# SECTION 4

# REACTOR COOLANT SYSTEM

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# SECTION 4

# REACTOR COOLANT SYSTEM

The reactor coolant system (RCS), shown in Figure 4.1-1, consists of three similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump, two isolation valves, and a steam generator. The system also includes a pressurizer, connecting piping, pressurizer safety and relief valves, and pressurizer relief tank, which are necessary for operational control. Instrumentation, which is shown on the flow diagram and is part of the RCS, is discussed in Section 7.

## 4.1 DESIGN BASES

### 4.1.1 Performance Objectives

The principal design data of the RCS are given in Table 4.1-1.

The RCS transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator.

The RCS provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values uncontrolled release to the secondary system and to other parts of the unit 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 RCS accommodates coolant volume changes within the protection system limits of the reactor as presented in Section 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 ensures that there is natural circulation capability following a loss-of-flow so as to permit decay heat removal without overheating the core. Part of the systems piping serves as part of the emergency core cooling system (ECCS) to deliver cooling water to the core during a loss-of-coolant accident (LOCA).

### 4.1.2 Design Criteria

Design criteria which apply to the RCS are given below.

Quality Standards

Those systems and components of the unit which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public are identified and designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they are identified. Where adherence to such codes or standards does not suffice to ensure a quality product in keeping with the safety function, they are supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria are identified. An indication of the applicability of codes, standards, quality assurance programs, test

procedures, and inspection acceptance criteria is provided. Where such items are not covered by applicable codes and standards, a showing of adequacy is provided.

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

Quality standards of 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 programs, test procedures and inspection acceptance levels are given in Section 4.3 and 4.5. Emphasis is placed on ensuring the 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.

Performance Standards

Those systems and components of the unit which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public are designed, fabricated, and erected to performance standards that enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind, or heavy ice. The design bases so established reflects:

1. Appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area.

2. An appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

RCS piping and components containing operating pressure, and supporting structures thereto are designed as Seismic Category I. Details are given in Section 4.1.3. The RCS is located in the containment structure whose design, in addition to being a Seismic Category I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Section 5.

Code records are maintained by Westinghouse or their vendors for the mandatory period, and thereafter either by Westinghouse or the licensee.

Records Requirements

The reactor licensee is responsible for ensuring the maintenance throughout the life of the reactor records of the design, fabrication, and construction of major components of the station essential to avoid undue risk to the health and safety of the public.

Records that should be maintained may or may not be under the physical control of the licensee. The licensee ensures that those records which are important, in that they have sole bearing on the health and safety of the public, are maintained.

Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary is designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

The RCS in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of unit operation or anticipated system interactions and maintain the stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure retaining boundary of the RCS is carried out in strict accordance with the applicable codes. In addition there are areas where equipment specifications for RCS components go beyond the applicable codes. Details are given in Section 4.5.

The construction materials of the pressure retaining boundary of the RCS are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime, as discussed in Section 9.1.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions so that continued safe operation is possible, as discussed in Section 7.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Sections of the system which can be isolated are provided with overpressure relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

Monitoring Reactor Coolant Leakage

Means are provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.

Positive indications in the main control room of leakage of coolant from the RCS to the containment are provided by equipment which permits continuous monitoring of containment air activity and humidity, and runoff from the condensate collecting pans under the cooling coils of the containment air recirculation stations. This equipment provides indication of normal background which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, containment air recirculation coolers heat load, condensate runoff and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

Further details are supplied in Section 4.2.7.

Reactor Coolant Pressure Boundary Capability

The reactor coolant pressure boundary is capable of accommodating, without rupture, the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release is taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

The reactor coolant boundary is shown to be capable of accommodating, without further rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Section 14.

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

By using the flexibility in the selection of control rod groupings, radial locations and positions as a function of load, the design limits the maximum fuel temperature associated with highest worth ejected rod to a value which precludes any resultant damage to the RCS 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 is evaluated as a theoretical, though not credible accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the RCS and the containment structure. The environmental consequences of rod ejection are less severe than from the hypothetical loss of coolant, for which public health and safety are shown to be adequately protected. Reference is made to Section 14.

Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

The reactor coolant pressure boundary is designed and operated to reduce to an acceptable level the probability of a rapidly propagating type failure. Consideration is given to the following:

1. Provisions for control over service temperature and irradiation effects which may require operational restrictions.

2. Design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation.

3. Design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The reactor vessel design bases are as follows:

1. PROTECTION AGAINST NON-DUCTILE FAILURE: Assurance of adequate fracture toughness in the reactor vessel material is provided by compliance with the requirements for fracture toughness testing set forth in 10 CFR 50.60.

   The replacement reactor vessel closure head met the 1989 issue of the ASME Pressure Vessel and Boiler Code, Section III for these requirements.

   Assurance that the fracture toughness properties remain adequate throughout the service life of the unit is provided by an irradiation surveillance program conforming to the requirements of 10 CFR 50.60 and 10 CFR 50.61.

   In accordance with that program, specimens are irradiated in capsules located near the core midheight and are removed from the vessel at specified intervals for testing and evaluation in accordance with 10 CFR 50 Appendix H. Based on the results of these tests, revised safe operating, heatup, and cooldown limits are established for the next inspection interval in accordance with 10 CFR 50 Appendix G.

   The details of these limits, including the criteria used in their development may be found in the Licensing Requirements Manual, Section 5.2, "Pressure and Temperature Limits Report."

2. CODES AND SPECIFICATIONS: Design and fabrication of the reactor vessel is carried out in strict accordance with ASME Section III.

3. DESIGN TRANSIENTS: Cyclic loads are introduced by normal power changes, reactor trip, startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected unit life. Vessel analyses result in a usage factor that is less than 1.

   With regard to the thermal and pressure transients involved in the LOCA, the reactor vessel is analyzed to confirm that the delivery of cold emergency core cooling water to the vessel following a LOCA does not cause a loss of integrity of the vessel.

   The design specifications require analysis to prove that the vessel is in compliance with the fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heating and cooling rate limits are as described in the Licensing Requirements Manual. These rates are reflected in the vessel design specifications.

4. INSPECTION: The internal surface of the reactor vessel is capable of inspection periodically using visual and/or nondestructive techniques over the accessible areas. During refueling, the vessel cladding is capable of being inspected in certain areas between the closure flange and the primary coolant inlet nozzles, and if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

   The closure studs can be inspected periodically using visual, magnetic particle, and/or ultrasonic techniques.

All pressure containing components of the RCS are designed, fabricated, inspected, and tested in conformance with the applicable codes. Further details are given in Section 4.1.6 and 4.5.

Reactor Coolant Pressure Boundary Surveillance

Reactor coolant pressure boundary components have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes is provided.

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surface of the reactor coolant piping, except for the area of pipe passing through the primary shielding concrete. Further details are given in Section 4.5.

### 4.1.3 Design Characteristics

Design data for the respective RCS components are listed in Tables 4.1-3, 4.1-4, 4.1-5, 4.1-6, 4.1-7 and 4.1-8 and Table 4.1-12.

Design Pressure

The RCS design and operating pressures together with the safety, power relief and pressurizer spray valves setpoints, and the protective system setpoint pressures are listed in Table 4.1-2. The selected design margin includes operating transient pressure change 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 RCS components. Relief valves are shown on Figure 4.1-1. Boundaries with the ECCS and auxiliary systems are discussed in Sections 6 and 9.

Design Temperature

The design temperature for each component is 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, 4.1-4, 4.1-5, 4.1-6, 4.1-7, 4.1-8 and 4.1-12.

Seismic Loads

The seismic loading conditions are established by the Operating Basis Earthquake (OBE) and Design Basis Earthquake (DBE). The former is selected to be typical of the largest probable ground motion based on the site seismic history. The latter is selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties. Refer to Section 2.5.

For the OBE loading condition, the RCS 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 ensures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the RCS 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 seismic analyses for equipment are defined in Appendix B.

Design and construction practices in accordance with these criteria ensure the integrity of the RCS under seismic loading.

An integrated dynamic analysis of the reactor coolant piping, the NSSS equipment and NSSS equipment supports has been 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 the secondary piping were considered. Time history forcing functions associated with the various breaks were used as input to the integrated dynamic model.

Westinghouse pressurized water reactors are designed to withstand the thermal-mechanical effects caused by loss of coolant accidents including the rupture of any Reactor Coolant System pipe of size up to and including the hypothetical "double-ended" rupture of its largest Reactor Coolant System piping. The reactor core and internals together with the Emergency Core Cooling System (ECCS) are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core is preserved following the accident. Pertinent information on the core and Internals Integrity Analysis can be found in Section 14.3.3 and References 1 and 2.

A detailed description of the dynamic system analysis methods, procedures, and acceptance criteria can be found in Reference 3. In this reference the structural integrity of the internals are analyzed and discussed for the combined dynamic effects of the Loss of Coolant Accident and the Design Basis Earthquake.

The equipment supports have been designed to accommodate the loads resulting from secondary piping LOCAs.

Results from these analyses were combined with operating conditions to obtain the resultant state of stress in both the piping system and the support system. These results showed that the reactor coolant support system and reactor coolant piping system will both be within an acceptable state of stress for all loading conditions.

The types of analyses used to analyze Seismic Category I systems and components are described in Appendix B.2 and B.3 for Stone & Webster and Westinghouse components, respectively. The use of elastic system analysis with both elastic and inelastic subsystem analysis for faulted conditions is permitted by the ASME Code in Appendix F and the draft of Subsection NG for reactor vessel internals. These methods of analysis are standard procedures and the applicability and/or limitations have been discussed and approved by the cognizant ASME working groups developing Appendix F and Subsection NG.

For the analysis of the reactor vessel internals, the effects of plastic deformation on the dynamic system analysis were evaluated. From the evaluation, it was concluded that the effects are negligible.

There are no reactor coolant pressure boundary components whose design will be based on experimental stress analysis (Appendix II of the ASME Code, Section III).

### 4.1.4 Cyclic Loads

The RCS and its components are designed to accommodate without reactor trip 10 percent of full power step changes in unit 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 power. The RCS can accept a complete loss of load from full power with reactor trip.

All components in the RCS are 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 transients used for design purposes and their bases are given in Table 4.1-10. During reactor startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.3.

The pressurizer surge line has been qualified for the cyclical effects of thermal stratification in conjunction with the design transients. This qualification was originally performed as a result of NRC Bulletin No. 88-11 (Reference 5).

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are 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 to be considered for equipment fatigue analyses are based upon engineering judgment and operating experience. Those transients are chosen which are representative of transients which prudently should be considered to occur during unit operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior.

It is difficult to discuss in absolute terms the transients that the unit will actually experience during the operating life. For clarity, however, each transient condition is discussed below in order to make clear the nature and basis for the various transients.

The five following transients are considered Normal Conditions:

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 hr 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 the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hr will not be usually 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 heating and cooling rate limits are as described in the Licensing Requirements Manual. Ideally, heatup and cooldown would occur only before and after refueling. In practice, additional unscheduled unit cooldowns may be necessary for plant maintenance. The frequency of maintenance shutdowns is expected to decrease as the unit matures.

2. UNIT LOADING AND UNLOADING: The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5 percent per minute between 0 percent load and full load. This load swing is the maximum possible consistent with operation of the reactor control system. The reactor coolant temperature varies with load as prescribed by the temperature control system.

3. STEP INCREASE AND DECREASE OF 10 PERCENT: The $\pm 10$ percent 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 unit output is tied. The reactor control system is designed to restore unit equilibrium without reactor trip following a $\pm 10$ percent step change in turbine load demand initiated from unit equilibrium conditions in the range between 15 percent and 100 percent full load. In effect, during load change conditions, the reactor control system or the operator attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized, and reactor coolant temperature is restored to its programmed setpoint 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 RCS 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 is ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure is also decreased from its peak pressure value and follows 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 reactor 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 operator may withdraw 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. Alternatively an operator may decrease turbine load as described by administrative controls.

   The reactor coolant average temperature is raised to a value above its initial equilibrium value at the beginning of the transient.

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 automatically initiates a secondary side steam dump system that avoids a reactor shutdown or lifting of steam generator safety valves. The unit is capable of accepting a 50 percent load rejection from full load without reactor trip. The steam dump system is capable of accepting 40 percent of full load steam flow at full load steam pressure. The remaining 10 percent of the total step change is assumed by the reactor rod control system as noted in Section 3. If a steam dump system were not provided to cope with this transient, there would be a large mismatch between the turbine demand and the reactor output which could cause reactor trip and lifting of steam generator safety valves.

5. STEADY STATE FLUCTUATIONS: The reactor coolant average temperature, for purposes for design, is assumed to increase or decrease a maximum of 6°F in 1 minute. The temperature changes are assumed to be around the programmed value of Tavg (Tavg ±3°F). The corresponding reactor coolant average pressure is assumed to vary accordingly (2250 ±25 psi).

The seven following transients are considered Upset Conditions:

1. LOSS OF LOAD WITHOUT IMMEDIATE TURBINE OR REACTOR TRIP: 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 RCS. The reactor and turbine eventually trip as a consequence of a high pressurizer level trip initiated by the reactor trip system. Since redundant means of tripping the reactor are provided as part of the reactor trip system, transients of this nature are not expected but are included to ensure a conservative design.

2. LOSS OF POWER: This transient applies to a blackout situation involving the loss of outside electrical power to the station and a reactor and turbine trip. 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 emergency diesel generator power. Steam is removed by the secondary system for reactor cooldown through the atmospheric relief valves.

3. 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 are a reactor and turbine trip from low reactor coolant flow which is then 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.

4. REACTOR TRIP FROM FULL POWER: A reactor trip from full power may occur from a variety of causes which result in temperature and pressure transients in the RCS 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 trip system causes the control rods to move into the core.

5. INADVERTENT AUXILIARY SPRAY: The inadvertent pressurizer auxiliary spray transient will occur if the auxiliary spray valve is opened inadvertently during normal operation of the unit. This will introduce cold water into the pressurizer with a very sharp pressure decrease as a result.

   The temperature of the auxiliary spray water is dependent upon the performance of the regenerative heat exchanger. The most conservative case is when the letdown stream is shut off and the charging fluid enters the pressurizer unheated. Therefore, for design purposes, the temperature of the spray water is assumed to be 100°F. The spray flow rate is assumed to be 200 gpm. It is further assumed that the auxiliary spray will, if actuated, continue for five minutes until it is shut off.

   The pressure decreases rapidly to the low pressure reactor trip point. At this pressure, the pressurizer low pressure reactor trip is assumed to be actuated; this accentuates the pressure decrease until the pressure is finally limited to the hot leg saturation pressure. In 5 minutes, the spray is stopped, and all the pressurizer heaters return the pressure to 2250 psia.

   For design purposes, it is assumed that no temperature changes in the RCS will occur as a result of initiation of auxiliary spray except in the pressurizer.

6. OPERATIONAL BASIS EARTHQUAKE: The mechanical stress transients resulting from the OBE are considered on a component basis. Fatigue analysis, where required by the codes, is performed by the supplier as part of the stress analysis report. The earthquake transients are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis as an integral part of the component load histogram used for fatigue evaluation.

7. RCS COLD OVERPRESSURIZATION: RCS Cold Overpressurization may occur during startup and shutdown conditions at low temperature, with or without existence of a steam bubble in the pressurizer. The likelihood of a Cold Overpressurization event is highest when the reactor coolant system is in a water-solid configuration. Such an event can result from any of a number of malfunctions or operator errors. All Cold Overpressurization events experienced thus far may be characterized as having resulted from the addition of mass (mass input transient) or the addition of heat (heat input transient). These types of transients are represented by composite “umbrella” design transients, referred to herein as RCS Cold Overpressurization.

The following transient is considered an emergency condition:

ACTIVATION OF THE PRESSURIZER CODE SAFETY VALVES: Although certain nuclear steam supply systems design transients (for example, loss of load) which are classified as upset conditions may actuate the safety valves, the extremely low number of actual safety valve actuations in operating pressurized water reactors justifies the emergency condition from the ASME design philosophy and a stress analysis viewpoint. However, if actuations of safety valves occur, the subject piping will be inspected for loss of integrity. This condition only applies to the pressurizer safety and relief valve piping.

The following four transients are considered Faulted Conditions:

1. RCS BOUNDARY PIPE BREAK: This postulated break was eliminated by the application of "leak-before-break" technology for excluding from the design basis the dynamic effects of postulated pipe ruptures in primary coolant piping as allowed by GDC-4 (dated December 17, 1986).

   In addition, postulated breaks for the pressurizer surge line have been eliminated by application of "leak-before-break" technology.

   The design of supports and restraints for postulated ruptures are presented in Section 5.2.

   Protection criteria against dynamic effects associated with pipe breaks is covered in Section 5.2.6.

2. STEAM LINE BREAK: For component evaluation, the following conservative conditions are considered.

   a. The reactor is initially in hot, zero power subcritical condition assuming all rods in except the most reactive rod which is assumed to be stuck in its fully withdrawn position.

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

   c. Subsequent to the break the reactor coolant temperature cools down to 212°F.

   d. The ECCS pumps restore the reactor coolant pressure.

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

   The dynamic reaction forces associated with circumferential steam line breaks were considered in the design of supports and restraints in order to ensure continued integrity of vital components and engineered safety features. Protection criteria against dynamic effects associated with pipe breaks is covered in Section 5.2.6.

3. STEAM GENERATOR TUBE RUPTURE: This accident postulates the double ended rupture of a steam generator tube resulting in a decrease in pressurizer level and RCS pressure. Reactor trip occurs 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 sufficiently, a high level alarm occurs and the feedwater regulating valves and feedwater isolation valves will close. At this time, the planned procedure for recovery from this accident calls for isolation of the steam line leading from the affected steam generator. Therefore, this accident results in a transient which is no more severe than that associated with a reactor trip. For this reason, a Steam Generator Tube Rupture event requires no special treatment insofar as fatigue evaluation is concerned.

4. DESIGN BASIS EARTHQUAKE: The mechanical stress transient resulting from the DBE is considered on a component basis.

   The design transients and the number of cycles of each that is normally used for fatigue evaluations are shown in Table 4.1-10. Faulted conditions are not included in fatigue evaluations.

Supplemental Transients

The following supplemental transients are part of the analytical basis for components with detailed fatigue analysis. Monitoring of the occurrences of these transients, except as noted below, is required to confirm the analytical assumptions, consistent with Technical Specification 5.5.3.

1. Pressurizer Insurge

   Surge line flow from the hot leg to the pressurizer can cause rapid cooling of the pressurizer nozzle and lower shell. This transient is most severe during plant startup and shutdown operation since there is a larger temperature difference ($\Delta T$) from the pressurizer liquid space to the hot leg temperature. Detailed analysis using various increments of $\Delta T$ has shown acceptable results. The surge line is monitored for temperature changes caused by insurges.

2. Selected CVCS Transients

   The following chemical and volume control system (CVCS) transients affect specific segments of piping and have been incorporated in their respective piping analyses. Continued monitoring of these transients is not required.

   a) Isolation of Charging Flow

      Isolation of charging flow has the effect of allowing the charging piping immediately adjacent to the RCS loop to heat towards RCS temperatures. On the restoration of charging flow, the piping would rapidly cool, causing a thermal transient.

   b) Isolation of Letdown Flow

      Isolation of letdown flow would allow the temperature of the letdown piping, outside of the zone heated by RCS loop flow, to slowly decay towards ambient. When flow is resumed, the pipe would be rapidly heated. Cooldown and reheat is considered a single cycle.

      Isolation of letdown flow could also affect the charging piping, given sufficient time, if the bulk temperature in the regenerative heat exchanger dropped so that charging flow was not preheated. Since the temperature transient in this piping would be dictated by flow, rather than heat decay, it might be fatigue significant.

   c) Placing Excess Letdown in Service

      Placing excess letdown in service has the effect of a rapid heatup of the excess letdown piping outside the RCS heated zone.

3. Safety Injection Actuation

   This transient was not a part of the original BVPS design basis, but was added because of an NRC Safety Evaluation Report. Injection of relatively cold SI fluid into the hot RCS would cause rapid cooling of the SI nozzle. This transient is monitored to ensure the administrative limit on the number of its occurrences is not exceeded without additional evaluation. See reference 7.

4. Auxiliary Feedwater Injections

   These transients can occur when the plant is at hot standby or no-load conditions. It is assumed that the low steam generation rate is made up by intermittent (slug) feeding of auxiliary feedwater into the steam generator.

   Feedwater additions required during plant heatup and cooldown operations are also assumed to be addressed by the Auxiliary Feedwater Injections transient.

5. Residual Heat Removal (RHR) System Actuation

   As the RCS is cooled down and reaches approximately 350˚F, the RHR system is placed in service. This introduces hot RCS fluid into the piping that was previously at ambient conditions (70˚F). The RHR piping temperature then follows RCS temperature as the plant is further cooled. There is no severe thermal transient when the RHR system is placed back in service following refueling.

Prior to unit startup, the following tests are carried out:

1. TURBINE ROLL TEST: This transient is imposed upon the unit during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature and the steam generated is used to perform a turbine roll test.

2. HYDROSTATIC TEST CONDITIONS: The pressure tests are outlined below:

   a. Primary Side Hydrostatic Test Before Initial Startup - The pressure tests covered by this section included both shop and field hydrostatic tests which occurred as a result of component or system testing. This hydrostatic test was performed using a separate hydro test pump prior to initial fuel loading at a water temperature which was compatible with the reactor vessel material design transition temperature requirements and a maximum test pressure of 3,107 psig or 1.25 times the design pressure. In this test, the primary side of the steam generator was pressurized to 3,107 psig coincident with the secondary side pressure of 0 psig.

   b. Secondary Side Hydrostatic Test Before Initial Startup - The secondary side of the steam generator was pressurized to 1.25 times the design pressure of the secondary side coincident with the primary side at 0 psig.

   c. Primary Side Leak Test - Subsequent to each time the primary system has been opened, a leak test is performed. During this test, the primary system pressure is for design purposes, assumed to be raised to 2500 psia, with the system temperature above the design transition temperature, while the system is checked for leaks.

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

During this leak test, the secondary side of the steam generator will be pressurized so that the pressure differential across the tubesheet does not exceed 1600 psi. This is accomplished by closing off the steam lines.

Since the tests outlined under item a. and b. occur prior to unit startup, the number of cycles is independent of unit life.

The system hydrostatic test at 1.25 times design pressure, the post-operational hydrostatic tests at 2485 psig, and the post-operational leak tests at 2235 psig shall be made at temperatures not lower than the Reference Temperature Nil-Ductility Temperature, RT(NDT) + 60°F. The limiting RT(NDT) was determined in accordance with calculation methods described in Reference 4. The limiting RT(NDT) will increase with fast neutron exposure. Thus, prior to each leak test, the RT(NDT) will be adjusted in accordance with the fluence curve or actual data obtained from the reactor vessel surveillance program.

The BVPS-1 reactor vessel was ordered and designed to the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition. No provisions have been made to update the vessel to the new criteria of the 1972 Summer Addenda. The replacement reactor vessel closure head was ordered and designed to ASME Section III, 1989 Edition.

### 4.1.5 Service Life

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

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

The RT(NDT) shift of the vessel material and welds due to radiation damage effects during service is monitored by a radiation damage surveillance program. Details are given in Section 4.5.

Reactor vessel design is 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 RCS components as required by Section III of the ASME Boiler and Pressure Vessel Code for Class A vessels, the operating conditions are established for the 40 yr design life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

For operation beyond the original 40-year service life, the 40-year design cycles were evaluated against 60-year projected operational cycles and were determined to be bounding for the period of extended operation, except in certain specific cases described in the License Renewal UFSAR Supplement (Managing the Effects of Component Aging, Sections 16.2.3.1 and 16.2.3.3). The Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section 16.1.27) will ensure that components do not exceed their fatigue design bases.

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

Environmental protection is afforded by close adherence to the water chemistry limits set forth in the Licensing Requirements Manual and by the absence of any deleterious conditions in the containment environment, and by pipe and component insulation.

Maintenance standards comply with the applicable codes and standards and with appropriate quality levels. Operating procedures are established and adhered to in accordance with the Beaver Valley Quality Assurance Program.

### 4.1.6 Codes and Classifications

All primary pressure-containing components of the RCS are designed, fabricated, inspected, and tested in conformance with the applicable codes listed in Table 4.1-11 and are Seismic Category I (Appendix B).

