change Description: **UCR-1649**	
AMERICAN ELECTRIC POWER COOK NUCLEAR PLANT NUCLEAR GENERATION GROUP BRIDGMAN, MICHIGAN	Title: **Cook Unit 2 Repaired Steam Generator General Arrangement**	
	UFSAR Figure: **4.2-4**	Sheet 1 of 1



16.6	REVISED PER 99-UFSAR-1226
REV. NO.	**DESCRIPTION**

REVISIONS

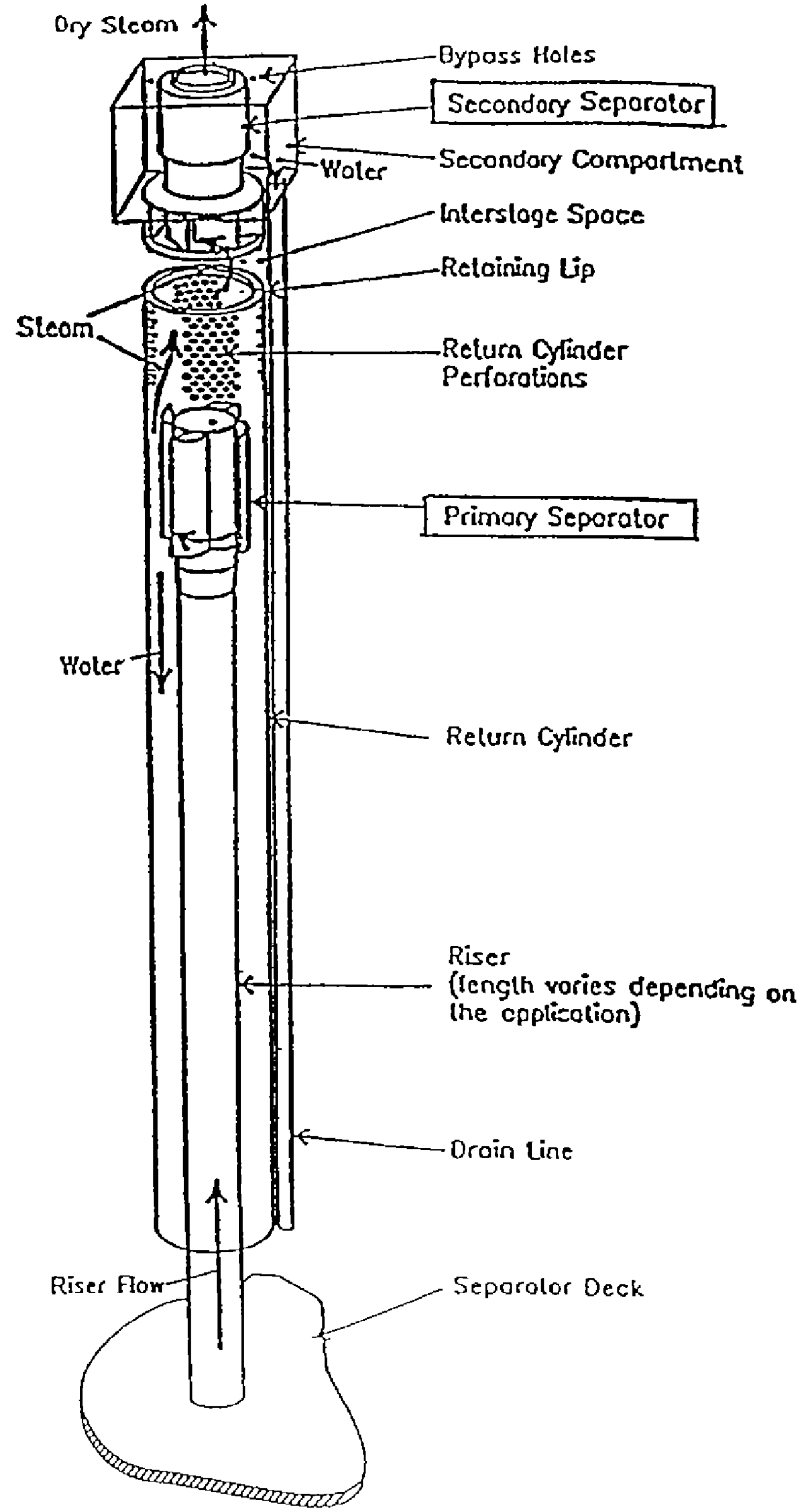
AMERICAN ELECTRIC POWER **COOK NUCLEAR PLANT** **NUCLEAR GENERATION GROUP** **BRIDGMAN, MICHIGAN**	TITLE **UNIT 1 MODEL 51R REPLACEMENT STEAM GENERATOR, GENERAL CONFIGURATION**
	DWG. NO. FSAR FIG. 4.2 - 4A — SH 1 of 1