The pressurizer surge line has been analyzed to ASME B&PV Code, Section III, 1986 Edition for cyclic fatigue qualification as required by NRC Bulletin No. 88-11. The original qualification analysis is documented in Reference 5.

## References for Section 4.1

1. "Westinghouse PWR Core Behavior Following a Loss of Coolant Accident," WCAP-7422, Westinghouse Electric Corporation (January 1970).

2. L. T. Gesinski, "Fuel Assembly Safety Analysis for Combined Seismic and Loss of Coolant Accident," WCAP-7950, Westinghouse Electric Corporation (July 1972).

3. G. J. Bohm, "Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation," WCAP-7332-L, Westinghouse Electric Corporation (February 1970).

4. W. S. Hazelton, S. L. Anderson, S. E. Yanichko, "Basis for Heatup and Cooldown Limit Curves," WCAP-7924, Westinghouse Electric Corporation (July 1972).

5. "Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line," WCAP-12727, Westinghouse Electric Corporation (November 1990).

6. Deleted by Revision 24.

7. NRC letter from Mr. A. Schwencer (NRC) to Mr. C. N. Dunn (Duquesne Light Co.), "Inadvertent Safety Injection During Cooldown," dated December 28, 1979.

# 4.2 SYSTEM DESIGN AND OPERATION

## 4.2.1 General Description

The RCS consists of three similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, two isolation valves, loop piping and instrumentation. The pressurizer surge line is connected to the hot leg of 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 Figure 4.1-1.

RCS design data are listed in Tables 4.1-1, 4.1-2, 4.1-3, 4.1-4, 4.1-5, 4.1-6, 4.1-7, 4.1-8 and Table 4.1-12.

Pressure in the system is controlled by the pressurizer, 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 Section 7. Spring-loaded steam 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 manufactured by Combustion Engineering, Inc. It is cylindrical with a welded hemispherical bottom head and a removable bolted, flanged and gasketed hemispherical upper head (closure head). The replacement reactor vessel closure head is manufactured by Equipos Nucleares, S.A. The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage may be detected by means of two leakoff communications, one between the inner and outer ring and one outside the outer O-ring. The vessel contains the core, core support structures, control rods, 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 or instrumentation adaptors. The seal arrangement at the upper end of these adaptors consists of a welded flexible canopy seal. Inlet and outlet nozzles are spaced evenly around the vessel. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear incore instrumentation. Each nozzle consists of a tubular member made of Inconel. Each tube is attached to the inside of the bottom head by a partial penetration weld.

Internal surfaces of the vessel which are in contact with primary coolant are weld overlay with 0.125 inches minimum of stainless steel. The exterior of the reactor vessel is insulated with canned stainless steel reflective sheets. The insulation is 3 inches thick and contoured to enclose the sides and bottom of the vessel. Access to vessel side insulation is limited by the neutron shield tank. The reactor vessel closure head is insulated with metal reflective insulation (MRI). The closure head MRI is installed in a stepped arrangement with 5-inch thick sides and a 4-inch thick top layer.

A one-piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. The top of the shield is bolted to the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attenuating much of the gamma radiation and some of the fast neutrons which escape from the core. This shield reduces thermal stresses in the vessel which result from heat generated by the absorption of gamma energy. The shield is described in Section 3.2.3.1.

Surveillance specimens made from reactor vessel steel are located between the reactor vessel wall and the thermal shield. These specimens will be examined at selected intervals to evaluate reactor vessel material RT(NDT) changes as described in Section 4.5.1.

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

A schematic of the reactor vessel is shown in Figure 4.2-1. The materials of construction are given in Tables 1.8-1 and 1.8-3 and the design parameters are given in Table 4.1-3. A description of the reactor vessel internals is given in Section 3.2.3.1.

#### 4.2.2.2 Pressurizer

The pressurizer is a vertical, cylindrical vessel with essentially hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. The surge line nozzle and removable electric heaters are installed in the bottom head. The heaters are removable for maintenance or replacement. A thermal sleeve is provided to minimize stresses in the surge line nozzle. The baffles in the lower section of the pressurizer prevent a cold insurge of water from flowing directly to the steam/water interface.

Spray line nozzles, relief and safety valve connections are located in the top head of the vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves also can be operated manually by a switch in the control room. The pressurizer relief and safety valves are monitored in the control room for positive valve position indication by the use of an acoustic monitoring system. A description of the pressurizer relief and safety valves acoustic monitoring system is provided in Section 7.8.

A small continuous spray flow is provided through a manual bypass valve around the power operated spray valves to ensure that the pressurizer liquid is homogeneous with the coolant and to prevent excessive cooling of the spray piping.

During an outsurge from the pressurizer, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. During an insurge from the RCS the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the power-operated relief valves for normal design transients. Heaters are energized on high water level during insurge to heat the subcooled surge water that enters the pressurizer from the reactor coolant loop.

The volume of the pressurizer is equal to, or greater than, the minimum required 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 100 percent loss of external 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 does not empty following reactor and turbine trip.

6. The safety injection signal is not activated during normal reactor trip and turbine trip.

Pressurizer Support

The skirt type support is attached to the lower head and extends for a full 360 deg around the vessel. The lower part of the skirt terminates in a bolting flange with bolt holes for securing the vessel to its foundation. The skirt type support is provided with ventilation holes around its upper perimeter to ensure free convection of ambient air past the heater plus connector ends for cooling.

The general configuration of the pressurizer is shown in Figure 4.2-2 and the design data are given in Table 4.1-4.

Pressurizer Spray

Two separate, automatically controlled spray valves with remote manual overrides are used to initiate 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 is selected 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 of full load.

The pressurizer spray lines and valves are large enough to provide adequate spray 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 (Section 9.1) 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.

Surge Line

The pressurizer surge line connects the pressurizer to one reactor coolant loop hot leg. The line enables continuous coolant volume pressure adjustments between the RCS and the pressurizer.

The surge line is sized to limit the pressure drop between the RCS and the safety valves with maximum allowable discharge flow from the safety valves. Overpressure of the RCS does not exceed 110 percent of the design pressure.

The surge line and the thermal sleeves at each end are designed to withstand the thermal stresses resulting from volume surges which 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. Discharges from smaller relief valves located inside the containment are also piped to the relief tank. 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 degasifiers via the vent and drain system (Section 9.7), provides a means for removing any noncondensable gases from the RCS 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 disks which discharge into the containment structure. 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 full-power pressurizer water level setpoint. 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 200 F. If the temperature in the tank rises above 120 F during unit operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the vent and drain system.

The spray rate is designed to cool the tank from 200 F to 120 F in approximately 3 hours 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 disks 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. This margin is to prevent deformation of the disks. The tank and rupture disk holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

The pressurizer relief tank level transmitter supplies a signal for an indicator with a high pressure alarm.

The pressurizer relief tank level transmitter supplies a signal for an indicator with high and low level alarms.

The temperature of the water in the pressurizer relief tank is indicated, and an alarm actuated by high temperature informs the operator that cooling of the tank contents is required.

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

<u>Discharge Piping</u>

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

#### 4.2.2.4 Steam Generators

The steam generators are Model 54F replacement steam generators, designed and analyzed to industry codes and standards that are, at a minimum, equivalent to the Model 51 original 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 tubesheet. Manways are provided for access to both sides of the divided head. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The units are primarily low-alloy steel. The heat transfer tubes are Ni-Cr-Fe alloy, the primary side of the tubesheets are clad with Ni-Cr-Fe alloy, and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. Heat transfer tubes may be plugged at the tubesheet where tube degradation is identified. A steam generator of this type is shown in Figure 4.2-3, and design data are given in Table 4.1-5.

The steam generators are designed to produce the steam flow required at full-power operation. The internal moisture-separating equipment is designed to ensure that the moisture carryover does not exceed 0.10 percent by weight, at 100 percent full load steam flow.

#### 4.2.2.5 Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage pump which employs a controlled leakage seal assembly. A view of a reactor coolant pump is shown in Figure 4.2-4 and the principal design parameters for the pumps are listed in Table 4.1-6. The reactor coolant pump performance and the Net Positive Suction Head, NPSH, characteristics are shown in Figure 4.2-7.

The pump consists of three areas from bottom to top. They are the hydraulics, the shaft seals, and the motor.

1. The hydraulic section consists of an impeller, diffuser, casing, thermal barrier, heat exchanger, lower radial bearing, bolting ring, motor stand, and pump shaft.

2. The shaft seal section consists of three primary devices. They are the number 1 controlled leakage, film riding face seal, and the number 2 and number 3 rubbing face seals. These seals are contained within the main flange and seal housing. A fourth sealing device called a shutdown seal is housed within the No. 1 seal area and is passively actuated by high temperature if seal cooling is lost.

3. The motor section consists of a vertical solid shaft, squirrel cage induction type motor, an oil lubricated double Kingsbury type thrust bearing, two oil lubricated radial bearings, and a flywheel.

Attached to the bottom of the pump shaft is the impeller. The reactor coolant is drawn up through the impeller, discharged through passages in the diffuser, and out through the discharge nozzle in the side of the casing. Above the impeller is a thermal barrier heat exchanger which limits heat transfer between hot system water and seal injection water.

High pressure seal injection water is introduced through the thermal barrier wall. A portion of this water flows through the seals; the remainder flows down the shaft through and around the bearing, and the thermal barrier where it acts as a buffer to prevent system water from entering the radial bearing and seal section of the unit. The heat exchanger provides a means of cooling system water to an acceptable level in the event that seal injection flow is lost. The water lubricated journal-type pump bearing, mounted above the thermal barrier heat exchanger, has a self-aligning spherical seat.

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.

Component cooling water is supplied to the motor bearing oil coolers and the thermal barrier cooling coil. The thermal barrier cooling coil ensures the cooling of seal water in the event of loss of injection water.

The motor is an air-cooled, Class F thermalastic epoxy insulated, squirrel cage induction motor. The rotor and stator are of standard construction and are cooled by air. Six resistance temperature detectors are located throughout the stator to sense the winding temperature. (One resistance temperature detector is in service, the remaining five are spares.)

Each of the reactor coolant pumps is equipped with two vibration pickups mounted at the top of the motor support stand. They are mounted in a horizontal plane to pick up radial vibrations of the pump. One is aligned perpendicular; the other is aligned parallel to the pump discharge; signals from all the reactor coolant pumps are sent to a multipoint selector switch mounted outside the reactor containment. The signals may be read on a vibration meter which shows amplitude or frequency of vibration. The maximum allowable vibration level is 20 mils for the pump shaft and 5 mils for the pump frame at running speed (1200 rpm approximate).

All parts of the pump in contact with the reactor coolant are austenitic stainless steel except for seals, bearings, and special parts.

A flywheel on the shaft above the motor provides additional inertia to extend flow coastdown. Each pump contains a ratchet mechanism to prevent reverse rotation. The reactor coolant pump flywheel is shown in Figure 4.2-5.

The pump shaft, seal housing, thermal barrier, bolting ring, and motor stand can be removed from the casing as a unit without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

The most adverse operating condition of the flywheels is visualized to be the loss-of-load situation. The following conservative design-fabrication conditions preclude missile production by the pump flywheels. The wheels are fabricated from rolled, vacuum-degassed, ASTM A-533

or equivalent[8] steel plates. Flywheel blanks are flame-cut from the plate, with allowance for exclusion of flame-affected metal. A minimum of three Charpy tests are made from each plate parallel and normal to the rolling direction; they determine that each blank satisfies design requirements. A Nil Ductility Transition Temperature, NDTT, less than +10°F is specified. The finished machine bores are also subjected to magnetic particle or liquid penetrant examination.

Plate for the flywheel was volumetrically examined in accordance with ASME Section III, Class 1 vessel plate requirements.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 4.2-6) less than 50 percent of the minimum specified material yield strength at room temperature (100 to 150°F).

The pump flywheels were designed, fabricated, and qualified, in accordance with Regulatory Guide 1.14 (10/27/71) which provides for a high level confidence in their integrity, and provides an acceptable basis for satisfying the requirements of Criterion 4, 10 CFR 50, Appendix A. An inservice inspection program is also in place that is in compliance with the intent of Regulatory Guide 1.14 (10/27/71) requirements, but has been relaxed in accordance with the findings documented in Westinghouse Owners Group Report WCAP 14535A[9].

4.2.2.6 Reactor Coolant Piping

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. All smaller piping which comprise part of the RCS boundary, such as the pressurizer surge line, spray and relief line, loop drains and connecting lines to other systems are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. All joints and connections are welded, except for the pressurizer relief and the pressurizer code 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. Both ends of the pressurizer surge line.
3. Pressurizer spray nozzle.

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

All 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 45 degrees down from the horizontal centerline. This enables the water level in the RCS 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 RCS during refueling and maintenance operation.

3. The differential pressure taps for flow measurement which are downstream of the steam generators on the 90 degree elbow.

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

1. The pressurizer spray line 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 Resistance Temperature Detector, RTD, hot leg scoops extend into the reactor coolant. In the original design these scoops collected a representative temperature sample for the RTD manifold. The RTD manifolds have been removed, now the scoops provide a convenient location for narrow range, thermowell mounted RTD's.

4. The wide range temperature detectors are located in RTD wells that extend into the reactor coolant pipes.

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

4.2.2.7 Valves

All valves in the RCS 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.

All original plant RCS valves that are 3 inches and larger, which contain radioactive fluid and which normally operate above 212°F, are provided with double-packed stuffing boxes and stem intermediate lantern gland leakoff connections. New or replacement RCS valves contain stuffing box designs that provide an equivalent level of leak protection to the original valves. All RCS valve leakoff connections within the containment are piped to a header which can discharge to either the pressurizer relief tank (PRT) or No. 1 primary drains tank (PDT). Leakage to the containment is essentially zero for these valves. Reactor coolant chemistry parameters are specified to minimize corrosion. Periodic analyses of coolant chemical composition ensures that the reactor coolant meets these specifications. The upper-limit coolant velocity of about 50 ft per sec precludes accelerated corrosion.

Valve leakage is minimized by design features as discussed above.

<u>Pressurizer Safety Valves</u>

The pressurizer safety valves are pilot operated valves consisting of two principle assemblies - a pilot and main valve. These two assemblies are coupled to provide a unitized, self actuated safety valve 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 2,485 psig.

A water seal is maintained below each safety valve seat to minimize leakage. The 6 inch pipes connecting the pressurizer nozzles to their respective code safety valves, are shaped in the form of a loop seal. Condensate, as a result of normal heat losses to the ambient, accumulates in the loop seal, thus flooding the valve seat. The water prevents any leakage of hydrogen gas or steam through the safety valve seats. This condensate temperature is held to 300°F minimum. This is accomplished by use of an insulated enclosure around the loop piping up to and including the valve inlet flange. The enclosure is attached to the pressurizer insulation with a portion of this insulation removed within the enclosure boundary. Heat from the pressurizer is held within the enclosure thus helping to maintain the loop seal temperature within the limits of 300 to 400°F. With the loop seal so heated, the piping support reactions are minimized due to flashing of the seal water upon activation of the safety relief valves.[5] If the pressurizer pressure exceeds the set pressure of the safety valves, they will start lifting, and the water from the seal will discharge during the accumulation period. Pressure switches on the safety valves and a temperature indicator in the safety valve discharge manifolds alert the operator to the passage of steam due to leakage or valves lifting. These pressure switches monitor the pilot valve chamber for leakage which could lift the safety valve below setpoint.

The pressurizer safety valve support is designed to withstand seismic, thermal and dead weight forces in addition to the valve discharge reactions. The valves are mounted and supported from the pressurizer by means of metal bands around the pressurizer from which the individual support for each valve is attached. Design of the support and the bands is capable of accepting the reaction forces from valve lifting.

Power Relief Valves

The pressurizer is equipped with power-operated relief valves which limit system pressure for a large power mismatch and thus prevent actuation of the fixed high-pressure reactor trip. The relief valves are operated automatically or by remote manual control. The operation of these valves also limits the undesirable opening of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power-operated relief valves if excessive leakage occurs.

Three Power-Operated Relief Valves (PORVs) are installed on the pressurizer as shown on Figure 4.1-1. The PORVs are designed to limit the pressurizer pressure to a value below the high pressure trip setpoint for all design transients up to and including the design percent step load decrease with steam dump but without reactor trip. Manual operator control of the PORVs may also be used as a recovery action to depressurize the RCS following a steam generator tube rupture event. The PORVs are spring loaded to close and require air to open. Air to the PORV actuators is normally supplied by the containment instrument air system; however, each of the three PORVs is also provided with a nitrogen backup supply to the normal containment instrument air system. A nitrogen backup supply is provided for each of the three PORVs in Modes 1-3. The backup nitrogen trains are seismically designed and supported and include a separate accumulator for each PORV. Nitrogen to the three accumulators is provided by the same source used to supply nitrogen to the safety injection accumulators.

Two of the three installed PORVs (PCV-RC-455C and PCV-RC-455D) are also used as part of the Overpressure Protection System (OPPS). The OPPS protects the RCS from pressure transients at low temperatures by preventing any pressure transient from exceeding 10 CFR 50 Appendix G limits, including the use of ASME Code Case N-640.

These two PORVs have a low pressure setpoint which is operational when the Reactor Coolant System decreases below the OPPS enabling temperature defined by the Licensing Requirements Manual. As described in Section 4.2.3, the design basis for the low pressure setpoint is to limit RCS overpressurization resulting from either mass input or heat input transients to the RCS. The design basis for OPPS assumes no operator action for the first ten minutes following the initiating event.

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

Valve Operability Tests

Full size proof tests to show that the pressurizer safety and relief valves and block valves would perform their intended function were performed. See Reference 6 for details.

Loop Stop Valves

The reactor coolant loop stop valves shown in Figure 4.2-9 are remotely controlled motor operated gate valves which permit any loop to be isolated from the reactor vessel. One valve is installed on each hot leg and one on each cold leg. Coolant is circulated in an isolated loop through a bypass line, which contains a remotely controlled motor-operated stop valve. This bypass valve is closed during normal loop operation. To protect the reactor coolant pump, a valve-pump interlock circuit prevents the starting of the reactor coolant pump in a given loop unless the cold leg (discharge) valve is closed and the bypass valve is open. The interlock also prevents pump operation if the bypass valve and either of the stop valves are closed.

Note that Tech Spec Amendment 195 revised the method for isolated loop startup; therefore, the interlock system for opening a cold leg loop stop valve, as described below, can be procedurally bypassed.

To ensure against an accidental startup of an unborated and/or cold isolated loop, an additional valve interlock system is provided which meets IEEE 279-1971. The additional interlocks are for the purpose of ensuring flow from the isolated loop to the remainder of the RCS takes place through the relief line stop valve (after system pressure is equalized through the loop drain header and the hot leg stop valve is opened) for a period of approximately 90 min. before the cold leg loop stop valve is opened.

The flow through the relief line is low (approximately 200 to 300 gpm) so that the temperature and boron concentration are brought to equilibrium with the remainder of the system at a relatively slow rate. The valve-temperature relief line flow interlock:

1. Prevents opening of a hot leg stop valve unless the cold leg loop stop valve is closed.

2. Prevents starting a reactor coolant pump unless:

    a. The cold leg loop stop valve is closed or

    b. Both the hot leg loop stop valve and cold leg loop stop valve are open.

3. Prevents opening of a cold leg loop stop valve unless:

    a. The hot leg loop stop valve has been opened a specified time.

    b. The loop bypass valve has been opened a specified time.

    c. Flow has existed through the relief line for a specified time.

    d. The cold leg temperature is within 20°F of the highest cold leg temperature in other loops, and the hot leg temperature is within 20°F of the highest hot leg temperature in the other loops.

The parameters of each reactor coolant loop stop valve are shown in Table 4.1-12.

#### 4.2.2.8 RCS Supports

The supports for the major components in the RCS are described in Section 5.2.

### 4.2.3 <u>Pressure-Relief</u>

Pressure-Relieving Devices

The RCS is protected against overpressure by protective circuits such as the high pressure trip and by relief and safety valves connected to the top head of the pressurizer. The relief and safety valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The valve design parameters are given in Table 4.1-8. The valves are further discussed in Section 4.2.2.7.

Overpressure Protection System (OPPS)

The OPPS consists of two separate trains each containing a PORV and associated actuation circuitry. Additionally, the two PORVs (PCV-RC-455C and PCV-RC-455D) used by the OPPS each have a separate backup nitrogen supply that is seismically designed and supported. Each nitrogen supply train includes a nitrogen accumulator. Each accumulator is sized to permit ten minutes of PORV actuation following a low temperature overpressure transient (LTOP) without any operator action. The OPPS design includes a pressure switch on each train that initiates an alarm in the control room when a nitrogen accumulator pressure falls below 600 psig.

The OPPS is designed to operate assuming the limiting single failure in the OPPS in addition to the failure mechanism that initiated the RCS overpressurization transient. Except for the case where a loss of power from station battery #2 occurs with a charging pump in service, the OPPS satisfies the single failure criteria. Since a loss of station battery 2 results in both isolation of the letdown line and loss of power to one PORV, an assumed single failure in the other OPPS PORV train would cause the entire OPPS to be unavailable. To address this scenario, a dedicated operator is provided to the benchboard whenever the RCS is in a water solid condition with a charging pump in service.

At low system temperatures, the allowable system pressure is significantly less than the design pressure of 2485 psig, necessitating additional means to alleviate concerns associated with brittle fracture of the reactor vessel. The enable temperature for the OPPS is given in the Licensing Requirements Manual. Therefore, overpressure mitigation provisions for the reactor vessel, as provided by the OPPS, must be available when the RCS and the reactor vessel are at temperatures less than the enable temperature.

During a normal plant heatup, the RCS is open to the Residual Heat Removal System (RHRS) and may be operated for a short period of time in a water solid mode until a steam bubble is formed in the pressurizer. The RHRS is provided with a self-actuated water relief valve to prevent overpressure caused either within the system itself or from transients transmitted from the RCS. During these low-temperature, low-pressure operating conditions, the OPPS is armed and in a ready status to mitigate pressure transients. In determining the OPPS setpoints, no credit is taken for RHR relief valve operation. When the reactor coolant temperature has increased above the enable temperature, the OPPS is manually disarmed.

During a normal plant cooldown, the OPPS is manually armed as the reactor coolant temperature is decreased below the enable temperature. Note that at this time there is a steam bubble in the pressurizer and the water level is at the normal level for no-load operation. The RHRS is normally placed in service by opening the suction isolation valves prior to the OPPS being placed in service. When the coolant temperature has decreased to about 160°F, the steam bubble may be collapsed and the reactor coolant pumps stopped. The steam bubble may be collapsed with one RCP running to a water solid condition during Mode 5 only. In addition, the steam bubble may be formed with one RCP running from a water solid condition through steam bubble formation. Restrictions of running the RCP water solid are proceduralized. From this point on in the cooldown, the OPPS will be in an active status ready to mitigate pressure transients which might occur.

Potential overpressurization transients to the RCS can be caused by either of two types of events to the RCS, mass input or heat input. Both types result in more rapid pressure changes when the RCS is water solid. Specifically, the OPPS design bases transients are defined as: 1) the mass input transient caused by a normal charging/letdown flow mismatch after the termination of letdown flow and 2) the heat input transient caused by the restart of a Reactor Coolant Pump (RCP) when a temperature asymmetry exists within the RCS due to the continued injection of cold seal injection water.

For a particular mass input transient to the RCS, the relief valve will be signaled to open at a specific pressure setpoint. However, there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters, and results in a maximum system pressure somewhat higher than the set pressure. Similarly, there will be a pressure undershoot, while the valve is relieving, both due to the reset pressure being below the setpoint and to the delay in stroking the valve closed. The maximum and minimum pressures reached in the transient are a function of the selected setpoint and must fall within the acceptable pressure range. Note that the pressure overshoot and undershoot for the mass input case is greatest at low temperatures. Thus, the overshoot calculation is limited to the most restrictive low temperature condition only. Whereas, the heat input evaluation calculates the pressure overshoot for a range of reactor coolant temperatures.

The range of allowable setpoints for the OPPS is determined by superimposing the results of the several mass input and heat input cases evaluated and selecting setpoints that will satisfy both types of transients. The selection of the pressure setpoint for the PORVs is based on the use of nominal upper and lower limits.[7] The OPPS is considered to be a mitigation system, as opposed to a protection system, and the use of nominal limits is understood and approved by the NRC. The steady-state pressure-temperature limit is used as the basis for setpoint selection to provide the greatest operational flexibility. This limit has been accepted by the NRC with the justification that "most transients occur during isothermal metal conditions."