16.6	REVISED PER 99-UFSAR-1226
REV. NO.	**DESCRIPTION**

REVISIONS

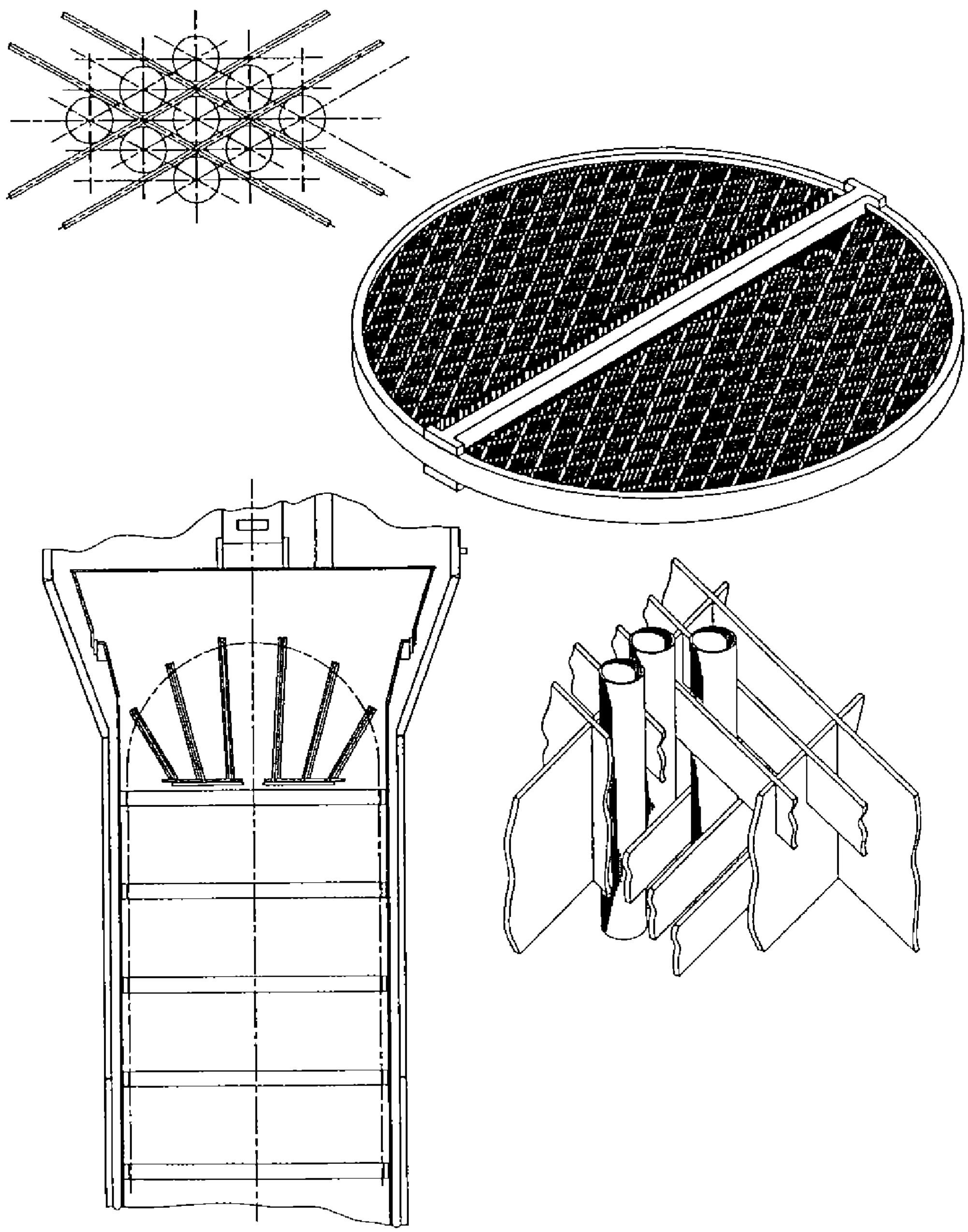
AMERICAN ELECTRIC POWER **COOK NUCLEAR PLANT** **NUCLEAR GENERATION GROUP** **BRIDGMAN, MICHIGAN**	**TITLE** **UNIT 1 MODEL 51R REPLACEMENT STEAM GENERATOR, STEAM DRUM ARRANGEMENT**
	DWG. NO. FSAR FIG. 4.2-4B SH 1 of 1



REV. NO.	DESCRIPTION
16.6	REVISED PER 99-UFSAR-1226

REVISIONS

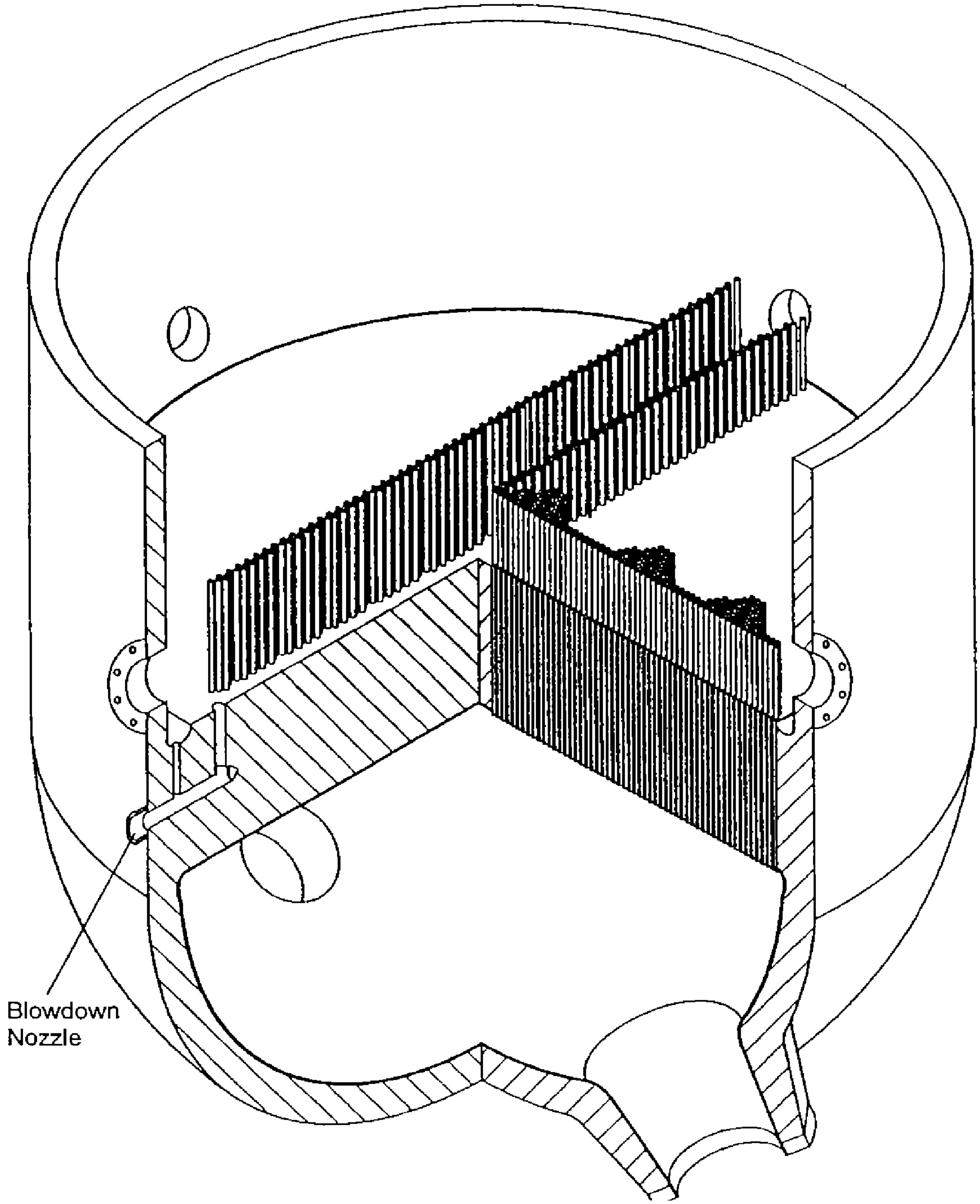
AMERICAN ELECTRIC POWER COOK NUCLEAR PLANT NUCLEAR GENERATION GROUP BRIDGMAN, MICHIGAN	TITLE UNIT 1 MODEL 51R REPLACEMENT STEAM GENERATOR, PRIMARY AND SECONDARY SEPARATORS
	DWG. NO. **FSAR FIG. 4.2-4C** — SH 1 of 1



16.6	REVISED PER 99-UFSAR-1226
REV. NO.	**DESCRIPTION**

REVISIONS

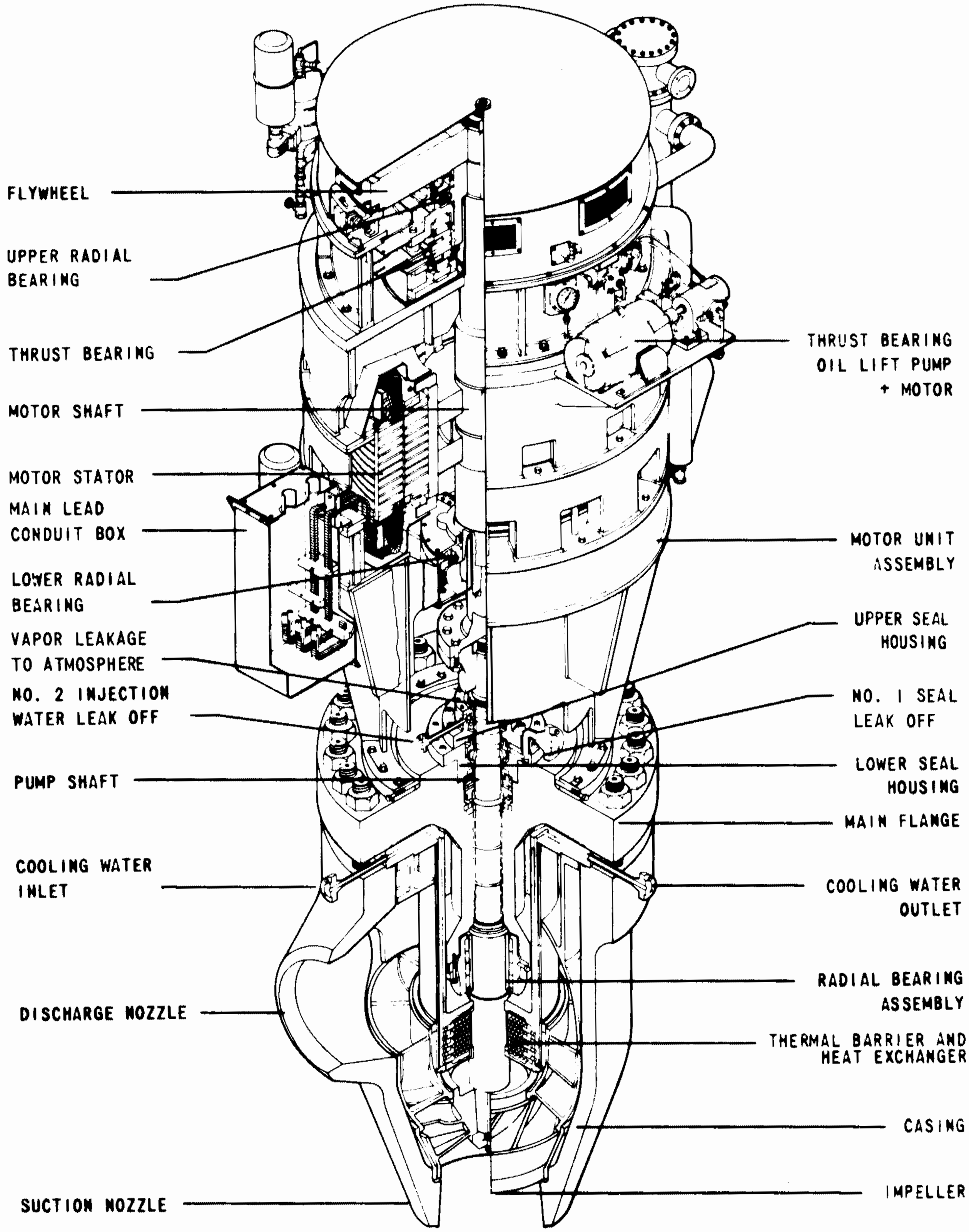
AMERICAN ELECTRIC POWER **COOK NUCLEAR PLANT** **NUCLEAR GENERATION GROUP** **BRIDGMAN, MICHIGAN**	**TITLE** **UNIT 1 MODEL 51R REPLACEMENT STEAM GENERATOR, B&W LATTICE GRID TUBE SUPPORT** **DWG. NO. FSAR FIG. 4.2 - 4D** SH 1 of 1



16.6	REVISED PER 99-UFSAR-1226
REV. NO.	**DESCRIPTION**

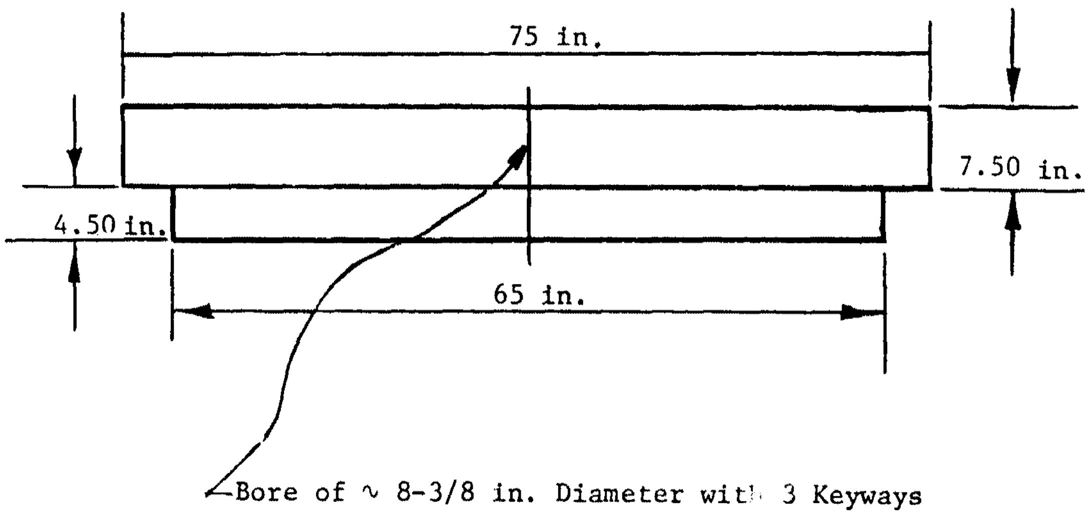
REVISIONS

AMERICAN ELECTRIC POWER **COOK NUCLEAR PLANT** **NUCLEAR GENERATION GROUP** **BRIDGMAN, MICHIGAN**	**TITLE** **UNIT 1 MODEL 51R REPLACEMENT STEAM GENERATOR, TUBESHEET BLOWDOWN CONFIGURATION**	
	DWG. NO. **FSAR FIG. 4.2 - 4E**	SH 1 of 1