The development of the reactor coolant system heatup and cooldown curves is accomplished by following the guidance provided by Regulatory Guide 1.99, Revision 2. These curves represent operational limitations to be followed during heatup or cooldown transitions and are not utilized by themselves for determining the OPPS setpoint since they do not represent steady-state pressure-temperature conditions.

Thus, the OPPS is designed to provide the capability, during relatively low temperature RCS operation, to prevent the RCS pressure from exceeding allowable limits. The OPPS is designed with redundant components to assure it will perform its function assuming any single active component failure. It is provided in addition to the administrative controls to reduce the likelihood that pressure transients will exceed the Licensing Requirements Manual pressure/temperature limits.

### 4.2.4 Protection Against Proliferation of Dynamic Effects

Engineered safety features and associated systems are protected from loss of function due to dynamic effects and missiles which might result from a LOCA. Protection is provided by missile shielding and/or segregation of redundant components.

The RCS is surrounded by concrete shield walls. These walls provide shielding to permit access into the containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plate.

The concrete deck over the RCS and the concrete floor under the RCS also provide for shielding and missile damage protection.

Steam generator lateral bracing is provided near the top of the lower shell section to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing is provided at a lower elevation to resist pipe rupture loads.

Missile protection afforded by the arrangement of the RCS is illustrated in the containment structure arrangement drawings which are given in Section 5. Criteria are discussed in Section 5.2.6. As can be seen from these drawings, a further protection against dynamic effects results from separation of loops by compartment walls.

### 4.2.5 Materials of Construction

Each of the materials used in the RCS is selected for the expected environment and service conditions. The major component materials are listed in Tables 1.8-1, 1.8-2 and 1.8-3.

All RCS materials which are exposed to the coolant are corrosion- resistant. They consist of stainless steels and Nickel-Chromium-Iron alloy (Ni-Cr-Fe), 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 specifications given in Table 4.2-2. Reactor coolant chemistry is further discussed in Section 4.2.8.

The water in the secondary side of the steam generators is held within the chemistry specifications given in Table 4.2-3 based on industry guidelines[10] so as to control deposits and corrosion inside the steam generators.

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, the presence of chlorides, fluorides, or dissolved oxygen. With regard to the former, 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 resulting from normal manufacturing procedures. However, the steam generator contains Ni-Cr-Fe tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in fluoride and chloride aqueous solutions has indicated that Ni-Cr-Fe alloy has excellent resistance to general and pitting-type corrosion in severe operating water conditions. Extensive operating experience with Ni-Cr-Fe units has confirmed this conclusion. The steam generator U-tubes are fabricated of Ni-Cr-Fe Alloy 690. The tubes undergo thermal treatment following tube-forming and annealing operations. Thermally treated Alloy 690 has been shown in laboratory tests and operating nuclear power plants to be very resistant to primary water stress corrosion cracking (PWSCC) and outside diameter stress corrosion cracking (ODSCC).

All external insulation of RCS components is compatible with the component materials. The reactor vessel, reactor coolant pumps, pressurizer, steam generators, and piping and valves 3 inches and larger requiring inservice inspection are insulated with removable stainless steel reflective insulation. Piping and valves smaller than 3 inches requiring inservice inspection are insulated with insulation having removable sections in the weld areas.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility which ensures a low NDT temperature, and thereby ensures that the finished vessel 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 change, including an increase in the NDT temperature.

The techniques used to measure and predict the integrated fast neutron (E > 1 Mev) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum neutron (E > 1 Mev) exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectrum at the sample can be applied with confidence to the adjacent section of reactor vessel, the vessel exposure is obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The original maximum integrated fast neutron (E > 1 Mev) exposure of the vessel was computed to be $2.59 \times 10^{19}$ n per $cm^2$ at 1/4 thickness for 40 yr. operation at 2,660 MWt at 80 percent load factor.

The predicted RT(NDT) temperature shift for an integrated fast neutron (E > 1 Mev) exposure of $2.59 \times 10^{19}$ n per $cm^2$ was 202°F, the value obtained from the curve shown in Figure 4.2-8 for irradiation (0.20 percent Cu).

To evaluate the RT(NDT) temperature shift of welds, heat affected zones, and base material for the vessel, test coupons of these materials types have been included in the reactor vessel surveillance program described in Section 4.5.

The methods used to measure the initial RT(NDT) temperature of the reactor vessel base plate material are given in Appendix 4A.

### 4.2.6 Maximum Heating and Cooling Rates

The RCS operating cycles used for design purposes are given in Table 4.1-10 and described in Section 4.1.5. The heating and cooling rate limits are as described in the Licensing Requirements Manual. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate of 55°F per hr, starting with a minimum water level. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogenous with the coolant.

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

## 4.2.7 Leakage

### 4.2.7.1 Leakage Detection

Coolant leakage from the RCS to the containment is indicated in the main control room by one or more of the following methods:

1. The containment air recirculation fan coolers normally maintain the containment atmosphere at its design operating temperature and humidity. Leakage from the RCS will increase the heat load on the coolers. A 5 gpm leak from the RCS will increase this heat load about 20 percent. The outlet temperature of cooling water for the containment air recirculation fan is indicated in the control room.

2. Containment atmosphere gas and particulate radiation monitors. Experience has shown that these monitors, which are described in Section 11.3.4, respond to reactor coolant leakage and provide an indication of such leakage. The time required to detect reactor coolant leakage depends on the size of the leak, reactor coolant activity level, and containment background activity.

   Only one of the two detectors is required to be OPERABLE. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE, but have recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. If there are few fuel element cladding defects and low levels of activation products (e.g., during startup), it may not be possible for the gaseous or particulate containment atmosphere radioactivity monitors to detect a 1 gpm increase within 1 hour. However, the gaseous or particulate containment atmosphere radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour, given an RCS activity equivalent to that assumed in the design calculations for the monitors.

3. Any leakage causes an increase in the amount of makeup water required to maintain normal level in the pressurizer and/or volume control tank. The primary grade water and concentrated boric acid makeup flow rate are both recorded and alarmed in the main control room.

4. Containment sump water level. Leakage causes the containment sump water level to increase. The containment sump level is also indicated and alarmed in the main control room.

5. Containment pressure, temperature, and humidity instrumentation. Leakage causes the containment pressure, temperature, and humidity to increase. The containment pressure, temperature, and humidity are indicated in the main control room and recorded in the data logger. The containment pressure is also alarmed in the main control room.

6. Reactor vessel flange leakage is collected in the primary drains tank (PDT) and indicated by high temperature in the flange leakoff line (alarm in the main control room). Leakage from this pathway is monitored by the alarm function which annunciates if the temperature detector in the head flange leakoff line reaches a predetermined temperature. Leakage from this pathway is accounted for since this leakage is collected in the PDT which is included in the RCS water inventory balance determinations.

Method 2 can only be used for leakage detection if there is sufficient radioactivity in the reactor coolant. If there is not, the other methods can be used to detect a leak.

Leakage from unidentified sources will pass to the containment structure in the liquid and vapor phase and will be collected in the containment sump. The containment structure has areas that may temporarily hold up small amounts of liquid and thus prevent the liquid from immediately reaching the containment sump. In addition, the containment sump also collects liquid from sources other than the reactor coolant boundary. The determination of exact reactor coolant pressure boundary leakage, accurate to 1 gpm, within 4 hours, by measuring collected water in the containment sump, is reliable. A response time of 4 hours is consistent with the guidance of NRC Generic Letter 84-04.

Intersystem leakage, such as leakage from the reactor coolant system can be detected by continuous radiation monitors, described in Section 11.3.3.3.13 and the Technical Specifications. Intersystem leakage from the reactor coolant system to the component cooling water system can be detected by a continuous radiation monitor described in Section 11.3.3.3.10.

The leakage detection system is capable of detecting a 1 gpm leak in 4 hours. This is to ensure that the RCS Primary loop piping leakage will be detected and action taken prior to a flaw reaching critical size and causing a pipe rupture. Reference 4 discusses critical cracks in the piping system. This reference shows that, for lines 3 inches or more in diameter, leakage through a critical through-wall crack is considerably greater than the minimum detectable leak. This report also notes that for pipes greater than 4 inches in diameter a crack capable of leaking at 5 gpm is considerably smaller than a critical crack. Therefore, catastrophic failure of the piping system is not expected for this 5 gpm leak. For lines 4 inches and smaller, core cooling analysis shows that breaks of this equivalent cross-sectional area will not result in reactor fuel clad damage; therefore, the sensitivity of 1 gpm under all conditions is justified.

#### 4.2.7.2 Leakage Prevention

RCS components are manufactured to exacting specifications which exceed normal code requirements as outlined in Section 4.1.6. Leakage through metal surfaces or welded joints is unlikely because of the welded construction of the RCS and the extensive nondestructive testing to which it is subjected.

Some leakage from the RCS 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. Because of the large number of joints and the difficulty of ensuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable. All valves three (3) inches and larger in lines connecting to the RCS which are normally subjected to RCS operating conditions are provided with leakoff connections. Some of these valves are equipped with backseats which limit leakage.

#### 4.2.7.3 Locating Leaks

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

Methods of locating leaks during a unit shutdown include visual observation for escaping steam or water, or of boric acid crystals near the leak. The boric acid crystals are transported outside the RCS in the leaking fluid and deposited by the evaporation process.

### 4.2.8 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of RCS surfaces.

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality listed in Table 4.2-2. Maintenance of the water quality to minimize corrosion is accomplished using the chemical and volume control system and sampling system which are described in Sections 9.1 and 9.6, respectively.

A soluble zinc compound (zinc acetate) is injected, by means of a Zinc Injection Skid, into the RCS inventory during normal plant operation in order to control radiation fields. The addition of zinc into the RCS is also performed as a means to inhibit general corrosion and primary water stress corrosion cracking of primary system material and components.

The zinc compound is injected into the Chemical and Volume Control System upstream of the charging pumps and downstream of the Chemical Mixing Tank.

In order to monitor and maintain the secondary water to the specifications given in Table 4.2-3, it is intended that automatic continuous analysis equipment will be provided for the Hotwell, Condensate Pump and Feed Pump sample points.

The necessary associated equipment will also be provided to enable electronic signals to be transmitted to panel mounted recorders which will also be provided with alarm set points respective to the particular analysis. In addition to the above, laboratory analyses are performed in accordance with procedures and schedules set forth in the BVPS Chemistry Manual.

In order to provide assurance that stress corrosion cracking problems on tubes will not be encountered during all conditions of steam generator operations, the general specifications listed below will be instituted in accordance with the BVPS Chemistry Manual.

Operating Conditions

1. Neutralizing amines are used for feedwater and steam pH control. An oxygen scavenger is used to control feedwater oxygen to less than or equal to .005 ppm.

2. Water chemistry control is utilized to minimize any impact of contamination which might prevail in the event of condenser leakage.

3. Continuous steam generator blowdown along with all volatile treatment (chemical feed) is normally maintained in all operating boilers to provide removal of steam generator impurities.

The steam generator tube material is thermally treated Ni-Cr-Fe Alloy 690 (i.e., Alloy 690 TT), which has been shown through both laboratory testing and operational experience to provide enhanced corrosion resistance compared to previous materials (e.g., Alloy 600) with regard to outer diameter stress corrosion cracking (ODSCC). Since austenitic stainless steel does not come in contact with this water, chloride stress corrosion will be of no significant consequence within the limits of the specification.

During inactive periods involving layup of the steam generators, measures are taken to minimize the amount of oxygen in the steam generator. All water and operating conditions employed during layup of the steam generators are required to meet those specifications as set forth in the BVPS Chemistry Manual. This manual will provide for oxygen scavenge during wet layup of the secondary water system.

### 4.2.9 Reactor Coolant Flow Measurements

Elbow taps are used in the RCS as an instrument device that indicates the status of the reactor coolant flow[3]. 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 is well established by the following equation:

$$P/P_o = (W/W_o)^2 \qquad (4.2\text{-}1)$$

where: $P_o$ is the referenced differential pressure
$W_o$ is the reference flow rate for "$P_o$"
$P$ is the pressure differential
$W$ is the referenced flow rate for "$P$"

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

### 4.2.10 Loose Parts Monitoring

The design of Beaver Valley Power Station Unit 1 included a loose parts monitoring system (LPMS) to detect abnormal metallic impacts in the primary coolant system. Through subsequent evaluation it has been determined that this system is not required. Therefore, various features may be eliminated, modified or maintained as an operating convenience.

Characteristic vibrations from such components as core internals, pumps and steam generator tubes will occur at low frequencies (0-200 HZ). Anomalous noise from such sources as pump bearings, steam separators or pump seal leakage would be identified at higher frequencies (up to 10 kHZ). Metal impacts from loose parts would be detected at super-audio frequencies.

### 4.2.11 Reactor Coolant Gas Vent System

The reactor coolant gas vent system provides a means to remotely vent noncondensable gases from the reactor vessel head or the pressurizer steam space. The system is designed in accordance with ASME III. The reactor coolant gas vent system was added in response to NUREG 0737 TMI Issue II.B.1.

The reactor coolant gas vent system is used following an accident which generates significant quantities of noncondensable gases that inhibit core cooling. The need for venting and the approximate volume of gas can be determined for the reactor from the reactor vessel level instrumentation and for the pressurizer from the pressurizer temperature and pressure. The gases may be discharged into the containment dome or the pressurizer relief tank (see Figure 4.1-1). The flow from the system is less than 100 cfm, $H_2$, depending on temperature and pressure. The venting operation is accomplished from the control room using keyswitches to open any valves requiring opening. The solenoid operated valves fail closed on a loss of control power. Position indicating lights on the control board provide a positive indication of each valve position.

Two solenoid operated valves in parallel are installed in both vessel head vent line and the pressurizer vent line to provide assurance that the required vent path may be established when needed. One solenoid operated valve is installed on each exhaust line such that each vent path has double isolation. Parallel valves are powered from separate emergency buses such that a vent path is available from the vessel head or pressurizer with the loss of power from either bus.

The potential flow from the reactor coolant system is restricted to less than the make up capacity of one charging pump by a 7/32 inch orifice at the vent line connections to the reactor vessel head and the pressurizer. Inadvertent actuation of both the reactor vessel and the pressurizer vent at the same time is prevented by use of a single key for the four control board keyswitches which control the parallel valves on each vent path. A single key is provided for actuation of either of the two exhaust line solenoid operated valve control board keyswitches.

Pressure instrumentation in the vent line will detect leakage past the reactor vessel head and pressurizer solenoid operated valves (provided all six of the solenoid operated valves are closed). During normal plant operation, the vent line pressure is displayed in the control room.

The exhaust line to containment discharges directly into the containment dome, without obstruction so as to provide good mixing of vented gases. The use of the pressurizer relief tank allows removal of small quantities of gas without releasing radioactive fluid/gas into the containment or increasing hydrogen concentration levels. Venting of large quantities of gas to the pressurizer relief tank will rupture the rupture disc providing a second path to containment for vented gas.

References for Section 4.2

1. Deleted by Revision 17.

2. Deleted by Revision 17.

3. J. W. Murdock, "Performance Characteristics of Elbow Flowmeteres," Transactions of the ASME, (September 1964).

4. J. J. Szyslowski, R. Salvatori, "Determination of Design Pipe Breaks for Westinghouse Reactor Coolant Systems" WCAP-7503 Revision 1, Westinghouse Electric Corporation (February 1972).

5. Duquesne Light Company to NRC submittal concerning NUREG-0737, Item II.D.1 Pressurizer Safety and Relief Line Piping and Support Evaluation dated June 24, 1983.

6. Duquesne Light Company to NRC submittal concerning NUREG-0737, Item II.D.1 Plant Specific Report, dated July 1, 1982.

7. "Beaver Valley Unit 1, FirstEnergy Nuclear Operating Company Overpressure Protection System Setpoints for Y-Capsule," Revision 1, dated April 2001, transmitted per Westinghouse letter, FENOC-01-0136, dated June 21, 2001.

8. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

9. P. L. Strauch, W. H. Bamford, B. A. Bishop, D. Kurek, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," WCAP-14535-A, Westinghouse Electric Corporation (January 1996).

10. EPRI Report TR-102134-R5, “PWR Secondary Water Chemistry Guidelines,” May 2000.

11. EPRI Report TR-016743-V2R1 Volume 2, “Guidelines for Procurement of Alloy 690 Steam Generator Tubing,” April 1999.

# 4.3 SYSTEM DESIGN EVALUATION

## 4.3.1 Safety Factors

The safety of the reactor vessel and all other RCS 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.

The analytical methods used to evaluate stresses for Westinghouse ASME Section III components are described below. These methods are used for elastic system analyses with both elastic and inelastic subsystem analysis for faulted conditions.

Stress analyses of reactor coolant pressure boundary piping systems are based on the elastic analytical procedures specified in ANSI-B31.1 Power Piping Code as discussed in Appendix B.2.1.

### Reactor Vessel

The following components of the reactor pressure vessel are analyzed in detail through systematic analytical procedures:

1. Control Rod Housings
2. Closure Head Flange and Shell
3. Main Closure Studs
4. Inlet Nozzle (and Vessel Support)
5. Outlet Nozzle (and Vessel Support)
6. Vessel Wall Transition
7. Core-Barrel Support Pads
8. Bottom Head to Shell Juncture
9. Bottom Head Instrument Penetrations, etc.

### Method of Analysis

Item 1. The replacement reactor vessel closure head, the head vent and RVLIS tubes, the CRDM housings and their significant junctures were analyzed with an updated finite element analysis.

Item 2. The original closure head with flange, vessel flange, vessel shell, and closure studs were all evaluated in the same analysis. An analytical model is developed by dividing the actual structure into different elements such as sphere, ring, long cylinder and cantilever beam, etc. An interaction analysis is performed to determine the stresses due to mechanical, thermal and seismic loads. These stresses are evaluated in light of the strength and fatigue requirements of the ASME Boiler and Pressure Vessel Code, Section III. The replacement reactor vessel head and simplified head assembly are provided with their own analysis.

Item 3. A similar analysis is performed for the vessel flange to vessel shell juncture; and main closure studs.

Item 4. For the analysis of nozzle and nozzle to shell juncture, the loads considered are internal pressure, operating transients, thermally induced and seismic pipe reactions, static weight of vessel earthquake loading and expansion and contraction, etc. A combination of methods is used to evaluate the stresses due to mechanical and thermal loads 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 enables the fatigue evaluation to be performed.

Item 5. Method of analysis for outlet nozzle and vessel supports is the same as described above.

Item 6. Vessel wall transition is analyzed by means of a standard interaction analysis. The thermal stresses are determined by the skin stress method where it is 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 7. Thermal, mechanical, and pressure stresses are calculated at various locations on the core barrel support pad and at the vessel wall. Mechanical stresses are calculated by the flexure formula for bending stress in a beam, pressure stresses are taken from the analysis of the vessel to bottom head juncture and thermal stresses are determined by the conservative method of skin stresses. The stresses due to the cyclic loads are multiplied by a stress concentration factor where applicable and used in the fatigue evaluation.

Item 8. The standard interaction analysis and skin stress methods are employed to evaluate the stresses due to mechanical and thermal stresses, respectively. The fatigue evaluation is made on a cumulative basis where superposition of all transients is taken into consideration.

Item 9. An interaction analysis is performed by dividing the actual structure into an analytical model composed of different structural elements. The effects of the redundants on the bottom head are assumed to be local only. It is also assumed that for any condition where there is 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 is made for the J weld.

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.

Fatigue Analysis Based on Transient Stresses

For fatigue evaluations, in accordance with the ASME Boiler and Pressure Vessel Code, maximum stress intensity ranges are derived from combining the normal and upset condition transients in a conservative manner. All the transient occurrences are accounted for. The stress ranges and number of occurrences are then used in conjunction with the fatigue curves in the ASME Boiler and Pressure Vessel Code to get the associated cumulative usage factors.

The design fatigue curves are obtained by applying either a factor or two on the stress or a factor of twenty on the number of cycles (whichever is more conservative) at each point on a "best fit" curve developed from a large amount of fatigue test data. These factors are a multiplication of factors intended to cover such effects as environment, size effect, surface condition, and scatter of test data.

The conservatism of the design fatigue curves has been demonstrated by the Pressure Vessel Research Committee in a series of cyclic pressurization tests of model vessels fabricated to the code. The results of the Pressure Vessel Research Committee 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. Similar fatigue tests have been performed on irradiated pressure vessel steels with comparable results.[1]

The failure criterion presented in the ASME Boiler and Pressure Vessel Code is used for the fatigue failure analysis. In no case does the cumulative usage factor exceed the code allowable usage factor of 1.0 and hence the fatigue design is adequate.

Thermal Stresses Due to Gamma Heating

The stresses due to gamma heating in the vessel wall are also calculated by the vessel vendor and combined with the other design stresses. They are compared with the code allowable limit for mechanical plus thermal stress intensities to verify that they are acceptable. The gamma stresses are low and thus have a negligible effect on the stress intensity in the vessel.

Thermal Stresses Due to Loss-of-Coolant Accident

In the event of a large loss-of-coolant accident, the RCS rapidly depressurizes, and the loss-of-coolant may empty the reactor vessel. If the reactor is at normal operating conditions before the accident, the reactor vessel temperature is approximately 550 F; and if the plant has been in operation for some time, part of the reactor vessel is irradiated. At an early stage in the depressurization transient, the ECCS rapidly injects cold coolant into the reactor vessel. This results in thermal stress in the vessel wall. To evaluate the effect of the stress three possible modes of failure are considered; ductile yield, brittle fracture, and fatigue.

Ductile Mode - The failure criterion used for this evaluation is that there shall be no gross yielding across the vessel wall using the material yield stress specified in Section III of the ASME Boiler and Pressure Vessel Code. The combined pressure and thermal stresses during injection through the vessel thickness as a function of time have been calculated and compared to the material yield stress at the times during the safety injection transient.

The results of the analyses showed that local yielding may occur only in approximately the inner 18 percent of the base metal and in the vessel cladding, complying with the above criterion.

Brittle Mode - The possibility of a brittle fracture of the irradiated core region has been considered utilizing fracture mechanics concepts. This analysis is performed assuming the effect of water temperature, heat transfer coefficients, and fracture toughness as a function of time, temperature, and irradiation. Both a local effect and a continuous crack effect have been considered with the latter requiring the use of rigorous finite element axisymmetric code. It is concluded on the weight of this evidence that thermal shock resulting from the loss-of-coolant accident will not produce instability in the vessel wall even at the end of the original 40-year license term. For the period of extended operation beyond the original 40-year license term, embrittlement of the vessel wall will be managed by the Reactor Vessel Integrity Program, as discussed in Section 16.1.35.

Fatigue Mode - The failure criterion used for the failure analysis was as presented in Section III of the ASME Boiler and Pressure Vessel Code. In this method the piece is assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceeds the code allowable usage factor of one.

The location in the vessel below the nozzle level which will see the emergency core cooling water and have the highest usage factor will be the incore instrumentation tube attachment welds to the vessel bottom heads. As a worst case assumption, the incore instrumentation tubes and attachment penetration welds are considered to be quenched to the cooling water temperature while the vessel wall maintains its initial temperature before the start of the transient. The maximum possible pressure stress during the transient is also taken into account. This method of analysis is quite conservative and yields calculated stresses greater than would actually be experienced. The resulting usage factor for the instrument tube welds considering all the operating transients and including the safety injection transient occurring at the end of the plant life is below the code allowable usage factor of 1.0.

It is concluded from the results of these analyses that the delivery of cold emergency core cooling water to the reactor vessel following a loss-of-coolant accident, does not cause any loss of integrity of the vessel.

All stress analyses information is contained in the reactor vessel vendor's stress report. The stress report includes a summary of the stress analyses for regions of discontinuity analyzed in the vessel, a discussion of the results including a comparison with the corresponding code limits, a statement of the assumptions used in the analyses, descriptions of the methods of analyses and computer programs used, a presentation of the actual hand calculations used, a listing of the input and output of the computer programs used, and a tabulation of the references cited in the report. The content of the stress report is in accordance with the requirements of the ASME Boiler and Pressure Vessel Code and all information in the stress report is reviewed and approved by Westinghouse. This report is available for review and use at Westinghouse.

A summary of the primary plus secondary stress intensity for components of the reactor vessel and the cumulative fatigue usage factors for the components of the 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 unit operation coupled with experience from nuclear power stations now in service, such as Yankee-Rowe.

The vessel design pressure is 2,485 psig, while the normal operating pressure is 2,235 psig. The resulting operating membrane stress is therefore amply below the code allowable membrane stress to account for operating pressure transients.

Assurance of adequate fracture toughness of the reactor coolant system is provided by compliance with the requirements for fracture toughness testing in the 1980 issue of the ASME Pressure Vessel and Boiler Code, Section III.

Heatup and cooldown limit curves are calculated using the most limiting value or RT(NDT) (reference nil-ductility temperature) for the reactor vessel. The most limiting RT(NDT) of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material fracture toughness properties and estimating the radiation-induced RT(NDT). RT(NDT) is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT(NDT) increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT(NDT) at any time period in the reactor's life, RT(NDT) due to the radiation exposure associated with that time period must be added to the original unirradiated RT(NDT). The extent of the shift in RT(NDT) is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Rev. 2 (Radiation Embrittlement of Reactor Vessel Materials).

The results of the irradiation surveillance program will be used to verify that the RT(NDT) used in developing the heatup and cooldown limit curves in the Licensing Requirements Manual is appropriate and to make any changes necessary to correct these curves.

The use of an RT(NDT) that include a RT(NDT) to account for radiation effects on the core region material, automatically provides additional conservatism for the nonirradiated regions. Therefore, the flanges, nozzles, and other regions not affected by radiation will be favored by additional conservatism approximately equal to the assumed RT(NDT).

Details of the radiation effects surveillance program will be based on the evaluation of the test results on the actual vessel material.

Changes in fracture toughness of the core region plates or forgings, weld metal, and associated heat treated zone due to radiation damage will be monitored by a surveillance program which conforms with ASTM E-185-82. The evaluation of the radiation damage in this surveillance program is based on pre-irradiation and post-irradiation testing by Charpy V-notch, dropweight test, and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core midheight and removable from the vessel at specified intervals.

Thermal Stresses Due to Main Steam Line Break

This plant meets the intent of the 1980 issue of ASME, Section III, Appendix G 2000, and the reactor vessel beltline material contains a low copper content; therefore, the integrity of the reactor vessel would be maintained during a postulated steam pipe break throughout its service life.

Irradiation Surveillance Program

The program will conform with ASTM E-185-82 as well as the applicable NRC requirements. Capsules will contain sufficient Charpy V-notch specimens of base metal, weld metal, and weld heat zone to monitor changes in the RT(NDT) of the core region material predicted to be most limiting based on original fracture toughness properties and estimated changes considering copper and phosphorous contents. Core region material sufficient for at least two additional capsules will be retained.

Details of the program are based on the evaluation of the test results on the actual vessel material. For additional details of the Irradiation Surveillance Program, see Section 4.5.

Capability for Annealing the Reactor Vessel

There are no special design features which would prohibit the insitu annealing of the vessel. If the unlikely need for an annealing operation was required to restore the properties of the material opposite the reactor core because of neutron irradiation damage, the operation could be performed with the use of a special electrical space heater assembly designed to raise the affected vessel area to the required temperature for the necessary holding period.[9][10]

The reactor vessel materials surveillance program is adequate to accommodate the annealing of the reactor vessel. The remaining surveillance capsules at the time of annealing would be removed and given a thermal cycle equivalent to the annealing cycle. They would then be reinserted in their normal position between the core internals assembly and the reactor vessel wall. Subsequent testing of the fracture toughness specimens from the capsules would then reflect both the radiation environment before any annealing operation and after any annealing operation.

Type 308 weld filler material is used for stainless steel welding applications to avoid microfissuring. As an option, Type 308L weld filler metal analysis is substituted for consumable inserts when this technique is used for the weld root closure. Bare weld filler metal materials, including consumable inserts used in inert gas welding processes, conform to ASME SFA-5.9 and are procured to contain not less than 5 percent delta ferrite. All weld filler metal materials used in flux shielded welding processes conform to ASME SFA-5.4 or SFA-5.9 and are procured in a wire-flux combination to be capable of providing not less than 5 percent delta ferrite in the deposit. Electrodes conforming to SFA-5.4 are of the -15 or -16 (lime type) current characteristics.

All welding materials are tested by the fabricator using the specific process(es) and the maximum welding energy inputs to be employed in production welding. These tests are in accordance with the requirements of ASME Section III, N-511, and in addition include delta ferrite determinations as applicable for stainless steel. These determinations are made by calculation using the "Schaeffler Constitution Diagram for Stainless Steel Weld Metal." Subsequent in-process delta ferrite determinations are not required. Other methods of ferrite determinations are usable on the basis of the developmental data and recommendations concurrently existing from the Advisory Subcommittee for Welding Stainless Steel of the High Alloy Committee in the Welding Research Council. The cobalt content may not exceed 0.20 percent in stainless steel cladding. In stainless steel welding, the interpass temperature is limited to 350°F maximum.

Methods that limit severely sensitized stainless steel are:

1. Reactor Vessel

   a. Primary nozzle safe ends are made of stainless steel weld metal buildup with over 5 percent ferrite and subjected to a postweld heat treatment.

   b. Bottom instrumentation nozzles, vent pipe assemblies, control rod drive mechanism assemblies have stainless steel safe ends attached after final postweld heat treatment.

   c. The CRDM housings are made of two pieces, the housing made of 690 Inconel and the adapter flange made of type 316 stainless steel. This assembly is welded to the closure head after post weld heat treatment of the head.

2. Steam Generators where:

   a. Primary nozzle safe ends made of forged austenitic stainless steel, subjected to heat treatment.

3. Pressurizer nozzles are made of Type 316 stainless steel or equivalent,[11] subjected to minimize time of postweld heat treatment.

Piping

Following is a basic description of the procedures that are used to perform a dynamic analysis of the reactor coolant loop piping system. This analysis is discussed in Appendix B.

The analysis of the reactor coolant loop/supports system is conducted based on an integrated analytical model which includes the effects of the supports and the supported equipment.

A three-dimensional, multi-mass, elastic-dynamic model is constructed to represent the reactor coolant loop/supports system. The seismic floor spectrum at the internal concrete to support interface, obtained from an elastic-dynamic model of the reactor containment internal structure, is used as input to the piping analysis. The dynamic analysis employs displacement method, lumped parameter and stiffness matrix formulations.

Normal Operating Loads

System design operating parameters are used as the basis for the analysis of equipment, coolant piping, and equipment support structures for normal operating loads. The analysis is 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 is in accordance with the provisions of ANSI B31.1 and ASME Section III. Results of the analysis give six generalized force components, three bending moments and three forces. These moments and forces are resolved into stresses in the pipe in accordance with the applicable codes. Stresses in the structural supports are determined by the material and section properties.

Seismic Loads

Analysis for seismic loads for the reactor coolant system piping is based on a dynamic model. The appropriate floor spectral accelerations are used as input forcing functions to the detailed dynamic model. The loads developed from the dynamic model are incorporated into a detailed support model to determine the support member stresses.

Blowdown Loads

Analysis of blowdown loads resulting from a LOCA is based on the time-history response of simultaneously applied blowdown forcing functions on a single broken loop dynamic model. The forcing functions are 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 model are incorporated into a detailed support model to determine the equipment support member stresses.

Combined Blowdown and Seismic Loads

The stresses in components resulting from normal loads and the worst case blowdown analysis are 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 occur at the same instant. These stresses are combined to determine that the reactor coolant loop/support system does not lose its intended functions under this highly improbable situation.

Steam Generators

The steam generator is designed and analyzed in accordance with the ASME Boiler and Pressure Vessel Code, (herein referred to as the ASME Code), Section III, 1989 Edition, no Addenda (Reference 3). The steam generator primary side is classified as ASME Code Class 1; the steam generator secondary side is classified as ASME Code Class 2. The design, analysis, material, fabrication, inspection, examination, and testing of steam generator components and assemblies are in accordance with ASME Code.

Structural analyses were performed in accordance with the ASME Code, Section III, 1989 Edition, no Addenda in order to demonstrate the structural integrity of the steam generator components. The analyses evaluated the primary side components (including channel head, divider plate, primary nozzles, tubesheet, and tube-to-tubesheet weld) and the secondary side components (including the upper shell, shell transition cone, lower shell, junction of tubesheet and stub barrel, feedwater nozzles, secondary manway openings, handholes, inspection ports, and shell taps).

The steam generator structural analyses (pressure boundary and internals) considered the following sources of loadings:

1. Deadweight
2. Primary system pressure
3. Secondary system pressure
4. Pipe nozzle loadings
5. Pipe rupture loadings

6. Seismic inertia and anchor movements
7. Operating Basis Earthquake (OBE)
8. Safe Shutdown Earthquake (SSE)
9. Flow loads (based on pressure, temperature and flow information)
10. Thermal loads (based on primary, steam and feedwater temperatures)

Normal operating conditions (pressures, temperatures and flows) were obtained from the plant parameters and the transient operating conditions were obtained from the NSSS design transients. The NSSS design transients include transient operating conditions for both the primary side (reactor coolant) and the secondary side. Loss-of-coolant primary pipe break hydraulic forcing functions were obtained from the LOCA hydraulic forces analysis for the steam generator. OBE and SSE seismic response spectra, pipe rupture loadings, and pipe nozzle loadings were obtained from applicable design specifications and the reactor coolant loop analysis for the steam generator. The Design Limits for Level A and B (Normal and Upset) Conditions and for Level D (Faulted) Conditions were obtained from Subsection NB and Appendix F, respectively, of the ASME Code. Service Limits were also obtained from the ASME Code. The steam generator loading conditions are outlined in Table 4.3-3.

The principal computer codes used in the dynamic and static analyses of the steam generators to determine mechanical loads, stresses, and deformations are general purpose finite element structural analysis computer codes that are used in the design and analysis of steam generator components. Structural analyses were performed for the following steam generator components:

1. Primary Chamber, Tubesheet, Stub Barrel Complex
2. Primary Nozzles
3. Primary Manways
4. Primary Chamber Divider Plate
5. Tube/Tubesheet Weld
6. Tubes
7. Secondary Shell
8. Handhole
9. Inspection Port
10. Minor Shell Taps
11. Feedwater Nozzle and Thermal Sleeve
12. Secondary Manways
13. Steam Outlet Nozzle and Elliptical Head
14. Lower Internals
15. Upper Internals
16. Feedring, Feedring Supports, and Spray Nozzle
17. Upper Shell Platform
18. Feedwater Elbow Thermal Liner

The steam generator structural analyses showed that all applicable requirements of the ASME Code are satisfied for the steam generator components. The structural analyses for steam generator components, including methodology, acceptance criteria and results (maximum stress range/allowable stress range and fatigue usage factors), are documented in the steam generator ASME Code Summary Stress Report.

### Thermal-Hydraulic Performance

The steam generator thermal-hydraulic performance is affected by the normal operating conditions, including the ranges for reactor vessel average temperature, feedwater temperature, and Steam Generator Tube Plugging (SGTP) level. At full power operation, the reactor vessel average temperature and steam generator tube plugging level determine steam pressure. Steam flow at full power is affected by feedwater temperature. Thus, variations in these parameters within their defined ranges impact steam generator thermal-hydraulic performance and affect secondary side steam pressure and steam flow. The full-load performance of the steam generator, including secondary side steam pressure and steam flow, is included in Table 4.1-5.

The steam generator is designed to operate with a full-load circulation ratio in the range of 3.3 to 3.6, depending on where reactor vessel average temperature, feedwater temperature and SGTP level are within their design ranges. The steam generator secondary side masses will vary depending on the actual operating conditions. The steam generator full-load performance data in Table 4.1-5 includes the ranges for secondary side conditions such as circulation ratio and secondary side masses.

The thermal-hydraulic performance characteristics used to evaluate the acceptability of steam generator operation at 2910 MWt (970 MWt/SG) include secondary side thermal-hydraulic parameters (e.g., heat flux, steam flow, steam pressure, secondary side masses), moisture carryover, circulation ratio, hydrodynamic stability, and local (liquid) dryout on tube wall (which addresses the combined effect of secondary side mass flow, heat flux, and pressure).

The thermal-hydraulic analyses performed to evaluate steam generator performance over the design range of operating parameters showed that all thermal-hydraulic characteristics are acceptable, including the thermal-hydraulic performance parameters, moisture carryover, hydrodynamic stability, and local dryout on tube walls. The thermal-hydraulic analyses for the steam generator, including methodology, acceptance criteria and results, are summarized in the following.

### Hydrodynamic Stability

Hydrodynamic stability prevents steady-state oscillations in secondary side thermal-hydraulic parameters. The hydrodynamic stability of a steam generator is characterized by a damping factor. A negative value of this parameter indicates stable operation, and thus small perturbations of steam flow, steam pressure, or feedwater temperature will die out rather than grow in magnitude. This in turn promotes stable water level and control.

A number of factors impact hydrodynamic stability, including power level, steam pressure, and downcomer subcooling (due to feedwater temperature). The steam generator analyses showed that for the design range of full power steam flow and feedwater temperature, the damping factors are substantially negative. Thus, the steam generators will remain hydrodynamically stable over the design range of operating parameters.

### Flow Induced Vibration and Wear

Analyses were performed to evaluate the steam generator tube bundle and support system (including flow distribution baffle, tube support plates, and Anti-Vibration Bars (AVB)) for potential vibration and wear. Consideration was given to potential sources of tube excitation including primary fluid flow within the U-tubes, mechanically induced vibration, and secondary fluid flow on the outside of the tubes. The effects of primary fluid flow and mechanically induced vibration are considered negligible during normal operation. The primary source of potential tube degradation due to vibration is due to the hydrodynamic excitation by the secondary fluid on the outside of the tubes. The three potential tube vibration mechanisms due to hydrodynamic excitation by the secondary fluid on the outside of the tubes are vortex shedding, turbulence, and fluidelastic vibration.

The steam generator analyses showed that calculated fluidelastic stability ratios are less than acceptance criteria, displacements and bending stresses from turbulence are small, and calculated wear depths after the 40 calendar year design operating period are below available design margins for local tube wear. The maximum calculated tube wear depth after the 40 calendar year design operating period is more than a factor of three times lower than the plugging margin available for steam generator operation. The flow induced vibration and wear analyses therefore demonstrate that the steam generator will not experience unacceptable tube degradation due to tube vibration and wear during the 40 calendar year design operating period.

The steam generator tube-to-tubesheet joint provides the structural barrier between the primary and secondary sides of the steam generator. The steam generator tubes are hydraulically expanded full depth through the thickness of the tubesheet to minimize the tube-to-tubesheet crevice while locating the transition as close as practical to the top of the tubesheet. Tubesheet thickness is measured to accurately adjust the hydraulically expanded distance. The hydraulic expansion process is preferred to other techniques, such as the mechanical roll or explosive expansion processes, based on the analysis of test results and experience.

### Tube Plugging Limits (Draft Regulatory Guide 1.121 Analysis)

Draft Regulatory Guide 1.121 describes a method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in- service inspection methods shall be removed from service. This level of acceptable degradation is referred to as the “plugging limit.”

Steam generator analyses were performed to define the “structural limits” for an assumed uniform thinning mode of tube degradation in both the axial and circumferential directions using the ASME Code minimum material properties. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for flawed tubes occurring in the field. The allowable tube plugging limits, in accordance with draft Regulatory Guide 1.121, are obtained by incorporating into the resulting structural limit a growth allowance for continued operation until the next scheduled inspection and also an allowance for eddy current measurement uncertainty.

Calculations are performed to establish the structural limit for the straight leg (free span) region of the tube for degradation over an unlimited axial extent and for degradation over a limited axial extent at the tube support plate and AVB intersections. The analyses include a specification on maximum allowable leak rate during normal operation consistent with the EPRI PWR Primary-to-Secondary Leak Guidelines (Reference 10). These guidelines define several monitoring and action level conditions, depending on the amount of leakage and the rate of leakage increase. The steam generator analyses results establish the tube structural limits for the straight leg (free span), tube support plate and AVB tube locations based on the normal, upset and accident conditions. The analyses showed that leakage monitoring using the EPRI PWR Primary-to-Secondary Leak Guidelines provides a reasonable likelihood that the plant can be shut down before the single tube postulated to be leaking would rupture under either normal or accident conditions.

Reactor Coolant System Loop Stop Valves

The subject equipment are designed as ASME Section III Class A Vessels.

The primary pressure boundary, such as the body and bonnet, are cast from CF8M, or equivalent,[11] having an allowable design stress intensity value (Sm) of 18,700 psi at 650°F. The other pressure containing material within the valve is the disk with an allowable Sm of 16,700 psi also at 650°F.

Calculations confirm that the subject valve can withstand the loadings of the reactor coolant system piping. These calculations were also verified by a strain gage hydrostatic test on the body, bonnet and bolting members of a typical valve.

The subject equipment is subjected to cold and hot operational cycle tests by the manufacturer. All areas are re-inspected after the hot operational tests and recorded.

Pressurizer

The pressurizer is 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 section.

The vessel loading conditions are as follows:

1. The pressurizer vessel, nozzles and vessel supports are designed to resist the following normal operational loadings:

   a. Weight of water based on the vessel filled with cold water, and including insulation.

   b. Normal loadings exerted by connecting piping.

2. The pressurizer vessel, nozzles and vessel supports are designed to resist the following seismic loadings:

   a. For the OBE, the pressurizer vessel is 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 and their combination with the normal operational loads is to be considered an upset condition. The components of loadings exerted by the external piping due to the OBE are included in this evaluation.

   b. For the DBE, pressurizer vessel function is not impaired so as 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 DBE is considered a faulted condition with the following exceptions:

      1) The combination of all primary stress intensities in the vessel support skirt is within the support skirt material yield strength specified in Section III of the ASME Boiler and Pressure Vessel Code.

      2) The stress intensity limits of the vessel associated with this earthquake condition in combination with the normal operational loads are as follows:

         $Pm \leq 1.2 * Sm$

         $Pl + Pb \leq 1.8 * Sm$

where: Pm is the maximum membrane stress.

The components of loadings exerted by the external piping due to the DBE are included in this evaluation.

3. The pressurizer vessel, nozzles and vessel supports are designed to resist the pipe break loadings in combination with the normal operational loads. The moment and forces are considered as acting in combination with each force separately. The pipe break accident is considered to be a faulted condition with the exception of the stress intensity limits being those specified under the DBE condition.

4. The pressurizer vessel, nozzles, and vessel supports are analyzed for the combination of normal operation loads plus the DBE loads plus the pipe break loads. The resulting stress intensities do not exceed the stress intensity limits of Paragraph N17.11 (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. If necessary, higher stress intensity values are adopted in the vessel supports where plastic instability analyses of the support and supported component system are performed in accordance with Paragraph N417.11 of ASME Code Section III.

Pressurizer Design Analysis

The occurrences for pressurizer design cycle analysis are defined as follows:

1. The temperature in the pressurizer vessel is always, for design purposes, assumed to equal saturation temperature for the existing RCS pressure, except in the pressurizer steam space subsequent to a pressure increase. In this case, the temperature of the steam space will exceed the saturation temperature since an isentropic compression of the steam is assumed.

   The only exception of the above occurs when the pressurizer is filled solid during plant start-up and cooldown.

2. The temperature shock on the spray nozzle is assumed to equal the temperature of the nozzle minus the cold leg temperature, and the temperature shock on the surge nozzle is assumed to equal the pressurizer water space temperature minus the hot leg temperature.

3. Pressurizer spray is assumed to be initiated instantaneously to its design value as soon as the RCS pressure increases 40 psi above the nominal operating pressure. Spray is assumed to be terminated as soon as the RCS pressure falls 40 psi below the normal operating pressure.

4. Unless otherwise noted, pressurizer spray is assumed to be initiated once per occurrence of each transient condition. The pressurizer surge nozzle is also assumed to be subject to one temperature transient per transient condition, unless otherwise noted.

5. At the end of each transient, except the faulted conditions, the RCS is assumed to return to a load condition consistent with the plant heatup transient.

6. Temperature changes occurring as a result of pressurizer spray are assumed to be instantaneous. Temperature changes occurring on the surge nozzle are also assumed to be instantaneous.

7. Whenever spray is initiated in the pressurizer, the pressurizer water level is assumed to be at the no-load level.

System Pressure

Whenever a steam bubble is present within the pressurizer, RCS pressure is maintained by the pressurizer. Analyses indicate that proper control of pressure is maintained for the operating conditions.

A safety limit has been set to ensure that the RCS pressure does not exceed the maximum transient value allowed under the ASME Code, Section III, and thereby ensure continued integrity of the RCS boundary.

Evaluation of plant conditions of operation which follow indicate that this safety limit is not required.

During start-up and shutdown, the rate of temperature change is controlled by the operator. When the reactor core is shutdown, the maximum heatup rate by pump energy may be limited by the installed pressurizer electrical heating capacity. This heatup rate takes into account the continuous spray flow provided to the pressurizer.

When the pressurizer is filled with water, i.e., near the end of the second phase of unit cooldown and during initial system heatup, RCS pressure is maintained by the residual heat removal system and the charging pumps.

Pressurizer Performance

The pressurizer has a minimum free internal volume. The normal operating water volume at full load conditions is 60 percent of the free internal vessel volume. Under part load conditions, the water volume in the vessel is reduced for proportional reductions in plant load to 25 percent of free vessel volume at zero (0) power level. The various plant operating transients are analyzed and the design pressure is not exceeded with the pressurizer design parameters as given in Table 4.1-4.

Pressure Setpoints

The RCS design and operating pressure, together with the safety, power relief, and pressurizer spray valves setpoints and the protection system setpoint pressures, are listed in Table 4.1-2. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times, and pressures drops, instrumentation and control response characteristics, and system relief valve characteristics.

Reactor Coolant Pump

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

Mathematical methods of the parts are prepared and used in the analysis which proceeds in two phases.

1. In the first phase, the design is 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 are 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 are used to give a rigorous analysis and conservative design.

Pump Performance

The reactor coolant pumps are sized to deliver flow at rates which equal or exceed the required flow rates. Initial RCS tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation.

The reactor trip system ensures that pump operation is within the assumptions used for loss-of-coolant flow analyses, which also ensures that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation.

An extensive test program has been conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long term tests were conducted on less than full scale prototype seals, as well as on full-size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled leakage shaft seal pump design.

The support of the stationary member of the No. 1 seal ("seal ring") is such as to allow large deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearing, the shaft could not deflect far enough to cause opening of the controlled leakage gap. The "spring-rate" of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Recent testing of pumps with the No. 1 seal entirely removed (full reactor pressure on the No. 2 seal) shows that relatively small leakage rates would be maintained for long periods of time (approximately 100 hours) even if the No. 1 seal fails entirely. The plant operator is warned of this condition by the increase in No. 1 seal leakoff and has time to close this line, and to conduct a safe plant shutdown without significant leakage of reactor coolant to the containment. Thus, it may be concluded that gross leakage from the pump does not occur, even if seals were to suffer physical damage.

The effect of loss of offsite power on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the cooling water for seal and bearing cooling. The emergency diesel generators are started automatically due to loss of offsite power so that component cooling water flow and seal water injection flow are automatically restored.

In the event of a loss of all seal cooling, reactor coolant begins to travel along the RCP shaft and displace the cooler seal injection water. The shutdown seal (SDS) actuates once the No. 1 seal package temperature reaches the SDS actuation temperature. SDS actuation limits the loss of reactor coolant through the RCP seal package.

Critical Speed

Both the damped and lateral natural frequencies of the RC pumps are determined by establishing a number of shaft sections and applying weights and moments of inertia for each section bearing spring and damping data. The torsional natural frequencies are similarly determined. The lateral and torsional natural frequencies are greater than 120 percent and 110 percent respectively of the running speed.

Coastdown Capability

It is important to reactor operation that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow in a station blackout condition, each reactor coolant pump is provided with a flywheel. Thus, the rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow.

Bearing Integrity

The design requirements for the reactor coolant pump 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 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.

Low oil levels in the motor bearings signals an alarm in the control room and require shutting down of the pump. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from loss of oil, is indicated and alarmed in the control room as a high bearing temperature. This, again, requires pump shutdown. Even if these indications are ignored, and the bearing proceeded to failure, the low melting point of Babbitt metal on the pad surfaces ensures that no sudden seizure of the bearing occurs. In this event the motor continues to drive, as it has sufficient reserve capacity to operate, even under such conditions. However, it demands excessive currents and at some stage is shut down because of high current demand.