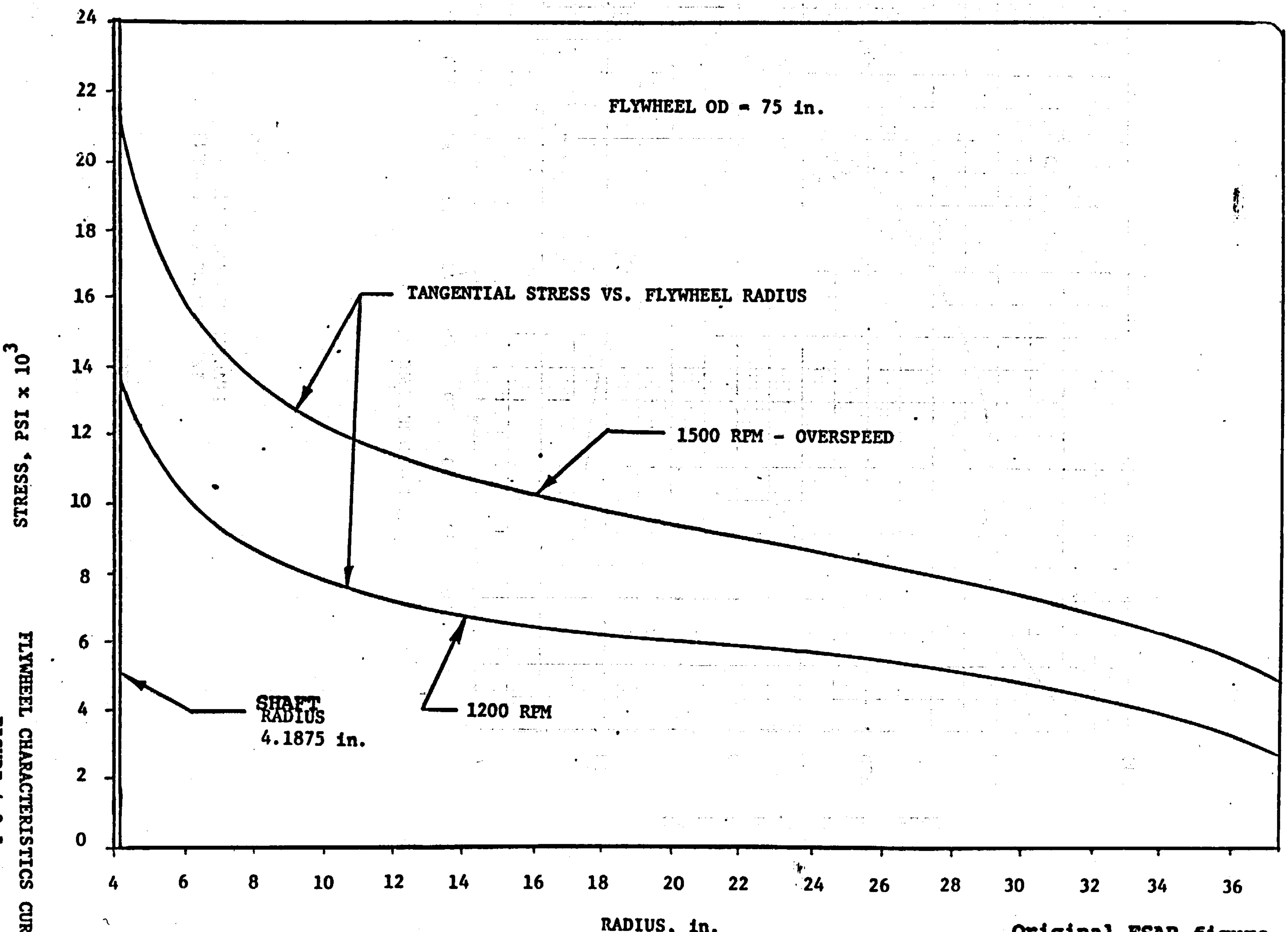
REACTOR COOLANT PUMP
FIGURE 4.2-5

REACTOR COOLANT PUMP FLYWHEEL



NOTE: The plates are bolted together with the bolts aligned perpendicular to the planes of the plates.

FIGURE 4.2-6

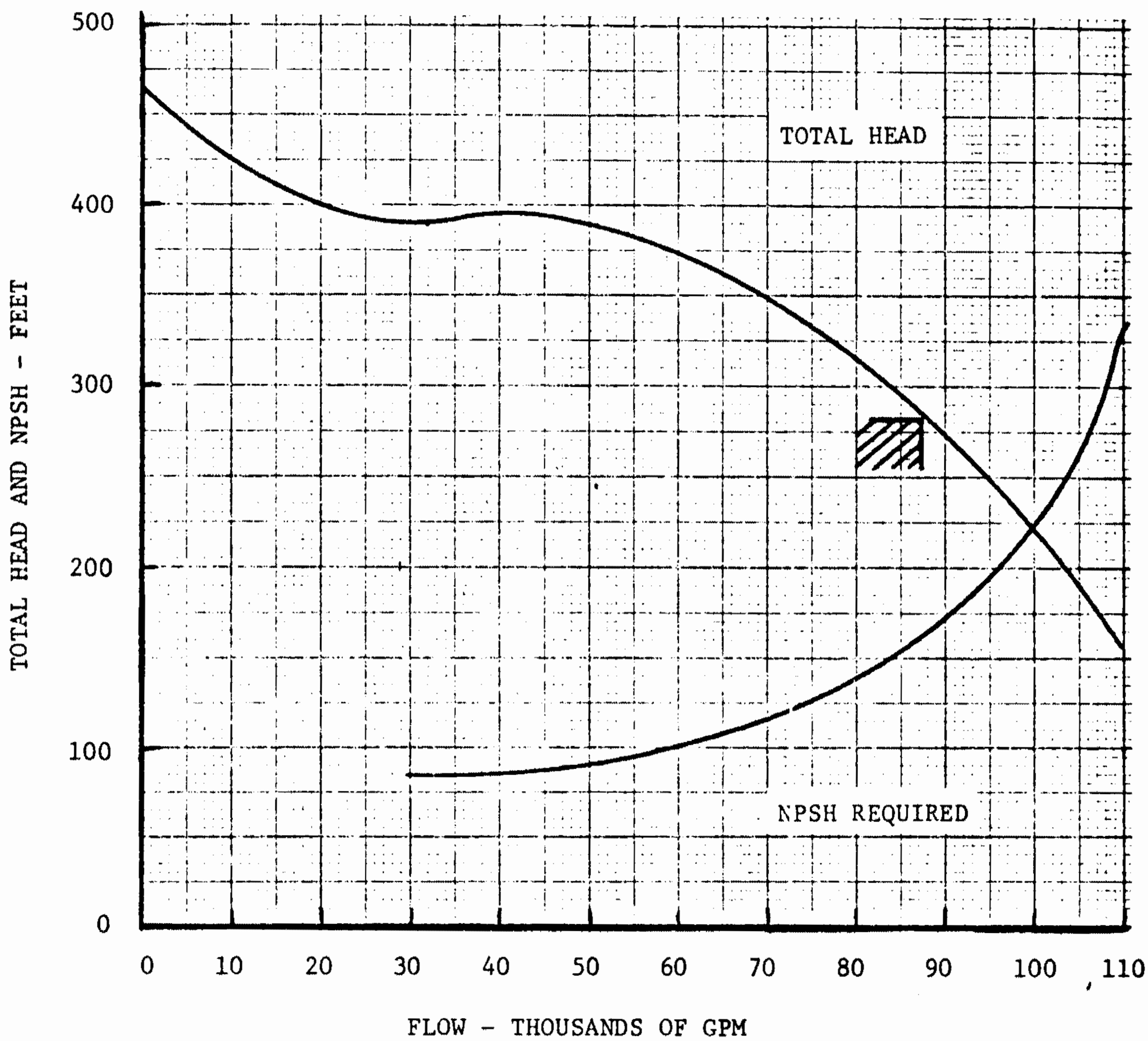


Original FSAR figure

FLYWHEEL CHARACTERISTICS CURVE

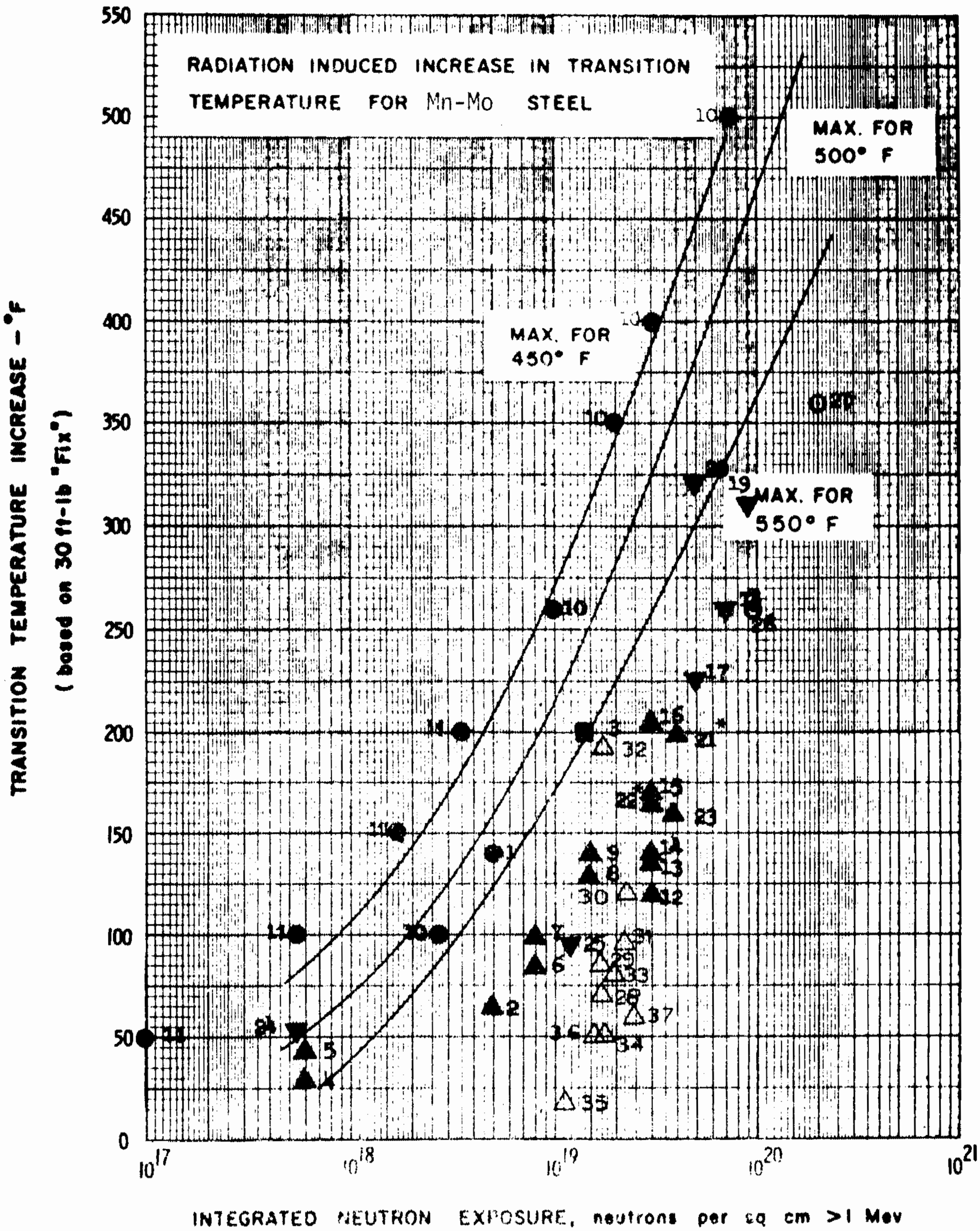
FIGURE 4.2-7

July, 1982



REACTOR COOLANT PUMP
PERFORMANCE CHARACTERISTICS

FIGURE 4.2-8



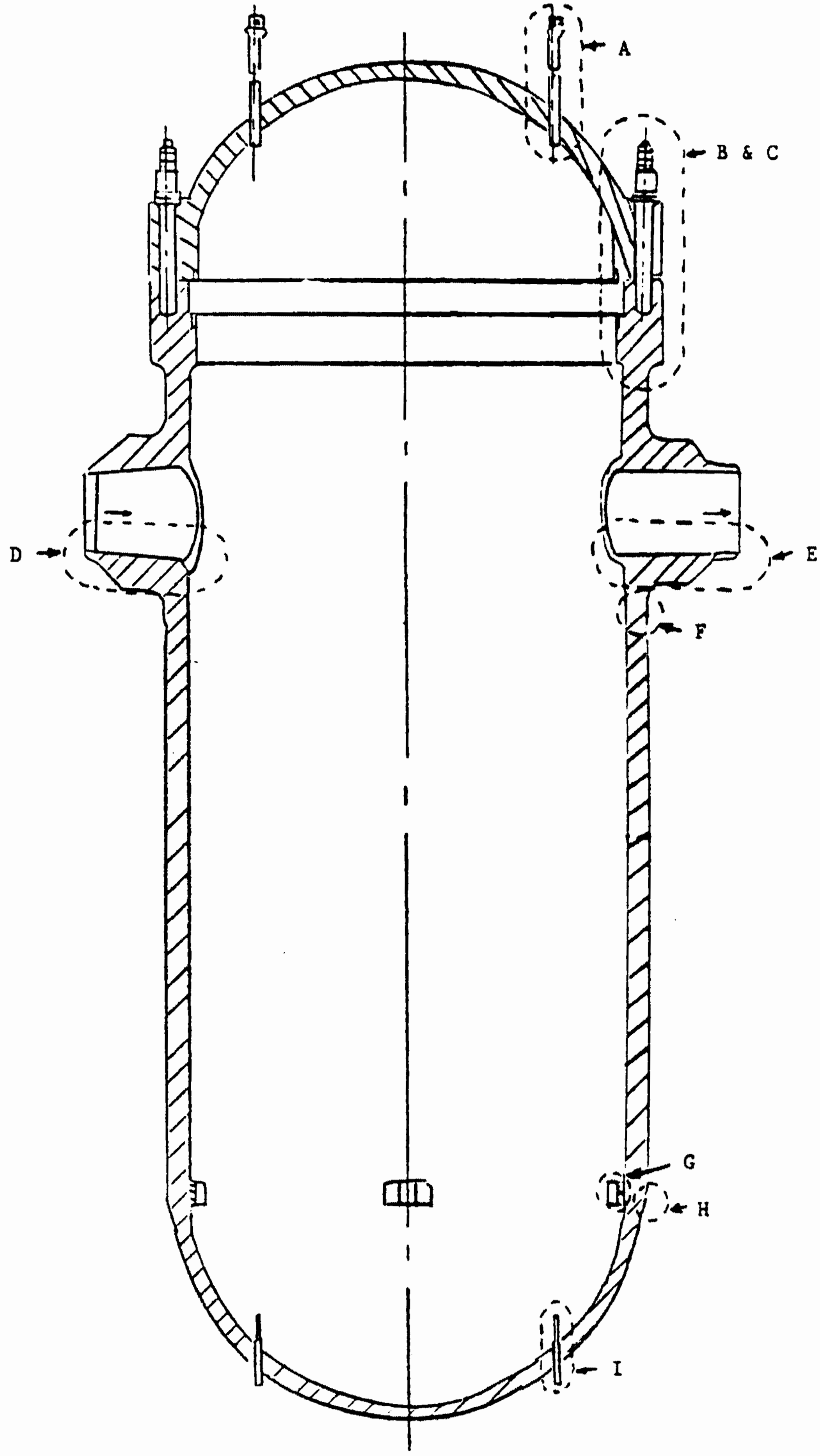
Code	Temp., °F
●	450
■	490
▲	550
△	550 A533B, Class 1
▼	475 to 540
⊙	600 A302B