The reactor coolant pump shaft is designed so that its critical speed is well above the operating speed.

Locked Rotor

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 fails in torsion just below the coupling to the motor, disengaging the flywheel and motor from the shaft.

This constitutes a loss-of-coolant flow in the loop. Following such a postulated seizure, the motor continues to run without any overspeed, and the flywheel maintains its integrity, as it is still 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 is precluded by graphite in the bearing. Any seizure in the seals results in a shearing of the anti-rotation pin in the seal ring. An inadvertent early actuation of the shutdown seal on the shaft sleeve, with the shaft still rotating, will not adversely affect RCP coastdown and 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 are initially by high temperature signals from the bearing water temperature

detector, excessively high No. 1 seal leakoff indications, and off-scale No. 1 seal leakoff indications, respectively. Following these signals, pump vibration levels are checked. Excessive vibration indicates mechanical trouble and the pump is shut down for investigation.

Shaft Seal Leakage

During normal operation, leakage along the reactor coolant pump shaft is controlled by three shaft seals arranged in series such that reactor coolant leakage to the containment is essentially zero. Charging flow is directed to each reactor coolant pump via a 5 micron seal water injection filter. It enters the pumps through the thermal barrier flange and flows down to the thermal barrier heat exchanger. Here the flow splits and a portion enters the RCS via the pump thermal barrier heat exchanger. The remainder flows up the pump shaft (cooling the lower bearing) and leaves the pump via the No. 1 seal where its pressure is reduced to that of the volume control tank. The water from each pump seal assembly is piped to a common manifold and then via a seal water filter through a seal water heat exchanger where the temperature is reduced to about that of the volume control tank. Leakage past the No. 1 seal provides a constant pressure on the No. 2 seal and leakage past the No. 2 seal provides a constant pressure on the No. 3 seal. A standpipe is provided to ensure a pressure of at least 7 feet of water on the No. 3 seal and warn of excessive No. 2 seal leakage. The first outlet from the standpipe has an orifice to permit normal No. 2 seal leakage to flow to the reactor coolant drain tank; excessive No. 2 leakage results in a rise in the standpipe level and eventually overflow to the primary drains transfer tank No. 1 via a second overflow connection.

In the event of a loss of all seal cooling, reactor coolant begins to travel along the RCP shaft and displace the cooler seal injection water. The shutdown seal (SDS) actuates once the No. 1 seal package temperature reaches the SDS actuation temperature. SDS actuation limits the loss of reactor coolant through the RCP seal package.

### 4.3.2 Balance of Interconnected Systems

The principal systems which are interconnected with the RCS are the main steam and feedwater systems, the chemical and volume control system (CVCS), the safety injection system, and residual heat removal system. The RCS is dependent upon the steam generators, and the main steam, feedwater, and condensate systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350 F. The layout of the system ensures a natural circulation capability so as to permit station cooldown following a loss of all reactor coolant pumps.

The flow diagrams of the steam and power conversion system are shown in Section 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 supplies water to the steam generators in the event that the main feedwater pumps are inoperative. The system is described in Section 10.

The safety injection system is described in Section 6.3. The CVCS and residual heat removal system are described in Sections 9.1 and 9.3, respectively.

### 4.3.3 System Integrity

The range of coolant temperature variation during normal operation is limited and the associated reactivity change is well within the capability of the rod control group movement.

For design evaluation, the heatup and cooldown transients are analyzed by using a rate of temperature change equal to 100 F per hour which corresponds to abnormal or emergency heatup and cooldown conditions. Over certain temperature ranges, fracture prevention criteria will impose a lower limit to heatup and cooldown rates.

For Unit cooldown, the boron concentration in the RCS is increased as required to satisfy the shutdown margin requirements of Technical Specifications and the concentration is verified by sampling.

It is therefore concluded that the temperature changes imposed on the RCS during its normal modes of operation do not cause any abnormal or unacceptable reactivity changes.

The design cycles as discussed in the preceding section are conservatively estimated for equipment design purposes and are not intended to be an accurate representation of actual transients or for all cases reflect operating experience.

Certain design transients, with an associated pressure and temperature curve, have been chosen and assigned an estimated number of design cycles for the purpose of equipment design. These curves represent an envelope of pressure and temperature transients on the RCS boundary with margin in the number of design cycles chosen based on operating experience.

To illustrate this approach, the reactor trip transient can be mentioned. Four hundred design cycles are considered in this transient. One cycle of this transient would represent any operational occurrence which would result in a reactor trip. Thus, the reactor trip transient represents an envelope design approach to various operational occurrences.

This approach provides a basis for fatigue evaluation to ensure the necessary high degree of integrity for the RCS components.

System hydraulic and thermal design parameters are used as the basis for the analysis of equipment, coolant piping, and equipment support structures for normal and upset loading conditions. The analysis is performed using a static model to predict deformation and stresses in the system. Results of the analysis given six generalized force components, three bending moments, and three forces. These moments and forces are resolved into stresses in the pipe in accordance with the applicable codes. Stresses in the structural supports are determined by the material and section properties assuming linear elastic small deformation theory.

As part of the design control on materials, Charpy V-notch toughness tests are run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator and pressurizer to ensure that during hydrotesting and power operation these components operate in the ductile region at all times. In addition, drop-weight tests are performed and Charpy V-notch transition temperature curves are established for the reactor vessel materials.

As an assurance of system integrity, all components in the system are hydrotested at 3,107 psig prior to initial operation.

4.3.4 Pressure Relief

Signals from the pressurizer pressure control channels are used to control pressurizer spray, heaters and power-operated relief valves.

In the event of a complete loss of heat sink, i.e., no steam flow to the turbine, protection of the reactor coolant system against overpressure is afforded by pressurizer and steam generator safety valves along with any of the following reactor trip functions:

1. Reactor trip on turbine trip
2. High pressurizer pressure reactor trip
3. Overtemperature $\Delta T$ reactor trip
4. Low-low steam generator water level reactor trip.

These cases are discussed in Reference 7. A detailed functional description of the process equipment associated with the high pressure trip is provided in Reference 8.

The upper limit of overpressure protection is based upon the positive surge of the reactor coolant produced as a result of turbine trip under full load, assuming the core continues to produce full power. The self-actuated safety valves are sized on the basis of steam flow from the pressurizer to accommodate this surge at a setpoint of 2,500 psia and a total accumulation of 3 percent. Note that no credit is taken for the relief capability provided by the power operated relief valves during this surge.

System components whose design pressure and temperature are less than the RCS design limits are provided with overpressure protection devices and redundant isolation means. System discharge from overpressure protection devices is collected in the pressurizer relief tank in the RCS. Isolation valves are provided at all connections to the RCS.

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

4.3.5 System Incident Analysis

Analyses of system incidents are discussed in Section 14.

## References for Section 4.3


1. Fatigue Properties of Irradiated Pressure Vessels by Gibbon et. al., ASTM STP 426, pp. 408 to 437.

2. Deleted by Revision 23.

3. Deleted by Revision 23.

4. Deleted by Revision 23.

5. Deleted by Revision 23.

6. Deleted by Revision 23.

7. M. A. Mangan, "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Westinghouse Electric Corporation (October 1971).

8. J. A. Nay, "Process Instrumentation for Westinghouse Nuclear Steam Supply Systems," WCAP-7671, Westinghouse Electric Corporation (April 1971).

9. EPRI NP-2493, "Development of a Generic Procedure for Thermal Annealing of an Embrittled Reactor Vessel Using a Dry Annealing Method," Project 1021-1 (July 1982).

10. EPRI NP-2712, "Feasibility of and Methodology for Thermal Annealing an Embrittled Reactor Vessel," Project 1021-1 (November 1982).

11. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

12. EPRI Report TR-104788-R2, “PWR Primary-to-Secondary Leak Guidelines,” Revision 2, April 2000.

## 4.4 SAFETY LIMITS AND CONDITIONS

### 4.4.1 System Heatup and Cooldown Rates

Operating limits for the RCS with respect to heatup and cooldown rates are defined in the Licensing Requirements Manual.

### 4.4.2 Reactor Coolant Activity

Release of activity into the reactor coolant does not in itself constitute a hazard. The unit systems are designed to limit activity released to the environment to values well within applicable regulatory limits.

### 4.4.3 Maximum Pressure

The RCS serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the RCS is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the RCS is ensured. Thus, the safety limit of 2,735 psig (110 percent of design pressure) has been established. This represents the maximum transient pressure allowable in the RCS under the ASME Code, Section III. RCS pressure settings are given in Table 4.1-2.

### 4.4.4 System Limiting Conditions of Operation

These conditions for the RCS 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 Nondestructive Inspection of Material and Components

Table 4.5-1 summarizes the quality assurance program for all RCS components. In this table all of the non-destructive tests and inspections which are required by Westinghouse specifications on RCS components and materials are specified for each component. All 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 RCS are equivalent to those used for the reactor vessel.

Westinghouse requires, as part of its reactor vessel specification, that certain special tests be performed. These tests are listed below.

Ultrasonic Examinations

1. During fabrication angle beam inspection of 100 percent of plate material is performed to detect discontinuities that may be undetected by longitudinal wave examination, in addition to the design code straight beam ultrasonic test.

2. The reactor vessel is examined after hydro-testing to provide a base line map for use as a reference document in relation to later inservice inspections.

Penetrant Examinations

The partial penetration welds for the control rod drive mechanism head adaptor are inspected by dye penetrant after the first layer of weld metal, after each 1/4 inch of weld metal, and the final surface. Bottom instrumentation tubes are inspected by dye penetrant after each layer of weld metal. Core support block attachment welds are inspected by dye penetrant after first layer of weld metal and after each 1/2 inch of weld metal. This is required to detect cracks or other defects, lower the weld surface temperature, cleanliness, and prevent microfissures. After hydrostatic testing, all austenitic steel surfaces are 100 percent dye penetrant inspected.

Magnetic Particle Examination

1. All surfaces of quenched and tempered materials with the inside diameter inspected prior to cladding and the outside diameter inspected after hydrotesting. This serves to detect possible defects resulting from the forming and heat treatment operations.

2. The attachment welds for the vessel supports, lifting lugs, and refueling seal ledge are inspected after the first layer of weld metal and after each 1/2 inch of weld thickness. Where welds are back chipped, the areas are inspected prior to welding.

3. After hydrostatic testing, all ferretic steel surfaces are 100 percent magnetic particle examined.

Inservice Inspection

The full penetration welds in the following areas of the installed irradiated reactor vessel are available for visual and/or nondestructive inspection.

1. Vessel Shell - The inside surface

2. Primary coolant nozzles - The inside surface

3. Closure head - The inside and outside surface
   Bottom head - The outside surface

4. Field welds between the reactor vessel, nozzles, and the reactor coolant piping

5. Vessel flange seal surface.

During the design phase, careful consideration has been given to access provisions for both visual and nondestructive inservice inspections of the reactor coolant primary and associated auxiliary systems and components.

Specific provisions made for inspection access in the design of the reactor vessel, system layout and other major primary coolant components are as follows:

1. All reactor internals are completely removable. The tools and storage space required to permit reactor internals removal for these inspections are provided.

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

3. The reactor vessel cladding is improved in finish by grinding to the extent necessary to permit meaningful examination of the vessel welds and adjacent base metal in accordance with ASME XI.

4. The cladding to base metal interface is ultrasonically examined to assure satisfactory bonding to allow the volumetric inspection of the vessel welds and base metal from the vessel inside surface.

5. The reactor closure head is stored in a dry condition on the containment floor during refueling, allowing direct access for inspection.

6. The insulation on the vessel closure head is removable, allowing access for the visual examination of head penetrations.

7. All reactor vessel studs, nuts, and washers are removed to dry storage during refueling, allowing inspection in parallel with refueling operations.

8. Access holes are provided in the core barrel flange allowing access for the remote visual examination of the clad surface of the vessel without removal of the lower internals assembly.

9. Access holes in the reactor cavity water seal and reactor vessel cavity upper

neutron shield segments provide access for the surface and visual examination of the primary nozzle safe-end welds.

10. Manways are provided in the steam generator channel head to provide access for internal inspection.

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

12. The insulation covering all component and piping welds and adjacent base metal is designed for ease of removal and replacement in areas where external inspection is planned.

13. Removable plugs are provided in the primary shield concrete above the main coolant pumps to permit remova1 of the pump motor to provide internal inspection access to the pumps.

14. The primary loop compartments are designed to allow personnel entry during refueling operations, to permit direct inspection access to the external portion of piping and components.

The reactor vessel presents access problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for the periodic nondestructive tests which are required by the ASME inservice inspection code. These are:

1. Shop ultrasonic examinations are performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal from the inside surface. The size of cladding bounding defect allowed is 3/4 inch parallel to the weld.

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

3. After the shop hydrostatic testing, selected areas of the reactor vessel are ultrasonically tested and mapped to facilitate the inservice inspection program.

4.5.1.2 Irradiation Surveillance Program

In the surveillance programs, 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. These programs are 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, and is in accordance with ASTM-E-185-82, except that tensile specimen orientation will be in the axial direction instead of the hoop direction as shown in Figure 1 of ASTM-E-185-82 and the number, type and location of specimen conform to ASTM-E-185-73. 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-185-82 would not provide any additional information on which the operational limits for the reactor vessel are set. The surveillance program will not include correlation monitors. Correlation monitors were used in the past because of inadequate neutron dosimeters. Present neutron dosimeters included in the capsules can be used to measure exposure throughout the life of the reactor vessel. The reactor vessel surveillance program uses eight specimen capsules which meets the requirements of ASTM-E-185-73. The capsules are located in guide baskets welded to the outside of the thermal shield as depicted in Figure 4.5-2 about 3 inches from the vessel wall directly opposite the center portion of the core. Sketches of an elevation and plan view showing the 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 oriented in the principal rolling direction and normal (transverse) to the principal rolling direction from the limiting SA-533 Grade B class 1 shell plates located in the core region of the reactor and associated weld metal and heat affected zone metal.

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-Al, Cd shielded Co-Al, 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.

Each of four Capsules, (S, V, W, and Z) will contain the following specimens:

| Material | No. of Charpys | No. of Tensiles | No. of WOL's |
|---|---|---|---|
| Limiting Plate* | 8 | - | - |
| Limiting Plate** | 12 | 2 | 4 |
| Weld Metal | 12 | 2 | - |
| Heat affected Zone Metal | 12 | - | - |

Each of four additional capsules (T, U, X, and Y) will contain the following specimens:

| Material | No. of Charpys | No. of Tensiles | No. of WOL's |
|---|---|---|---|
| Limiting Plate* | 8 | - | - |
| Limiting Plate** | 12 | 2 | - |
| Weld Metal | 12 | 2 | 4 |
| Heat affected Zone Metal | 12 | - | - |

*Specimens oriented in the principal rolling direction

**Specimens oriented normal to the principal rolling direction

The following dosimeters and thermal monitors are included in each capsule:

Dosimeters

Pure Cu
Pure Fe
Pure Ni
CoAl (0.15% Co)
CoAl (Cadmium shielded)
$U^{238}$ (Cadmium shielded)
$Np^{237}$ (Cadmium shielded)

Thermal Monitors

97.5% Pb, 2.5% Ag (579½F Melting Point)
97.5% Pb, 1.72% Ag, 0.75% Sn (590½F Melting Point)

The chemistry and heat treatment of the surveillance material are provided in Table 4.5-2.

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall with the specimens being located between the core and the vessel and with the sequenced removal and reinsertion of capsules as noted in the tentative schedule. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the NDTT measurements are representative of the vessel at a later time in life. Data from fracture toughness samples (WOL) are expected to provide additional information for use in determining allowable stresses for irradiated material.

The reactor vessel material irradiation surveillance capsules are removed and examined, to determine changes in material properties, at the intervals shown in Table 4.5-3. The revised heatup and cooldown limit curves for normal operation are developed from these examinations and are provided in the Licensing Requirements Manual. These curves can also be developed using ex-vessel neutron dosimetry.

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 Appendix 4A.

For a detailed evaluation of the capsule results see the capsule analysis reports identified in the references for Section 4.5.

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 all capsules withdrawn.

Ex-Vessel Neutron Dosimetry System

In addition to the surveillance capsules within the reactor vessel as described above, supplemental neutron dosimetry is established through the use of ex-vessel neutron dosimetry.

Ex-vessel neutron dosimetry provides a verification of fast neutron exposure distributions within the reactor vessel wall beltline region and establishes a mechanism to enable long term monitoring of this portion of the reactor vessel. This neutron measurement system is located external to the reactor vessel, which allows for ease of dosimetry removal and replacement. The reactor vessel beltline region fluence information obtained by the system can be used to predict the shift in the reference nil ductility transition temperature ($RT_{NDT}$) and evaluate radiation damage in this region of the reactor vessel. Once internal surveillance capsules have been removed, and with the results of the neutron transport calculations, the ex-vessel neutron measurements allow the projection of embrittlement gradients through the reactor vessel with minimum uncertainty. Comprehensive sensor sets, including radiometric monitors, are employed at discrete locations within the reactor cavity to characterize the neutron energy spectrum variations axially and azimuthally over the beltline region of the reactor vessel. In addition stainless steel bead chains are used in conjunction with the encapsulated dosimeters to complete the mapping of the neutron environment between the discrete locations chosen for spectrum determinations.

The ex-vessel neutron dosimetry is installed in the annular air gap between the reactor vessel insulation and the neutron shield tank. The ex-vessel neutron dosimetry consists of aluminum dosimeter capsules (containing radiometric monitors) connected to and supported by stainless steel bead chains, which are supported by a tubular bracket attached to one of two support bars. Each support bar is supported from above by two support brackets that rest on the floor in the nozzle gallery and are mechanically affixed to the reactor vessel reflective insulation. At their bottoms the chain loops are mechanically secured with stainless steel threaded chain connectors to the grating of the platform under the reactor vessel. The ex-vessel neutron dosimetry measures fluence for approximately 1/8 of the vessel wall circumference. Neutron transport calculations then determine the fluence for the entire vessel beltline wall. Ex-vessel neutron dosimetry can be used to develop pressure and temperature limit curves without the need for internal surveillance capsules.

4.5.1.3 Electroslag Weld Quality Assurance

The 90 degree elbows used in the reactor coolant loop piping are electroslag welded. The following efforts were performed for quality assurance 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, $H_20$ Quench

c. 6 Transverse Tensile Bars - 2050°F, $H_20$ Quench + 750°F stress relief heat treatment

d. 6 Transverse Tensile Bars - 2050°F, $H_20$ Quench, tested at 650°F

e. 12 Guided Side Bend Test Bars.

2. The casting segments were surface conditioned for 100 percent radiographic and penetrant inspections. The acceptance standards were ASTM E-186[(2)] 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 percent 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 analysis 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 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 an 8 inch thick and from a 12 inch thick weldment and successfully tested for both the two wire and the three wire techniques, respectively. They are:

   a. Two wire electroslag process - 8 inch 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 inch thick weldment

      1) 6 Transverse Tensile Bars - 750°F post weld stress relief

      2) 7 Guided Side Bend Test Bars

3) 21 Charpy V-Notch Specimens

4) Full section macroexamination 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 two 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-restart-repaired region and successfully tested. They are:

1) 2 Transverse Tensile Bars - As Welded

2) 4 Guided Side Bend Test Bar

3) Full Section Macroexamination of Weld and Heat Affect Zone.

d. All of 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[3], obtained was between one-half of one percent to one percent. There were no discontinuities evident in any of the electroslag welds.

1) The casting segments were surface conditioned for 100 percent 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 thicknesses up to 4-1/2 inches and ASTM E-280[4], severity level 2 for section thicknesses greater than 4-1/2 inches. The penetrant acceptance standards were ASME Boiler and Pressure Vessel Code Section III, paragraph N-627.

2) The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Boiler and Pressure Vessel Code Section III, paragraph N-627.

3) The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100 percent radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section III, paragraph N-627.

4) Weld metal and base metal chemical and physical analyses were determined and certified.

5) Heat treatment furnace charts were recorded and certified.

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:

Heat Input vs Output

The heat input is determined by the product of volts times current and they are 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:

1. The thickness of the weld is in most cases the reason for changes.

2. The weld gap variation during the weld cycle will also require changes. For example, the procedure qualifications provide for welding thicknesses from 5 inches to 11 inches with two wires. The current and voltage are varied to accommodate this range.

3. 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 inch, or opening to perhaps as much as 1-1/2 inches. In either case, a change in current may be necessary.

4. The heat output is controlled by the heat sink of the section thickness and metered water flow through the water-cooled shoes. The nominal temperature of the discharged water is 100°F.

Weld Gap Configuration

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

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 inch and may vary in depth by plus or minus 3/8 inch without affecting the weld. This is measured by a stainless steel dipstick.

Weld Cross Section Configuration

It is noted that the higher the current or heat input and the lower the heat output that the dilution of weld metal with base metal is greater, causing 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.

Welder Qualification

Welder Qualification in accordance with ASME Boiler and Pressure Vessel Code, Section IX rules is required, using transverse side bend test specimens per Table Q.24.1.

### 4.5.1.4 Inservice Inspection Capability

Inservice inspection boundaries were established in accordance with the requirements of IS-120 of Section XI of the ASME Boiler and Pressure Vessel Code. Specific provisions were made as discussed in Section 4.5.1.1, to ensure compliance with the requirements of IS-141 and IS-142.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component.

As indicated above, the only sophisticated remote inspection equipment currently required is for inspection of the reactor vessel. The baseline inspection was performed by Westinghouse utilizing a remote reactor vessel ultrasonic inspection tool which will perform the code required inspection of the circumferential and longitudinal shell welds, the flange to vessel weld, the ligaments between the flange holes, the nozzle to vessel welds, and the nozzle to safe-end to pipe welds.

The vessel inspection tool used for preservice inspection has two major structural components, the center column assembly which is supported and aligned from the vessel internals support flange by the head assembly. These two structures form the rigid base assembly from which all the scanning arm and drive assemblies are mounted and operated. Two configurations of the tool can be arranged depending on the areas of the vessel to be examined. In one configuration the scanning arm and transducer array necessary to examine the vessel shell welds are mounted on the main carriage assembly. The main carriage assembly can traverse the full length of the vessel up and down the center column and also provide a 360 degree circumferential rotation around the inside of the vessel. In the second configuration, the nozzle scanning assembly is mounted beneath the main carriage and can be rotated for 360 degrees to provide alignment with any of the vessel nozzles. The tool, in this configuration, can be installed in the vessel without removal of the lower internals package. The scanning attachment for the examination of the vessel flange weld and ligaments can be installed on the tool in either of the two configurations described above. Figures 4.5-4 and 4.5-5 show the pre-service inspection tool in both these configurations.

For reactor vessel scanning, a data acquisition system is used which combines simplicity with operational reliability and adaptability. It employs an electronic system with a receiver or data channel for each transducer unit. The signal received from the transducer is transmitted through an electronic distance amplitude correction device and depending on the amplitude, gated through individual transigates to a printer. Simultaneously with the gated signal, information is received from the position indication encoder system which provides position coordinates. An audio/visual alarm is also triggered. The resultant indication is printed on a recording device together with its position reference.

Following the completion of the scan of all the areas scheduled for inspection, the tool operator(s) will, operating on manual control, return the transducer array to each of the indication position coordinates and fully evaluate the nature of the indication to sufficient extent to be able to determine the size, shape, and orientation. Special transducer arrays with variable angle search units were provided to assist in this activity during the pre-service inspections. The ongoing inservice inspections of the Reactor Vessel shall be done in accordance with the requirements of 10 CFR 50.55a(g), Inservice Inspection.

For the manual ultrasonic scanning of required components the licensee has reviewed and approved all inspection procedures to insure that all inspection results are recorded in a consistent and congruent manner to avoid ambiguity in interpretation.

The data from the various baseline and inservice inspections, collected in accordance with the applicable above-referenced procedures, have been collected into comprehensive reports tabulating all the results. The reports describe the scope of the inspection, the procedures utilized, the equipment utilized and names and qualifications of the personnel and all the examination results including all instrument calibration criteria in sufficient detail to ensure repeatability of each examination.

The only areas where it is expected that high radiation levels will prohibit the access of personnel for direct examination of component areas or systems, is the reactor vessel. The special design provisions and tooling required to perform the code required examinations in these areas have been discussed above. Westinghouse is carrying out a continuing program of radiation surveys during refueling programs in operating plants to ensure that any possible future problem areas are detected at an early stage. Should additional experience in the maintenance and inspection of operating plants indicate that other areas exist where access will be either limited or impossible, steps will be taken to develop any remotely operated inspection equipment considered necessary to meet the commitments of examinations defined in the Technical Specifications.

The detailed inservice inspection program is presented in the Beaver Valley Power Station Unit 1 Ten Year Plan detailing the ASME Section XI requirements. The preservice inspection program was based on ASME Section XI dated July 1, 1971. Both programs reflect a modified inspection program based on component accessibility and local radiation restrictions. The results from the various inspections and the changes in inspection technology will be evaluated and incorporated into the inspection program on a continuing basis to ensure the integrity of the systems scheduled for inspection.

## References for Section 4.5


1. S. E. Yanichko, S. L. Anderson, W. T. Kaiser, "Analyses of Capsule V from the Duquesne Light Company Beaver Valley Power Station Unit No. 1 Reactor Vessel Radiation Surveillance Program", WCAP-9860, Westinghouse Electric Corporation (January 1981).

2. "ASTM Heavy-Walled 2 to 4-1/2 inch (51 to 114 mm) Steel Castings", ASTM E-186, The American Society for Tests and Materials.

3. "ASTM Standard Recommended Practice for Radiographic Testing", ASTM E-94, The American Society for Tests and Materials.

4. "ASTM Heavy-Walled (4-1/2 to 12 inch (114 to 305 mm) Steel Castings", ASTM E-280, The American Society for Tests and Materials.

5. R. S. Boggs, S. L. Anderson, W. T. Kaiser, "Analysis of Capsule U from the Duquesne Light Company Beaver Valley Power Station Unit No. 1 Reactor Vessel Radiation Surveillance Program", WCAP-10867, Westinghouse Electric Corporation (September 1985).

6. S. E. Yanichko, S. L. Anderson, L. Albertin, "Analyses of Capsule W from the Duquesne Light Company Beaver Valley Power Station Unit No. 1 Reactor Vessel Radiation Surveillance Program", WCAP-12005, Westinghouse Electric Corporation (November 1988).

7. C. Brown, E. Terek, J. Conermann, R. Les Bencini, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-15571, Westinghouse Electric LLC (November 2000).

8. E. J. Long, E. T. Hayes, “Analysis of Capsule X from the FirstEnergy Nuclear Operating Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program”, WCAP-17896, Westinghouse Electric Company LLC (September 2014).

9. D. M. McNutt III, A. J. Markivich, “Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation”, WCAP-18102, Westinghouse Electric Company LLC (February 2018).

TABLES FOR SECTION 4

Table 4.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS*

| | | |
|---|---|---|
| Station Design Life, years | 40*** | |
| Number of Heat Transfer Loops | 3 | |
| Design Pressure, psig | 2,485 | |
| Nominal Operating Pressure, psig | 2,235 | |
| Total System Volume Including Pressurizer and Surge Line, $ft^3$ (ambient conditions) | 9,768 | |
| Total Heat Output (100% power, including pump heat) $10^6$ Btu/hr | 9,929 | |
| | High Tavg (580.0°F) | Low Tavg (566.2°F) |
| Reactor Vessel Coolant Temperature at Full Power: | | |
| Inlet, °F | 543.1 | 528.5 |
| Outlet, °F | 617.0 | 603.9 |
| Reactor Coolant Pump Suction Temperature at Full Power, °F | 542.8 | 528.2 |
| Coolant Temperature Rise in Vessel at Full Power, °F | 73.9 | 75.4 |
| Total Coolant Flow Rate, $10^6$ lb/hr | 99.3 | 101.1 |
| Steam Generator Tube Plugging, %** | 0 | 22 |
| Steam Pressure at Full Power, psia** | 783** | 623 |

* Parameter ranges are given, where appropriate, for high temperature and low temperature operating conditions, and are based on Tavg values of 580.0°F and 566.2°F, respectively.

** Range includes operation over Tavg range specified above, as well as the effect of up to 22% steam generator tube plugging.

*** The renewed facility operating license has extended the operating term of the plant to 60 years, but did not affect other parameters in this table.

Table 4.1-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS (PSIG)

| | |
|---|---|
| Design Pressure | 2,485 |
| Operating Pressure | 2,235 |
| Safety Valves | 2,485 |
| Power Relief Valves | 2,335 |
| Pressurizer Spray Valves (begin to open) | 2,260 |
| Pressurizer Spray Valves (full open) | 2,310 |
| High Pressure Trip | 2,385 |
| High Pressure Alarm | 2,310 |
| Low Pressure Trip | 1,920* |
| Low Pressure Alarm | 2,185 |
| Hydrostatic Test Pressure | 3,107 |
| Backup Heaters On | 2,210 |
| Proportional Heaters (begin to operate) | 2,250 |
| Proportional Heaters (full operation) | 2,220 |
| SIS Low Pressure Injection (Operational Setpoint) | 1,845 |

* Low Pressure Trip (Operational Setpoint) 1,945

Table 4.1-3

REACTOR VESSEL DESIGN DATA

| | |
|---|---|
| Design/Operating Pressure, psig | 2,485/2,235 |
| Hydrostatic Test Pressure, psig | 3,107 |
| Design Temperature, F | 650 |
| Overall Height of Vessel and Closure Head, ft, in. (Bottom head OD to top of control rod mechanism adapter) | 42, 7-3/16 |
| Thickness of Insulation, min, in. | 3 |
| Number of Reactor Closure Head Studs | 58 |
| Diameter of Reactor Closure Head Studs, in. | 6 |
| ID of Flange, in. | 149-9/16 |
| OD of Flange, in. | 184 |
| ID at Shell, in. | 157 |
| Inlet Nozzle ID, in. | 27-1/2 |
| Outlet Nozzle, ID, in. | 29 |
| Clad Thickness, min, in. | 1/4 |
| Lower Head Thickness, min, in. (base metal) | 5 |
| Vessel Belt-Line Thickness, min, in. (base metal) | 7-7/8 |
| Closure Head Thickness, in. | 6-3/16 |
| Reactor Coolant Inlet Operating Temperature, F | See Table 4.1-1 |
| Reactor Coolant Outlet Operating Temperature, F | See Table 4.1-1 |
| Reactor Coolant Flow, lb/hr | See Table 4.1-1 |
| Total Water Volume Below Core, $ft^3$ | 979.0 |
| Water Volume in Active Core Region, $ft^3$ | 540.0 |

Table 4.1-3 (CONT'D)

REACTOR VESSEL DESIGN DATA

| | |
|---|---|
| Total Water Volume to Top of Core, $ft^3$ | 2,331.0 |
| Total Reactor Vessel Water Volume, (with core and internals in place), $ft^3$ | 3,617 |

Table 4.1-4

PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATA

| | |
|---|---|
| Pressurizer | |
| Design/Operating Pressure, psig | 2,485/2,235 |
| Hydrostatic Test Pressure (cold), psig | 3,107 |
| Design/Operating Temperature, °F | 680/653 |
| Water Volume, full power, $ft^3$* | 840 |
| Steam Volume, full power, $ft^3$ | 560 |
| Surge Line Nozzle Diameter, in. | 14 |
| Shell OD, in. | 92.25 |
| Electric Heaters Capacity, kw | 1,364 |
| Heatup Rate of Pressurizer Using Heaters Only, °F/hr | 55 (approximately) |
| Design (Minimum) Spray Rate, gpm | 600 |
| Pressurizer Relief Tank | |
| Design Pressure, psig | 100 |
| Rupture Disk Release Pressure, psig | 85 ±5 percent |
| Design Temperature, °F | 340 |
| Normal Water Temperature, °F | 120 (Containment ambient) |
| Total Volume, $ft^3$ | 1,300 |
| Total Rupture Disk Relief Capacity, lb/hr at 100 psig | $1.02 \times 10^6$ |

*60% of net internal volume (@ full power Tavg of 580°F)

Table 4.1-5

STEAM GENERATOR DESIGN DATA*

| | | |
|---|---|---|
| Number of Steam Generators | | 3 |
| Design Pressure, reactor coolant/steam, psig | | 2,485/1,085 |
| Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig | | 3,107 |
| Design Temperature, reactor coolant/steam, F | | 650/600 |
| Total Heat Transfer Surface Area, $ft^2$ | | 54,500 |
| Heat Transferred, $10^6$ Btu/hr | | 3,310 |
| | High Tavg (580.0°F) | Low Tavg (566.2°F) |
| Reactor Coolant Flow, lb/hr | 33.1 | 33.7 |
| Steam Conditions at Full Load**, outlet nozzle: | | |
| Steam flow, $10^6$ lb/hr | 4.02/4.34 | 4.00/4.32 |
| Steam temperature, °F | 515.8 | 490.2 |
| Steam pressure, psia | 783 | 623 |
| Steam generator tube plugging, % | 0 | 22 |
| Maximum moisture carryover, wt % | 0.10 | 0.10 |
| Feedwater temperature, °F | 400/455 | 400/455 |
| Overall Height, ft-in. | | 67-9 |
| Shell OD, upper, in. | | 176.92 |
| Shell OD, lower, in. | | 135.5 |
| Number of U-tubes | | 3,592 |
| U-Tube Outer Diameter, in. | | 0.875 |
| Tube Wall Thickness, (nominal), in. | | 0.050 |
| Number of Manways | | 4 |
| Inside Diameter, in. | | 16 |
| Number of Handholes | | 6 |
| Inside Diameter, in. | | 6 |
| Number of Inspection Ports | | 2 |
| Inside Diameter, in. | | 4 |

TABLE 4.1-5 (CONT'D)

STEAM GENERATOR DESIGN DATA*

| | Rated Load** | No Load** |
|---|---|---|
| Reactor Coolant Water Volume, $ft^3$ | 956 to 1136 | 1136 |
| Primary Side Fluid Heat Content, $10^6$ Btu | 24.69 to 29.66 | 29.06 |
| Secondary Side Water Volume, $ft^3$ | 1836 to 2083 | 3419 |
| Secondary Side Steam Volume, $ft^3$ | 3797 to 3550 | 2214 |
| Secondary Side Fluid Heat Content, $10^6$ Btu | 49.52 to 57.40 | 92.17 |

* Quantities are for each steam generator.

** Design Data for Rated Load operation is provided for design ranges for reactor vessel average temperature, feedwater temperature, and steam generator tube plugging level. Design data for No Load operation is provided for 0% steam generator tube plugging level.

Table 4.1-6

REACTOR COOLANT PUMPS DESIGN DATA

| | |
|---|---|
| Number of Pumps | 3 |
| Design Pressure/Operating Pressure, psig | 2,485/2,235 |
| Hydrostatic Test Pressure (cold), psig | 3,107 |
| Design Temperature (casing), F | 650 |
| Rpm at Nameplate Rating | 1,190 |
| Operating Temperature (suction), F | See Table 4.1-1 |
| Developed Head, ft | 280 |
| Capacity, gpm | 88,500 |
| Seal Water Injection, gpm | 8 |
| Seal Water Return, gpm | 3 |
| Pump Discharge Nozzle ID, in. | 27 1/2 |
| Pump Suction Nozzle ID, in. | 31 |
| Overall Unit Height, ft, in. | 25, 5-1/4 |
| Water Volume, $ft^3$ | 81 |
| Pump-Motor Moment of Inertia, $lb\text{-}ft^2$ | 82,000 |
| Motor Data: | |
| Type | AC Induction Single Speed, Air Cooled |
| Voltage | 4,160 |
| Insulation Class | F Thermalastic Epoxy |
| Phase | 3 |
| Frequency, Hz | 60 |
| Starting | |
| Current, amp | 4,800 |
| Input (hot reactor coolant), kW | 4,407 |
| Input (cold reactor coolant), kW | 5,852 |
| Power, hp (nameplate) | 6,000 |
| Pump Weight, lb (dry) | 174,300 |

Table 4.1-7

REACTOR COOLANT PIPING DESIGN PARAMETERS

| | |
|---|---|
| Reactor Inlet Piping, ID, in. | 27-1/2 |
| Reactor Inlet Piping, nominal thickness, in. | 2.69 |
| Reactor Outlet Piping, ID, in. | 29 |
| Reactor Outlet Piping, nominal thickness, in. | 2.84 |
| Coolant Pump Suction Piping, ID, in. | 31 |
| Coolant Pump Suction Piping, nominal thickness, in. | 3.02 |
| Pressurizer Surge Line Piping, ID, in. | 11.188 |
| Pressurizer Surge Line Piping, nominal thickness, in. | 1.406 |
| Design/Operating Pressure, psig | 2,485/2,235 |
| Hydrostatic Test Pressure (cold), psig | 3,107 |
| Design Temperature, F | 650 |
| Design Temperature (pressurizer surge line), F | 680 |
| Water Volume (all 3 loops including surge line), $ft^3$ | 1099 |
| Design Pressure, Pressurizer Relief Lines, (From Pressurizer to Safety Valve), psig | 2,485 @ 680 F |
| Design Temperature, Pressurizer Relief Lines, (From Safety Valve to Pressurizer Relief Tank), F | 600 @ 600 F |

Table 4.1-8

PRESSURIZER VALVES DESIGN PARAMETERS

| | |
|---|---|
| Pressurizer Spray Control Valves | |
| Number | 2 |
| Design Pressure, psig | 2,485 |
| Design Temperature, F | 650 |
| Design Flow (minimum) For Valves Full Open, each, gpm | 300 |
| Fluid Temperature, F | 543.5 |
| Pressurizer Safety Valves | |
| Number | 3 |
| Relieving Capacity (rated), lb/hr | 345,000 Each |
| Set Pressure, psig | 2,485 |
| Fluid | Saturated Steam |
| Backpressure: | |
| Normal, psig | 3 to 5 |
| Expected During Discharge, psig | 500 |
| Pressurizer Power Relief Valves | |
| Number | 3 |
| Design Pressure, psig | 2,485 |
| Design Temperature, F | 650 |
| Relieving Capacity, maximum at 2,350 psig, lb/hr | 198,000 Each |
| Fluid | Saturated Steam |

Table 4.1-9

REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

| | Pressure Drop, psi (estimated) |
|---|---|
| Across Pump Discharge Leg (including valve) | 3.31 |
| Across Vessel, including nozzle | 44.16 |
| Across Hot Leg (including valve) | 3.20 |
| Across Steam Generator | 32.60 |
| Across Pump Suction Leg | 3.15 |
| Total Pressure Drop | 86.42 |

Table 4.1-10

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

| Normal Conditions | Design Occurrences |
|---|---|
| Heatup and Cooldown at 100°F/hr (pressurizer cooldown 200°F/hr) | 139 (each)(2)(6) |
| Unit Loading and Unloading at 5 percent of full power/min | 18,300 (each)(1) |
| Step Load Increase and Decrease of 10 Percent of Full Power | 2,000 (each) |
| Large Step Load Decrease | 200 |
| Steady State Fluctuations | infinite |
| **Upset Conditions** | |
| Loss of Load, without immediate turbine or reactor trip | 80 |
| Loss of Power (blackout with natural circulation in the RCS) | 40 |
| Loss of Flow (partial loss of flow one pump only) | 80 |
| Reactor Trip from Full Power | 400(2) |
| Inadvertent Auxiliary Spray | 10(2) |
| Operational Basis Earthquake | |
| Westinghouse analyzed components(5) (20 earthquakes of 20 cycles each) | 400 |
| Piping components (5 earthquakes of 10 cycles each) | 50 |
| RCS Cold Overpressurization | 10(2) |
| **Faulted Conditions** | |
| Main Reactor Coolant Pipe Break* | 1 |
| Steam Pipe Break | 1 |
| Steam Generator Tube Rupture | (1 included in upset condition "Reactor Trip from Full Power," above) |
| Design Basis Earthquake | 1 |

Table 4.1-10 (CONT'D)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

| | |
|---|---|
| <u>Test Conditions</u> | |
| Turbine Roll Test | 10 |
| Hydrostatic Test Conditions | |
| a. Primary Side | 5 |
| b. Secondary Side | 5 |
| c. Primary Side Leak Test | 50 |
| <u>Supplemental Transients</u> | |
| Pressurizer Insurge | Varies(2)(3) |
| Selected CVCS Transients | Varies |
| a) Isolation of Charging Flow | 7,000 |
| b) Isolation of Letdown Flow | 1,500(8) |
| c) Placing Excess Letdown in Service | 7,000 |
| Safety Injection Actuation | 60(2)(4) |
| Auxiliary Feedwater Injections | 18,300 |
| RHR System Actuation | 600(7) |

* Leak-Before-Break assumed for design of component supports, reactor cavity water seal, upper reactor cavity neutron shield segments and pipe stress.

(1) Applicable to all RCS components except the Reactor Vessel Closure Studs, for which the applicable number of occurrences for the Unit Loading and Unloading transient is 10,400.

(2) Critical transient where the predicted count may approach the design count during the period of extended operation. Occurrences of this transient must be counted in accordance with Technical Specification 5.5.3.

(3) The design occurrences depend on specific attributes and severity of the actual transient.

(4) Evaluation has shown that detailed analysis could qualify approximately 110 cycles. An administrative limit of 60 cycles is imposed. Safety Injection Actuations are monitored to ensure the administrative limit on the number of its occurrences is not exceeded without additional evaluation.

Table 4.1-10 (CONT'D)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

(5) The reactor vessel and pressurizer are analyzed for 50 cycles of Operational Basis Earthquake.

(6) Analysis to address the effect of environmental fatigue for the pressurizer surge line to hot leg nozzle has assumed 139 plant heatup and cooldown events. This analytical limit is tracked as part of the Metal Fatigue Program described in Section 16.1.27.

(7) Analysis to address the effect of environmental fatigue for the RHR tee connection has assumed 600 occurrences of placing RHR in service. This analytical limit is tracked as part of the Metal Fatigue Program described in Section 16.1.27.

(8) Analysis to address the effect of environmental fatigue for the charging system nozzle has assumed 1500 occurrences of the Isolation of Letdown Flow transient. This analytical limit is tracked as part of the Metal Fatigue Program described in Section 16.1.27.

Table 4.1-11

REACTOR COOLANT SYSTEM CODE REQUIREMENTS

| Component | Code/Standard* | Code Class | Addenda | ASME Code Case |
|---|---|---|---|---|
| Reactor Coolant System | | | | |
| Reactor Vessel | ASME III, 68 | A | Thru W 68 | 1332-3<br>1335-2<br>1336<br>1514 |
| Reactor Vessel | ASME XI, 95 | A | Thru 96 | N-640 |
| Reactor Vessel Closure Head | ASME III, 89 | 1 | NA | 2142-1,<br>2143-1,<br>N-525,<br>N-474-1 |
| Full Length CRDM Housing | ASME III, 68 | A | Thru W 69 | NA |
| Steam Generator (tube side) | ASME III, 89 | 1 | NA | N-20-3<br>N-71-16<br>N-411-1<br>N-474-2 |
| (shell side) | ASME III, 89 | 2 | NA | NA |
| Reactor Coolant Stop Valves (Body Bonnet) | ASME III, 68 | A | Thru W 68 | NA |
| PRESSURIZER | ASME III, 65 | A | Thru W 66 | 1401 |
| Reactor Coolant Piping, Fittings and Fabrication | ANSI B31.1, 67 | NA | NA | NA |
| Surge Pipe, Fittings, and Fabrication | ANSI B31.1, 67 | NA | NA | NA |
| Loop Bypass Line | ANSI B31.1, 67 | NA | NA | NA |
| Reactor Coolant Narrow Range Temperature Detector Thermowells | ASME III | 1 | NA | NA |
| Reactor Coolant Wide Range Temperature Detector Thermowells | ANSI B31.1, 67 | NA | NA | NA |
| Safety Valves | ASME III | NA | Thru S 68 | NA |
| Relief Valves | ASA 16.5 | NA | NA | NA |

*ASA, ANSI: American National Standards Institute
(Under USAS B31.1-1967 there is no class as such)

ASME III: American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III

ASTM: American Society for Testing and Materials

TEMA: Tubular Exchangers Manufacturer's Association

NA - Not applicable

TABLE 4.1-11 Cont'd

REACTOR COOLANT SYSTEM CODE REQUIREMENTS

| Component | Code/Standard* | Code Class | Addenda | ASME Code Case |
|---|---|---|---|---|
| Valves to Reactor Coolant System Boundary | ASA 16.5, MSS-SP 66A | NA | NA | NA |
| Pressurizer Relief Tank | ASME VIII | NA | Thru S 68 | NA |
| CRDM Head Adapter Plugs | ASME III, 89 | NA | NA | NA |
| Reactor Coolant Pump Standpipe Orifice | No Code | NA | NA | NA |
| Reactor Coolant Pump Standpipe | ASME VIII | NA | NA | NA |
| Reactor Coolant Pump | | | | |
| Casing | ASME III, 68 | A | NA | 1355 |
| Main Flange | ASME III, 68 | A | NA | NA |
| Thermal Barrier | ASME III, 68 | A | NA | NA |
| No. 1 Seal Housing | ASME III, 68 | A | NA | NA |
| No. 2 Seal Housing | ASME III, 68 | A | NA | NA |
| Pressure Retaining | | | | |
| Bolting | ASME III, 68 | A | NA | NA |
| Remaining Parts | ASME III, 68 | A | NA | NA |
| Reactor Coolant Pump Motor | NEMA MG1, 67 | NA | NA | NA |
| Shaft Coupling | NEMA MG, 67 | NA | NA | NA |
| Armature | NEMA MG1, 67 | NA | NA | NA |
| Flywheel | ASTM A-533, Grade B, or equivalent,** Class 1 + E Spec Provisions | NA | NA | NA |
| Motor Bolting | NEMA MG1 | NA | NA | NA |
| Upper Oil Cooler | TEMA | C | NA | NA |
| Lower Oil Cooler | TEMA | C | NA | NA |

** The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 4.1-12

LOOP STOP VALVES

| | |
|---|---|
| Design/Normal Operating Pressure, psig | 2,485/2,235 |
| Hydrostatic Test Pressure Shop Test/ Preoperational Hydrostatic Test Pressure, psig | 3,350/3,107 |
| Design Temperature, F | 650 |
| Hot Leg Valve Size, nominal, in. | 29 |
| Cold Leg Valve Size, nominal, in. | 27.50 |
| Open/Close Travel Time, sec | 210 |

Table 4.2-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Reactor Coolant System - Operating Condition

| | |
|---|---|
| Conductivity | ~ 1 to 40 μmhos/cm at 25°C |
| pH | 4.2 to 10.5 at 25°C<br>(dependent upon boron concentration). |
| Oxygen | ≤ 0.10 ppm at RCS Temp. > 180°F<br>< 0.005 ppm for power operation<br>≤ 1.0 ppm for 24 hour transient |
| Chloride | ≤ 0.15 ppm steady state limit<br>≤ 1.50 ppm for 24 hour transient |
| Fluoride | ≤ 0.15 ppm steady state limit<br>≤ 1.50 ppm for 24 hour transient |
| Hydrogen | 25 to 50 cc/kg during power operation<br>(normal range 30 to 40 cc/kg). |
| Total suspended solids | 1.0 ppm |
| $Li^7OH$ | As a function of Boron Concentration:<br>≤ 4.20 ppm Li |
| Boric Acid | ~ 0 to 4000 ppm (Boron concentration of an isolated RCS loop to be maintained ≥ boron concentration of operating loops). |

Table 4.2-3

STEAM GENERATOR WATER (STEAM-SIDE) CHEMISTRY SPECIFICATION
(Normal Operation)

| PARAMETER | NORMAL VALUE | ACTION LEVEL |
|---|---|---|
| pH | ≥ 8.0 | |
| Cation Conductivity μmho/cm | ≤ 0.8 | |
| Sodium, ppb | ≤ 20 | See Chapter 9 of the BVPS-Chemistry Manual for a description of the Action Levels. |
| Chloride, ppb | ≤ 20 | |
| Silica, ppb | ≤ 300 | |
| Sulfate, ppb | ≤ 20 | |

Table 4.3-1

SUMMARY OF PRIMARY PLUS SECONDARY STRESS INTENSITY
FOR COMPONENTS OF THE REACTOR VESSEL

| Area | Calculated Stress Intensity (psi) | Allowable Stress Intensity (psi)* (at Operating Temperature) |
|---|---|---|
| Head Flange | 52,140 | 80,100 |
| Vessel Flange | 73,100 | 80,100 |
| Closure Studs | 104,440 | 118,800 |
| CRDM Housings | 50,650 | 69,900 |
| Outlet Nozzles | 58,540 | 80,100 |
| Inlet Nozzles | 51,670 | 80,100 |
| Vessel Wall Transition | 31,720 | 80,100 |
| Bottom Head to Shell Juncture | 36,530 | 80,100 |
| Core Support Guides | 44,070 | 69,900 |
| Instrumentation Tubes | 57,280 | 69,900 |

* As provided in Section III of ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

Table 4.3-2

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL

| Item | Usage Factor* |
|---|---|
| Head Flange | 0.055 |
| Vessel Flange | (1) |
| Closure Studs | 0.999 (2) |
| CRDM Housings | 0.065 |
| Outlet Nozzles | 0.051 |
| Outlet Nozzle Support Pads | 0.247 |
| Inlet Nozzles | 0.066 |
| Inlet Nozzle Support Pads | 0.039 |
| Vessel Wall Transition | 0.010 |
| Bottom Head to Shell Juncture | 0.005 |
| Core Support Guides | 0.060 |
| Instrumentation Tubes | 0.220 |

(1) This location is not limiting.
(2) This Fatigue Usage Factor for the Reactor Vessel Closure Studs is acceptable for no more than 10,400 occurrences of the Unit Loading and Unloading transient.