Numbers 1 through 37 (SEE ATTACHED SHEETS)

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE FOR Mn-Mo STEEL

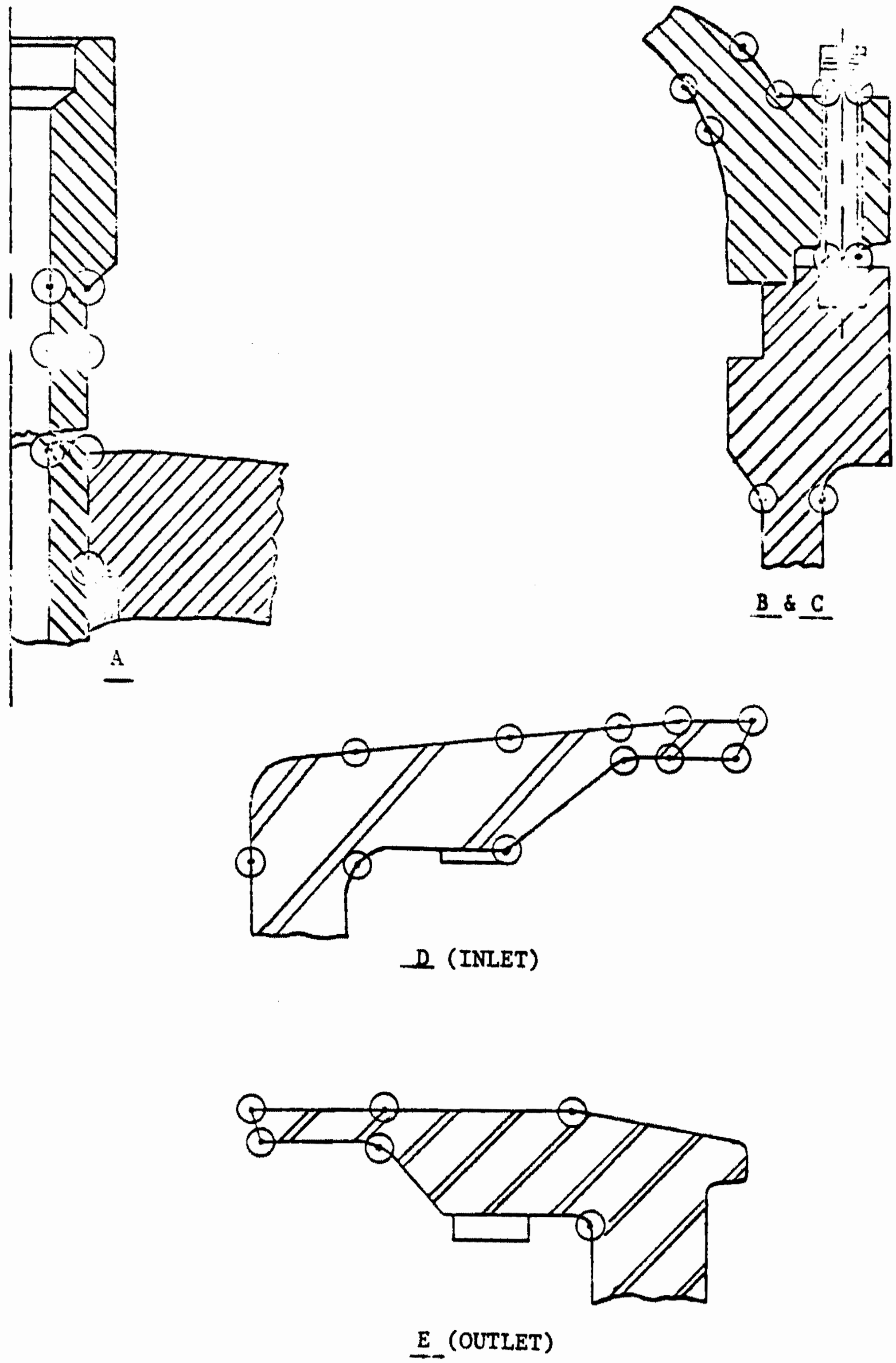
FIGURE 4.2-9

July, 1982



Unit 1 Reactor Vessel Stress Analysis: Areas Examined

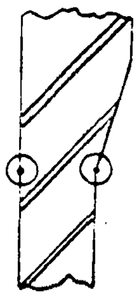
FIGURE 4.3-1 July, 1982



Unit 1 Reactor Vessel Stress Analysis:

Details - Upper

FIGURE 4.3-2
July, 1982



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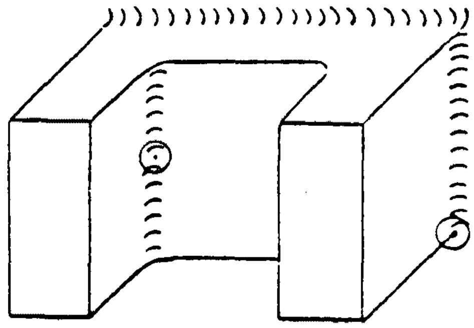

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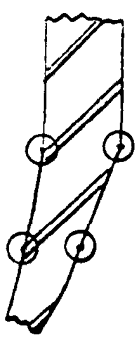

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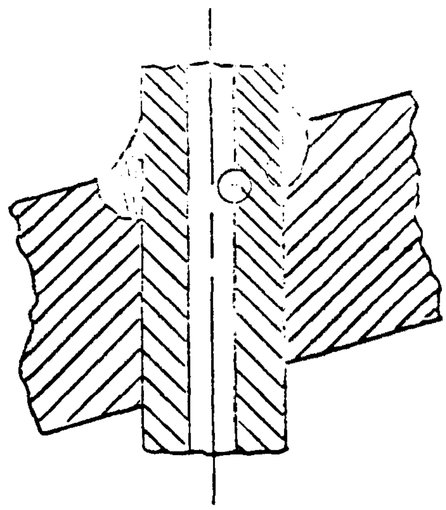


I

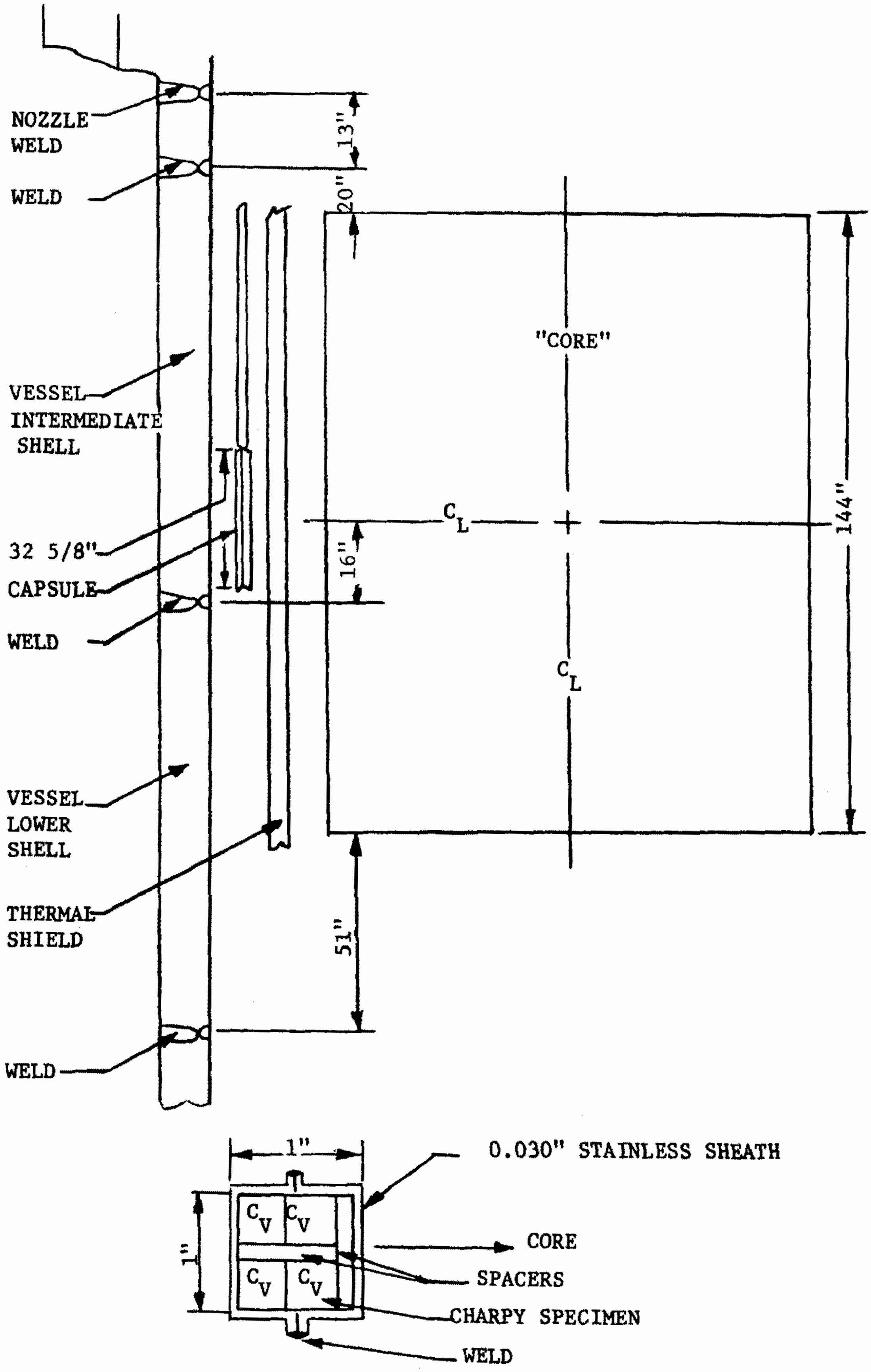
NOTE:

THE POINTS CIRCLED IN THE SKETCHES REPRESENT THE GENERAL LOCATION AND GEOMETRY OF THE AREAS OF DISCONTINUITY AND/OR STRESS CONCENTRATION.