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

Table 4.3-3

STEAM GENERATOR STRUCTURAL ANALYSIS
LOAD COMBINATIONS

| Loading Conditions | External Mechanical Loads(4) | Pressure and Thermal Loads |
|---|---|---|
| Design | Deadweight + Pressure + OBE(1) | Design Pressures and Temperatures |
| Normal (Level A) | Deadweight + Pressure + Thermal | Pressure and Thermal |
| Upset (Level B) | Deadweight + Pressure + Thermal | Pressure and Thermal |
| Upset (Level B) | Deadweight + Pressure + Thermal + OBE | Pressure and Thermal |
| Faulted (Level D) | Deadweight + Pressure(1) | Pressure and Thermal |
| Faulted (Level D) | Deadweight + Pressure + DBE(1) | Pressure and Thermal |
| Faulted (Level D) | Deadweight + Pressure + DBE + Pipe Rupture(1)(2)(3) | Pressure and Thermal |
| Test | Deadweight + Pressure | Pressure and Temperature |

Notes:
(1) Inside the Code defined region of reinforcement, thermal pipe loads must be included in the analysis.
(2) Deadweight + Pressure + $[( |\text{Pipe Rupture}| - |\text{Pressure}| )^2 + (\text{DBE})^2 ]^{1/2}$.
(3) Pipe Rupture is either LOCA, Steamline Break, or Feedwater Line Break.
(4) Upper Shell Platform loadings must be included with external mechanical loads, as applicable.

Table 4.5-1

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
TYPE OF NDE TEST TO BE PERFORMED

| Component | RT* | UT* | PT* | MT* | ET* |
|---|---|---|---|---|---|
| 1. Steam Generator | | | | | |
| 1.1 Tubesheet | | | | | |
| 1.1.1 Forging | | yes | | yes | |
| 1.1.2 Cladding | | yes(+) | yes(++) | | |
| 1.2 Channel Head | | | | | |
| 1.2.1 Forging | | yes | | yes | |
| 1.2.2 Cladding | | yes | yes | | |
| 1.3 Secondary Shell and Head | | | | | |
| 1.3.1 Forging | | yes | | | |
| 1.4 Tubes | | yes | | | yes |
| 1.5 Nozzles (forgings) | | yes | | yes | |
| 1.6 Safe Ends | | yes | yes | | |
| 1.7 Weldments | | | | | |
| 1.7.1 Shell, Circumferential | yes | | | yes | |

Table 4.5-1 (CONT'D)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
TYPE OF NDE TEST TO BE PERFORMED

| Component | RT* | UT* | PT* | MT* | ET* |
|---|---|---|---|---|---|
| 1.7.2 Cladding (channel head-tubesheet joint cladding restoration) | | yes | yes | | |
| 1.7.3 Steam and Feed-water Nozzle to Shell | yes | | | yes | |
| 1.7.4 Support brackets | | | | yes | |
| 1.7.5 Tube to Tube-sheet | | | yes | | |
| 1.7.6 Instrument Connections (primary and secondary) | | | | yes | |
| 1.7.7 Temporary Attachments after Removal | | | | yes | |
| 1.7.8 After Hydro-static Test (all welds and complete channel head - where accessible) | | | | yes | |

Table 4.5-1 (CONT'D)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
TYPE OF NDE TEST TO BE PERFORMED

| Component | RT* | UT* | PT* | MT* | ET* |
|---|---|---|---|---|---|
| 1.7.9 Safe Ends to Nozzle | yes | | yes | | |
| 2. Pressurizer | | | | | |
| 2.1 Heads | | | | | |
| 2.1.1 Plates | 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 | |

Table 4.5-1 (CONT'D)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
TYPE OF NDE TEST TO BE PERFORMED

| Component | RT* | UT* | PT* | MT* | ET* |
|---|---|---|---|---|---|
| 2.5.2 Shell, Circumferential | yes | | | yes | |
| 2.5.3 Cladding | | | yes | | |
| 2.5.4 Instrument Connections | | | yes | | |
| 2.5.5 Support Skirt | | | | yes | |
| 2.5.6 Temporary Attachments after Removal | | | | yes | |
| 2.5.7 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 | | | | | |
| 3.1 Fittings and Pipe (castings) | yes | | yes | | |

Table 4.5-1 (CONT'D)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
TYPE OF NDE TEST TO BE PERFORMED

| Component | RT* | UT* | PT* | MT* | ET* |
|---|---|---|---|---|---|
| 3.2 Fittings and Pipe (forgings) | | yes | yes | | |
| 3.3 Weldments | | | | | |
| 3.3.1 Circumferential | yes | | yes | | |
| 3.3.2 Nozzle to Run-pipe (No RT for nozzles less than 4 in.) | yes | | yes | | |
| 3.3.3 Instrument Connections | | yes | 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 | | | |

Table 4.5-1 (CONT'D)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
TYPE OF NDE TEST TO BE PERFORMED

| Component | RT* | UT* | PT* | MT* | ET* |
|---|---|---|---|---|---|
| 4.3 Weldments | | | | | |
| 4.3.1 Circumferential | yes | | yes | | |
| 4.3.2 Instrument Connections | | | yes | | |
| 5. Reactor Vessel | | | | | |
| 5.1 Forgings | | | | | |
| 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 | | |
| 5.1.5 Instrumentation Tube | | yes | yes | | |
| 5.1.6 Main Nozzles | | yes | | yes | |
| 5.2 Plates | | yes | | yes | |
| 5.3 Weldments | | | | | |
| 5.3.1 Main Steam | yes | yes** | | yes | |

Table 4.5-1 (CONT'D)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
TYPE OF NDE TEST TO BE PERFORMED

| Component | RT* | UT* | PT* | MT* | ET* |
|---|---|---|---|---|---|
| 5.3.2 CRD Head Adapter Connection | | | yes | | |
| 5.3.3 Instrumentation Tube Connection | | | yes | | |
| 5.3.4 Main Nozzles | yes | yes** | | yes | |
| 5.3.5 Cladding | | yes(+++) | yes | | |
| 5.3.6 Nozzle-Safe Ends (if weld deposit) | yes | yes** | yes | | |
| 5.3.7 Head Adapter Forging to Head Adapter Tube | yes | | yes | | |
| 5.3.8 All Ferretic Weld Accessible after Hydrotest | | | | yes | |
| 5.3.9 All Non-ferretic Welds Accessible after Hydrotest | | | yes | | |
| 5.3.10 Seal Ledge | | | | yes | |
| 5.3.11 Head Lift lugs | | | | yes | |

Table 4.5-1 (CONT'D)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
TYPE OF NDE TEST TO BE PERFORMED

| Component | RT* | UT* | PT* | MT* | ET* |
|---|---|---|---|---|---|
| 5.3.12 Core Pad Welds | | yes | yes | yes | |
| 6. Valves | | | | | |
| 6.1 Castings | yes | | yes | | |
| 6.2 Forgings (no UT for valves two inches and smaller) | | yes | yes | | |

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current
** UT - Map for Section XI

(+) Flat Surfaces Only
(++) Weld Deposit Areas Only
(+++) UT of Clad Bond-to-Base Metal
(++++) Or a UT and ET

Table 4.5-2

CHEMICAL ANALYSES (PERCENT) AND HEAT TREATMENT
OF SURVEILLANCE MATERIAL

Chemical Analyses (Percent)

| Element* | Plate B6093-1 | As Deposited Weld Metal |
|---|---|---|
| C | 0.20 | 0.110 |
| Si | 0.180 | 0.270 |
| Mo | 0.550 | 0.480 |
| Cu | 0.200 | 0.260 |
| Ni | 0.540 | 0.620 |
| Mn | 1.310 | 1.370 |
| Cr | 0.140 | 0.015 |
| V | 0.001 | 0.001 |
| Co | 0.014 | 0.014 |
| S | 0.015 | 0.006 |
| P | 0.010 | 0.018 |
| Al | 0.028 | 0.010 |
| $N_2$ | 0.004 | 0.014 |
| Sn | 0.010 | 0.008 |

*Elements not listed are less than 0.01 weight percent.

Heat Treatment

| | |
|---|---|
| Lower shell plate B6903-1: | 1550/1650 -- 4 hours, Water quenched<br>1225°F ± 25°F -- 4 hours, Air cooled<br>1150°F ± 25°F -- 40 hours, Furnace cooled |
| Weldment | 1150°F ± 25°F -- 15 hours, Furnace cooled |

Table 4.5-3

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

| Capsule | Current (Original) Capsule Location | Lead Factor[(a)] | Withdrawal EFPY[(b)] | Fluence[(c)] [n/cm$^2$, E > 1.0 MeV] |
|---|---|---|---|---|
| V | 165° | 1.47 | 1.2 | $2.97 \times 10^{18}$ |
| U | 65° | 1.00 | 3.6 | $6.18 \times 10^{18}$ |
| W | 245° | 1.05 | 5.9 | $9.52 \times 10^{18}$ |
| Y | 295° | 1.14 | 14.3 | $2.10 \times 10^{19}$ |
| X | 285° | 1.57 | 26.6 | $4.99 \times 10^{19}$ |
| S[(d)] | 285° (45°/295°) | 0.74[(d)] | Note[(d)] | $2.58 \times 10^{19}$[(d)] |
| T[(e)] | 65° (55°) | 0.94[(e)] | Note[(e)] | $3.28 \times 10^{19}$[(e)] |
| Z[(f)] | 165° (305°) | 1.20[(f)] | Note[(f)] | $4.18 \times 10^{19}$[(f)] |

Notes:

a) Lead factor from WCAP-18102-NP, Revision 0, Table 2-12.

b) Effective full power years (EFPY) from plant startup. Changes to this column will require prior NRC approval (except to indicate that capsules have been removed as specified in Section III.B.3, Appendix H of 10 CFR 50).

c) Fluence from WCAP-18102-NP, Revision 0, Table 2-11.

d) Capsule S was moved to Capsule Y location at the end of Cycle 19, and then moved to the Capsule X location at the end of Cycle 22. Reported fluence value and lead factor are accumulated through the end of Cycle 24. Capsule S should remain in the reactor. If additional metallurgical data is needed for BVPS-1, such as in support of a second license renewal to 80 total years of operation, withdrawal and testing of capsule S should be considered.

e) Capsule T was moved to Capsule U location at the end of Cycle 10. Reported fluence value and lead factor are accumulated through the end of Cycle 24. Capsule T should remain in the reactor and continue to accrue irradiation for potential future testing, if needed.

f) Capsule Z was moved to Capsule V location at the end of Cycle 10. Reported fluence value and lead factor are accumulated through the end of Cycle 24. Based on the current information, Capsule Z should be withdrawn after 39 EFPY, which corresponds to the peak vessel fluence at 1 of 1 the end of license extension (50 EFPY), $5.89 \times 10^{19}$ n/cm$^2$ (E > 1.0 MeV).

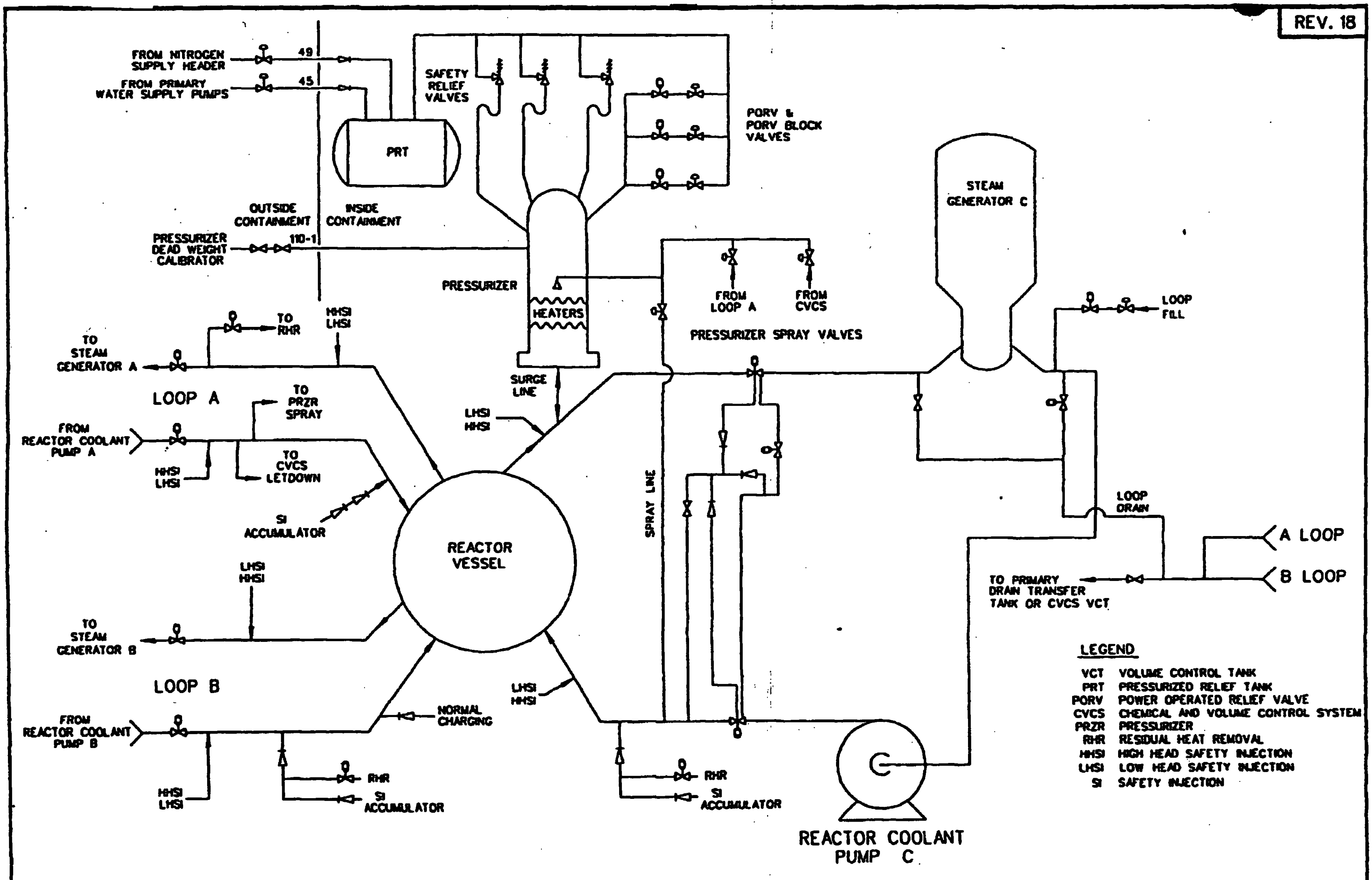



FIGURE 4.1-1
REACTOR COOLANT SYSTEM

REFERENCE: STATION DRAWING RM-37
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 18

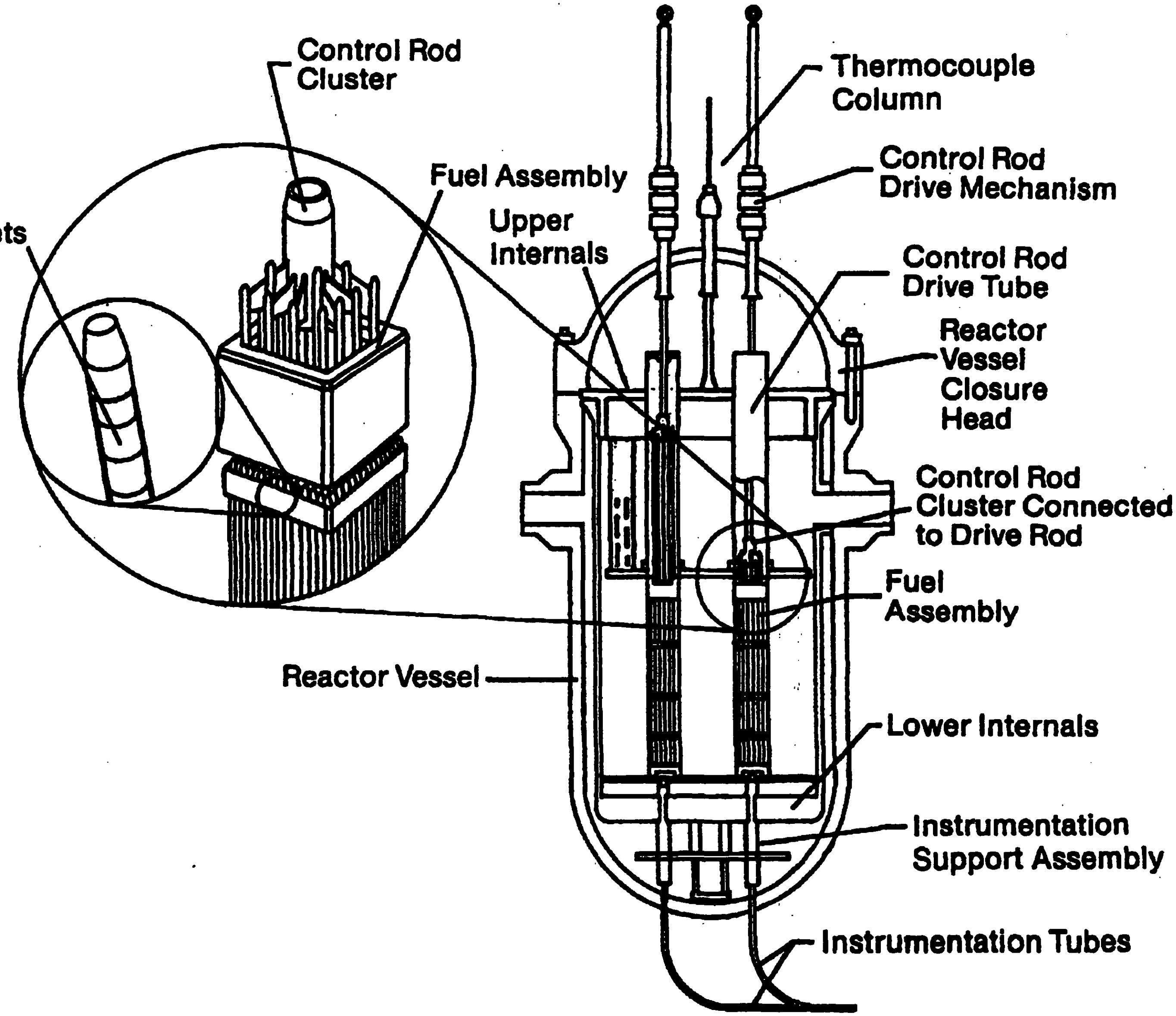



FIGURE 4.2-1
REACTOR VESSEL SCHEMATIC

BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

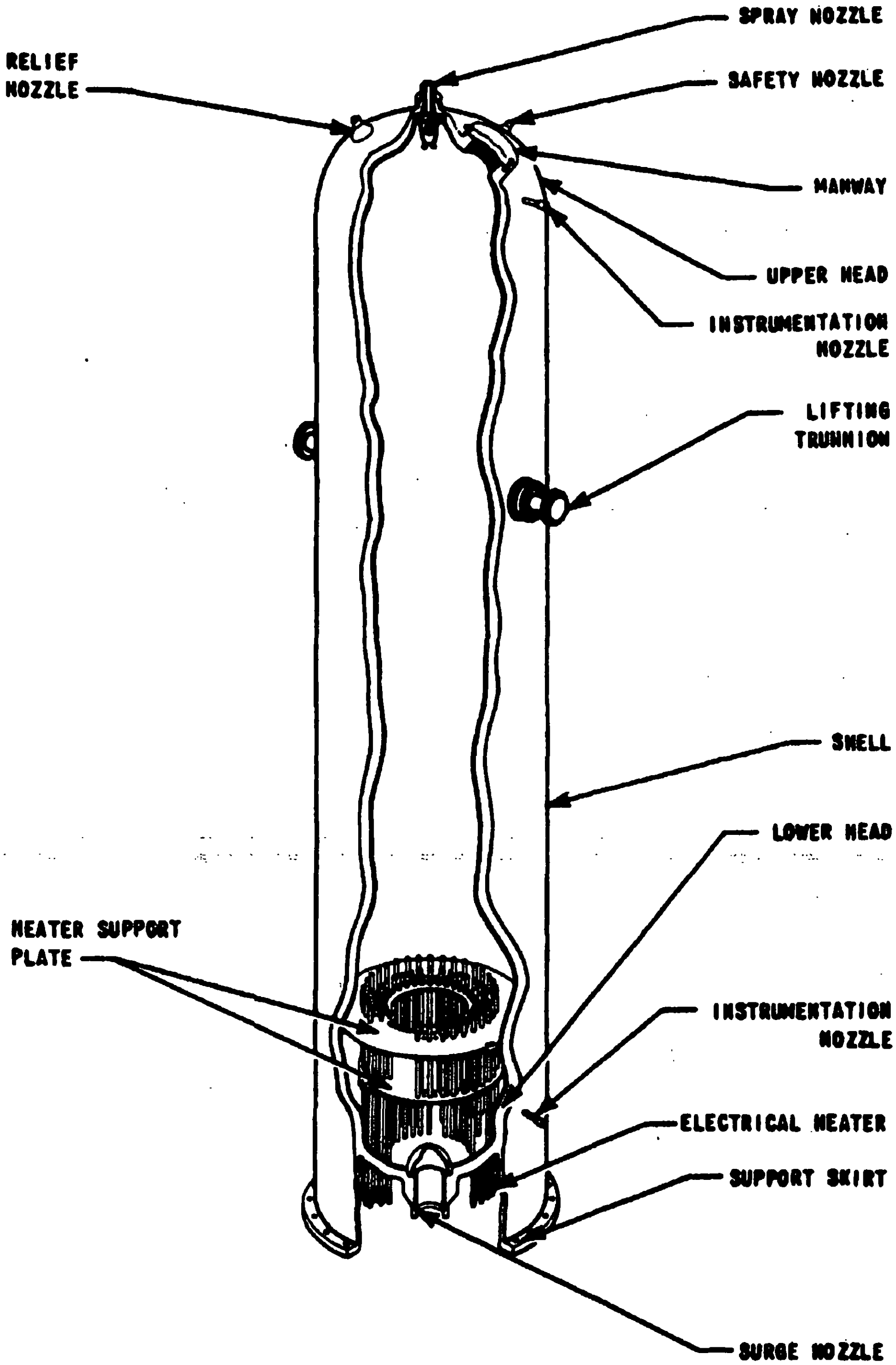



FIGURE 4.2-2
PRESSURIZER
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 23