Unit 1 Reactor Vessel Stress Analysis: Details - Lower

FIGURE 4.3-3

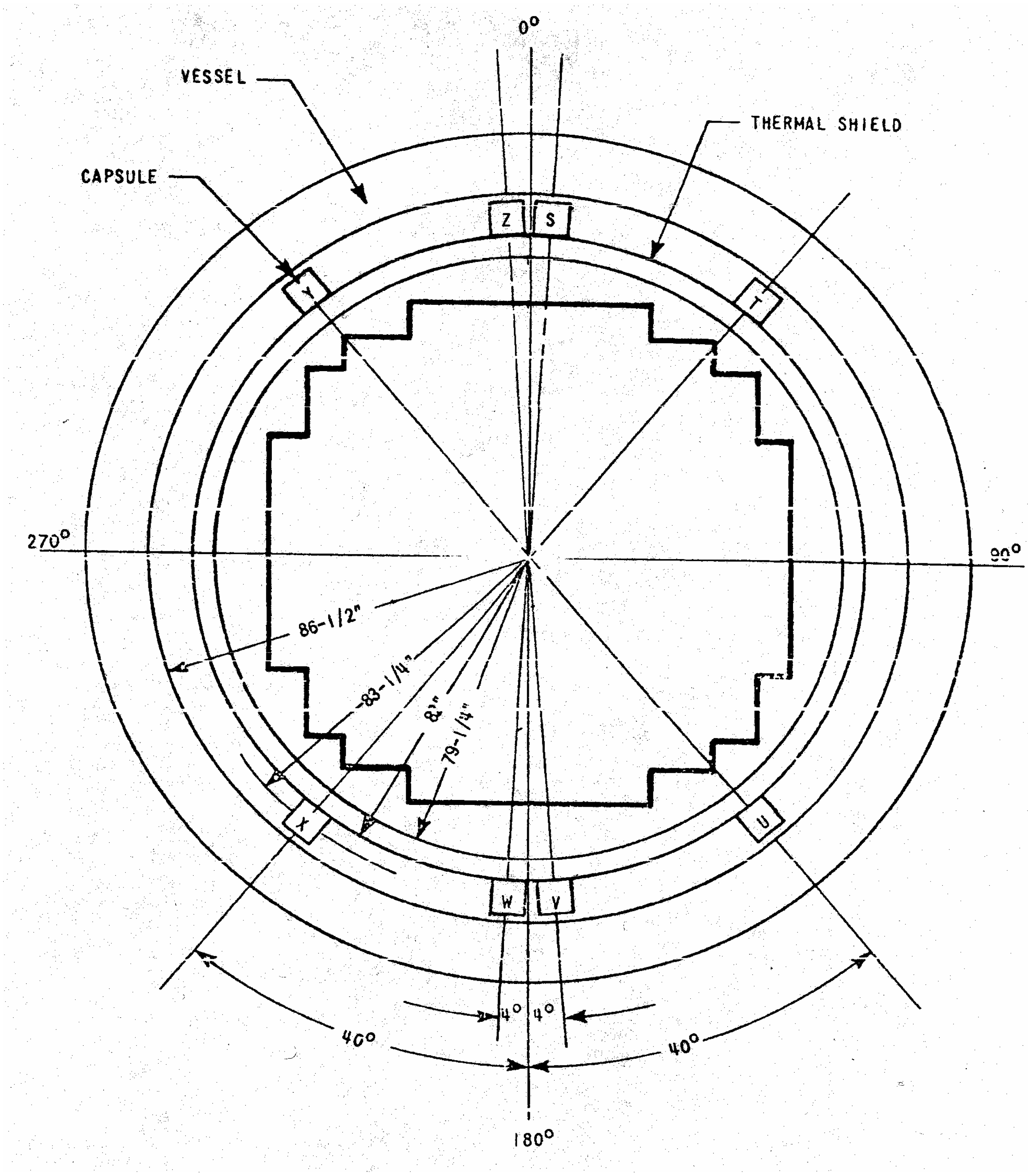
July, 1982



SURVEILLANCE CAPSULE
ELEVATION VIEW

July, 1982

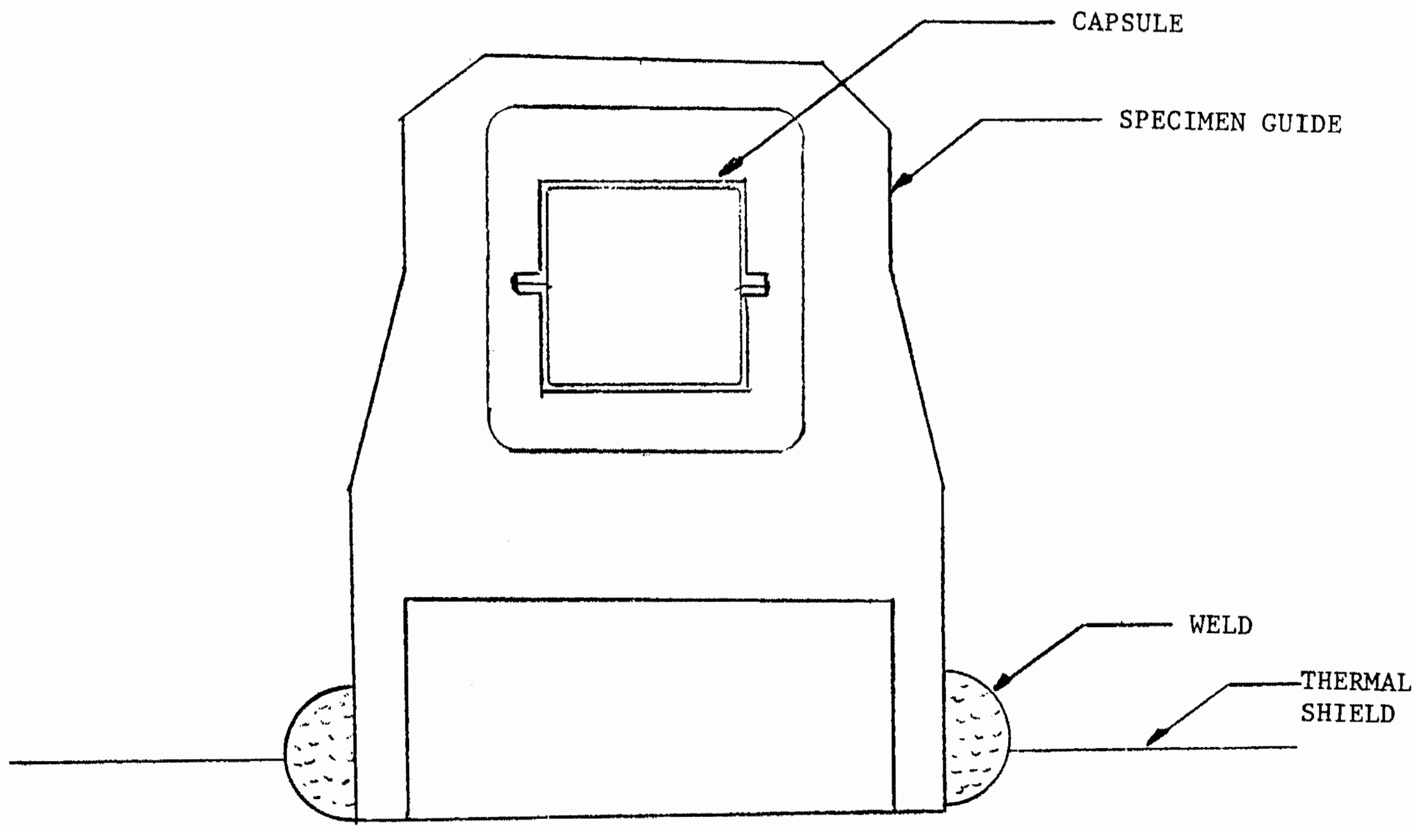
Figure 4.5-1



Notes: Capsule "W" was moved back to the 4 degree location and renamed as "S" in March 2010 (U1C23).
Capsule "S" was renamed as "W" at the 184 degree location.
Figure 4.5-2 is current as of March 2010. Figure 4.5-2A is superseded.

Revision: **23**	Change Description: **UCR-1968**

AMERICAN ELECTRIC POWER COOK NUCLEAR PLANT NUCLEAR GENERATION GROUP BRIDGMAN, MICHIGAN	Title: **Surveillance Capsule Plan View**	
	UFSAR Figure: **4.5-2**	Sheet 1 of 1



SPECIMEN GUIDE TO THERMAL SHIELD ATTACHMENT

Figure 4,5-3

July, 1982