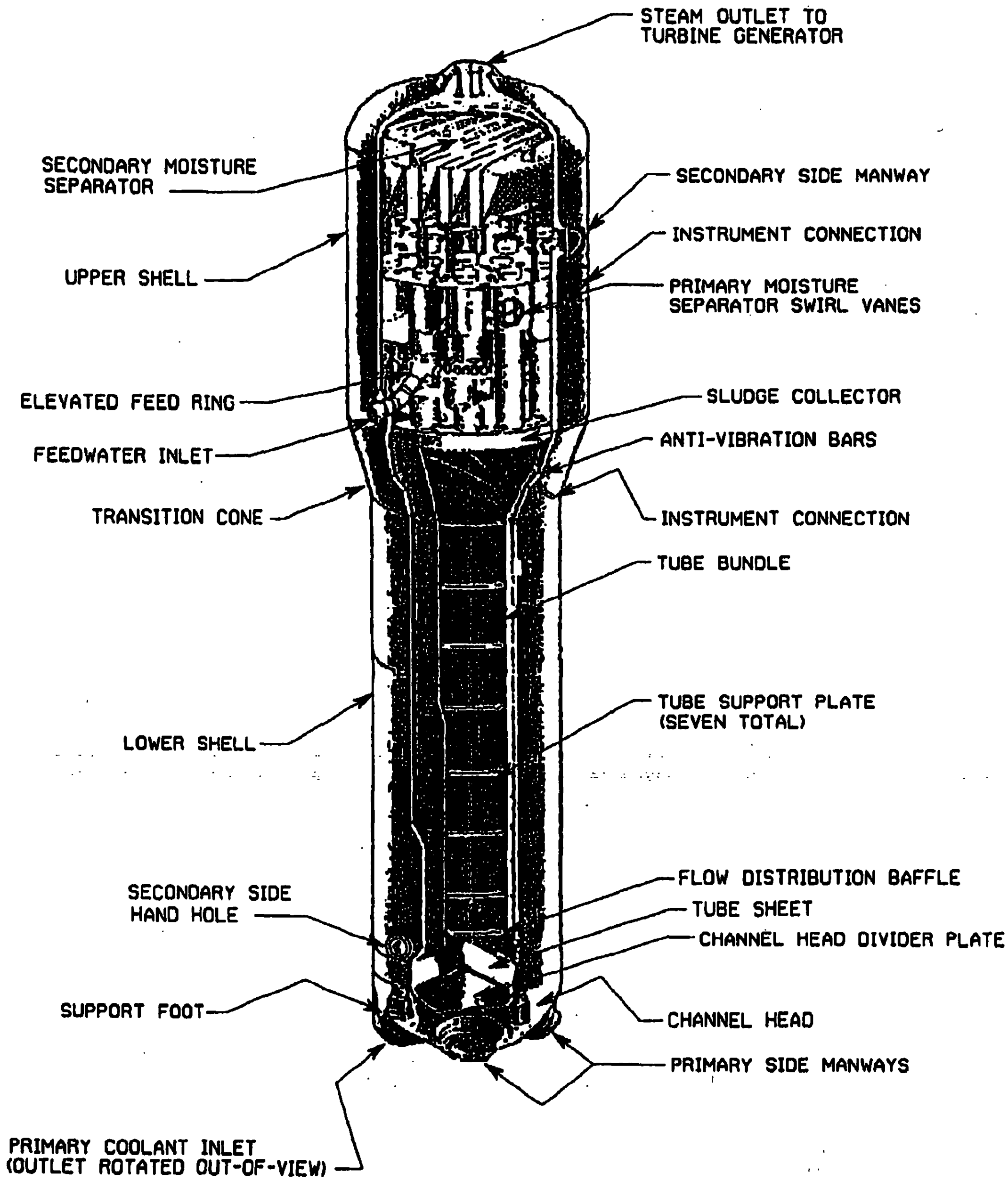



FIGURE 4.2-3

STEAM GENERATOR
WESTINGHOUSE MODEL 54F

BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

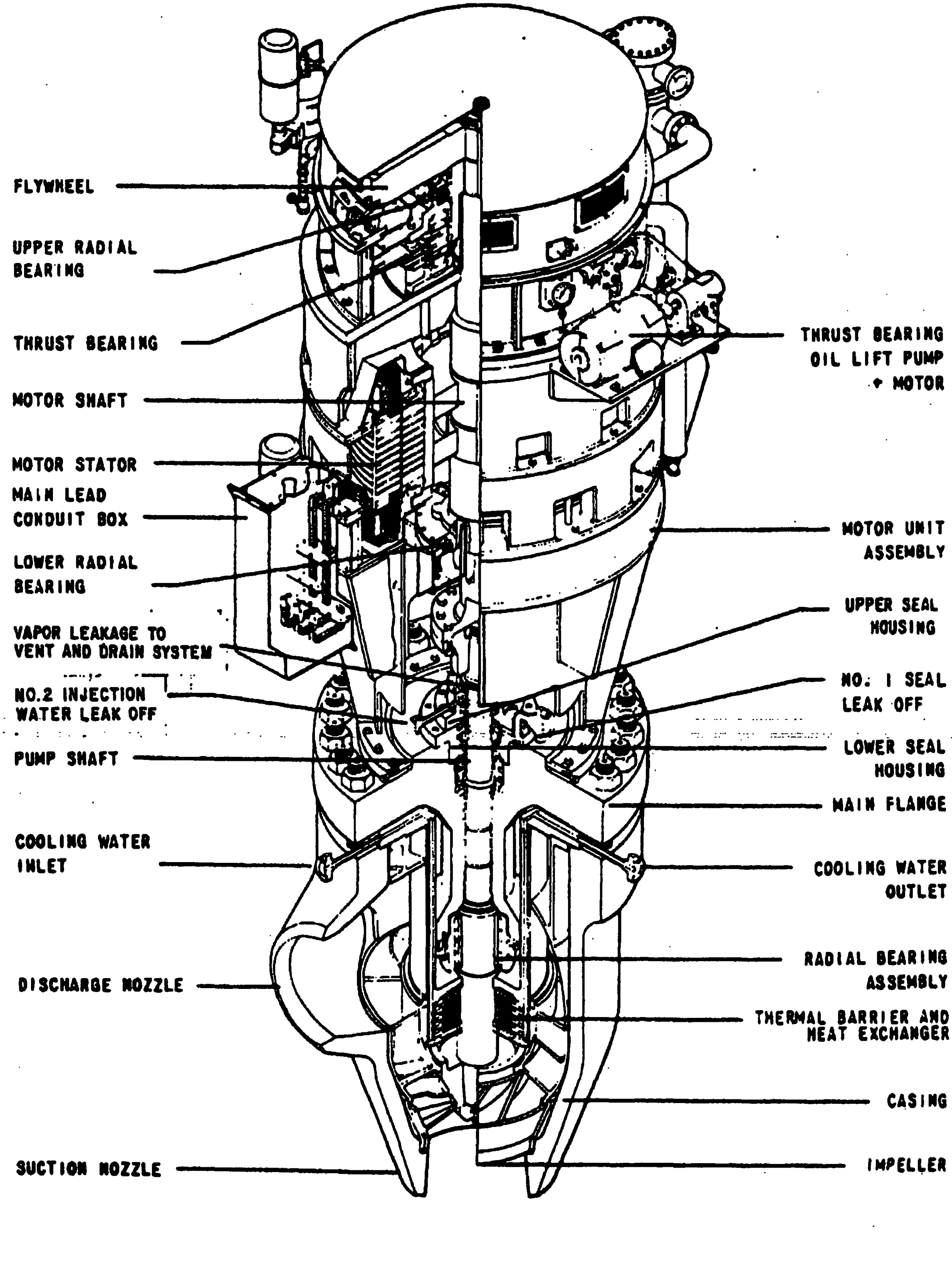
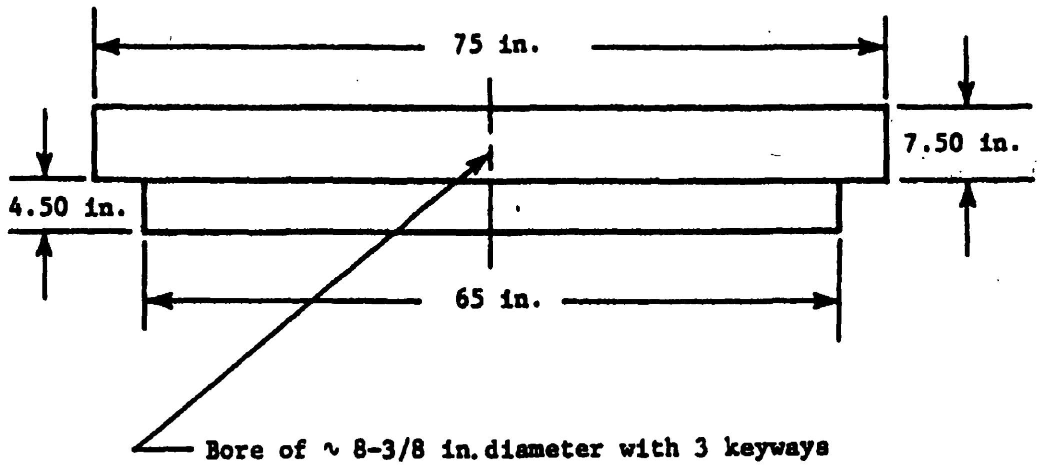



**FIGURE 4.2-4**
**REACTOR COOLANT PUMP**
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT




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

FIGURE 4.2-5
REACTOR COOLANT
PUMP FLYWHEEL
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

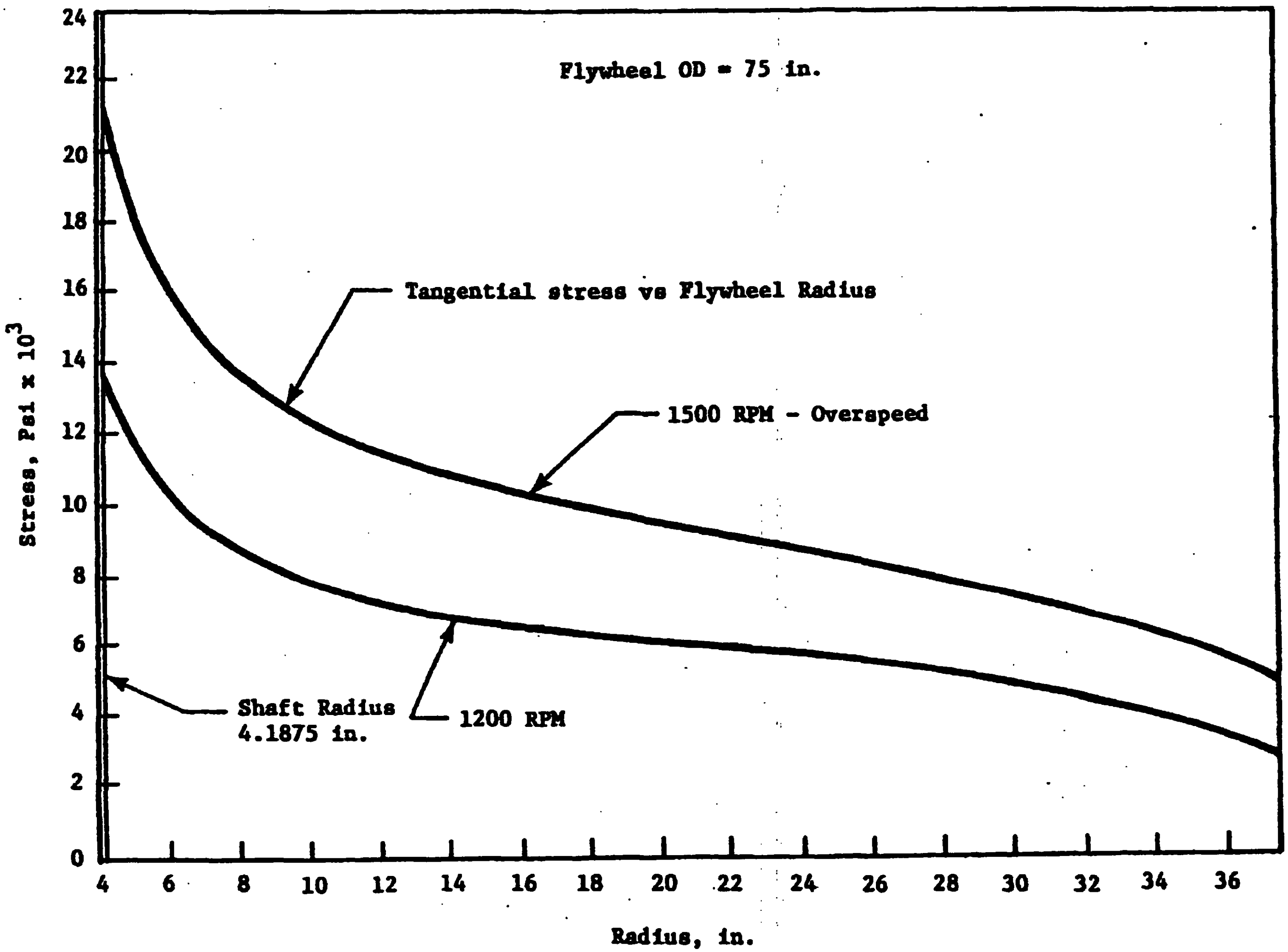
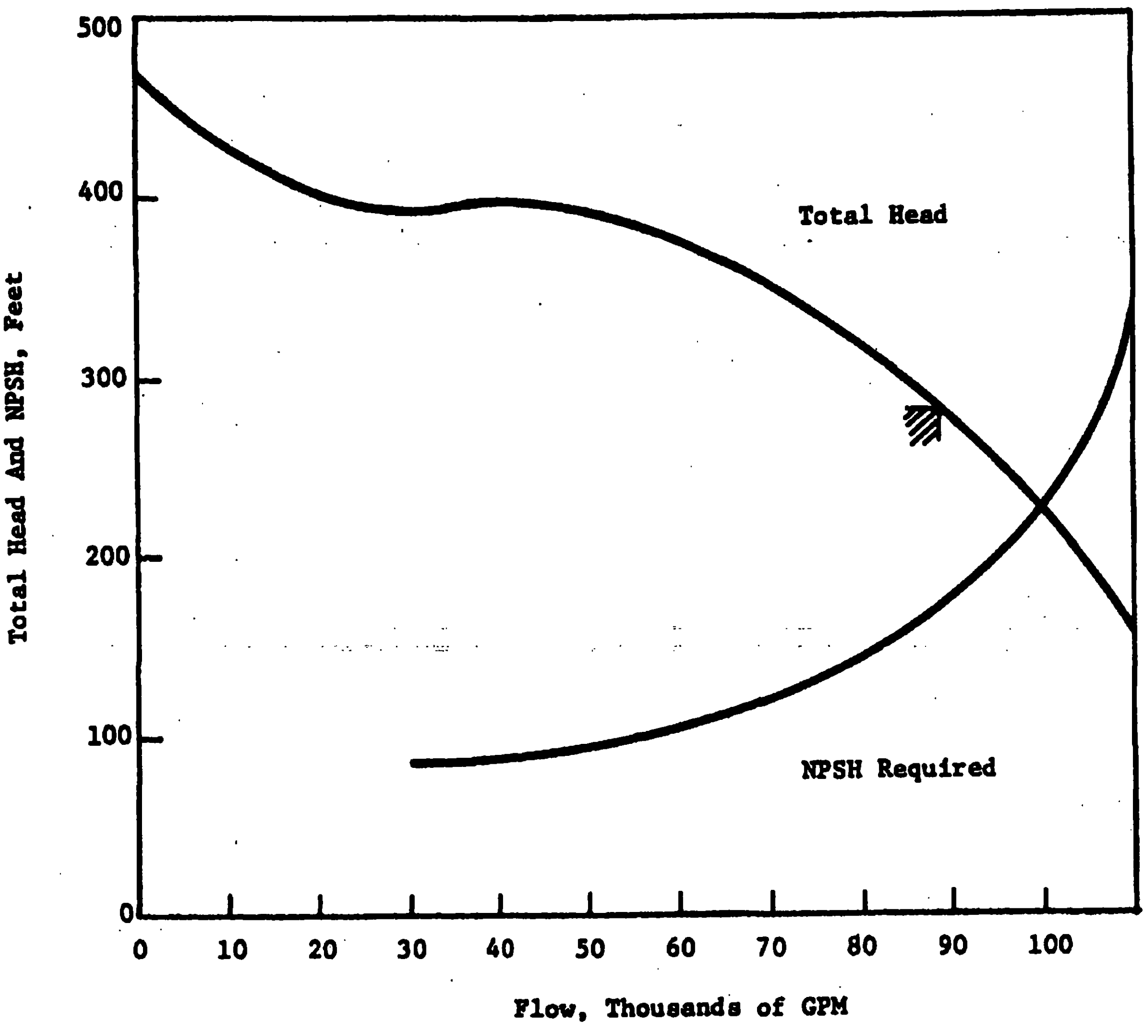



FIGURE 4-2-6
FLYWHEEL CHARACTERISTICS CURVE
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT




Design Point = 280/88,500

FIGURE 4.2-7
REACTOR COOLANT PUMP
PERFORMANCE CHARACTERISTICS

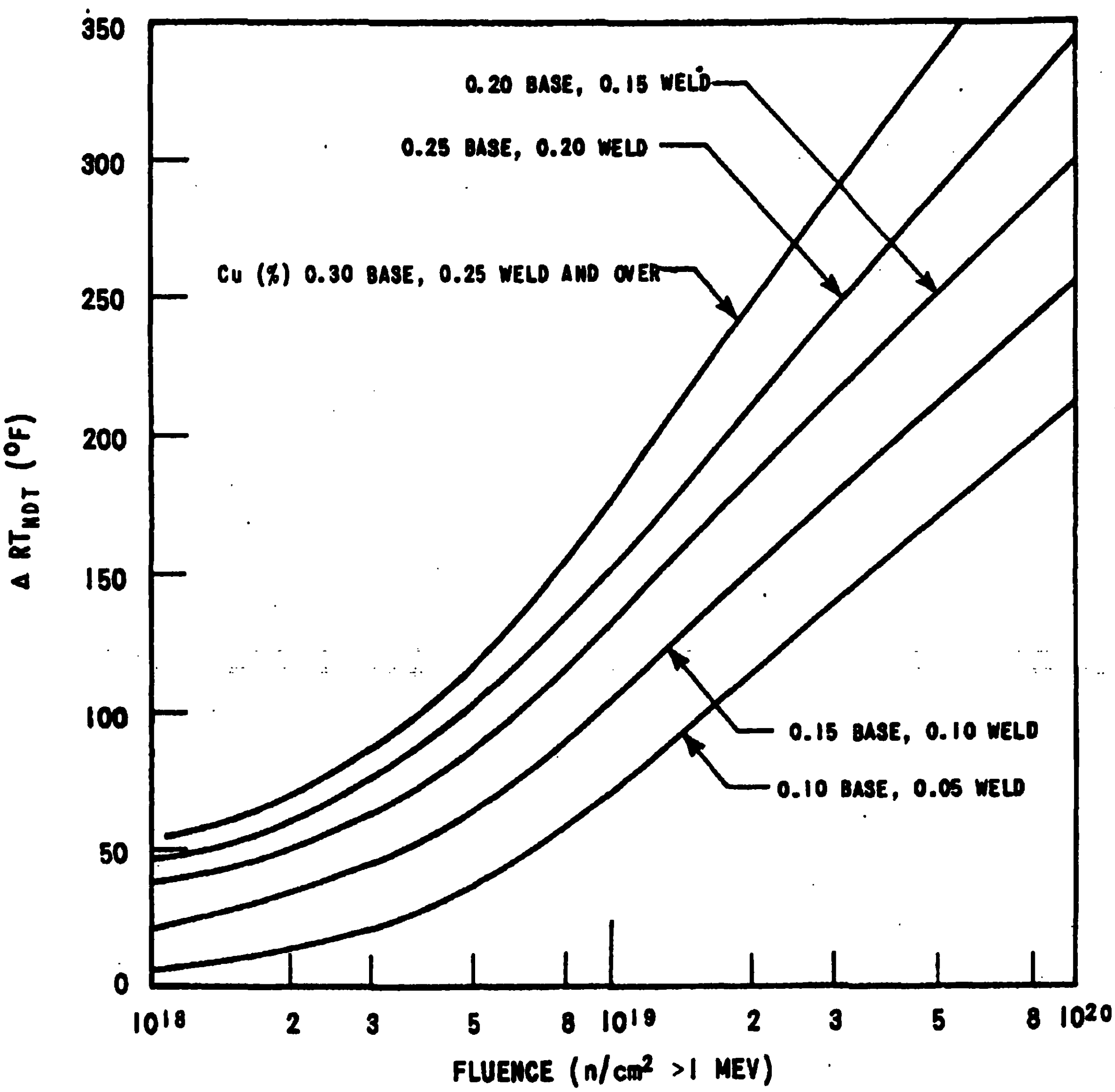



FIGURE 4.2-8
EFFECT ON FLUENCE AND COPPER CONTENT ON SHIFT OF $RT_{NDT}$ FOR REACTOR VESSEL STEELS EXPOSED TO 550 F TEMPERATURE
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

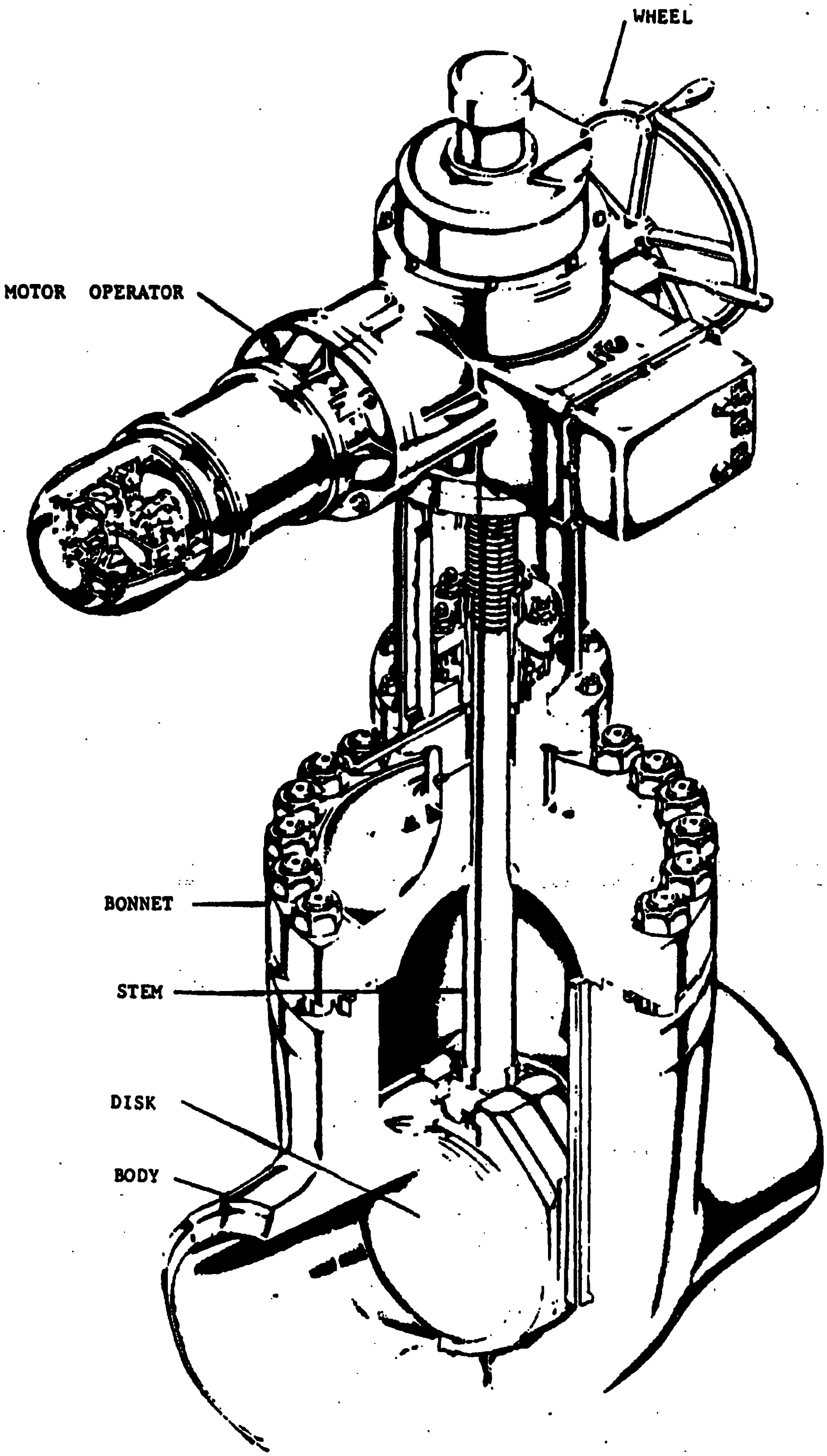



FIGURE 4.2-9
REACTOR COOLANT LOOP STOP VALVE
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

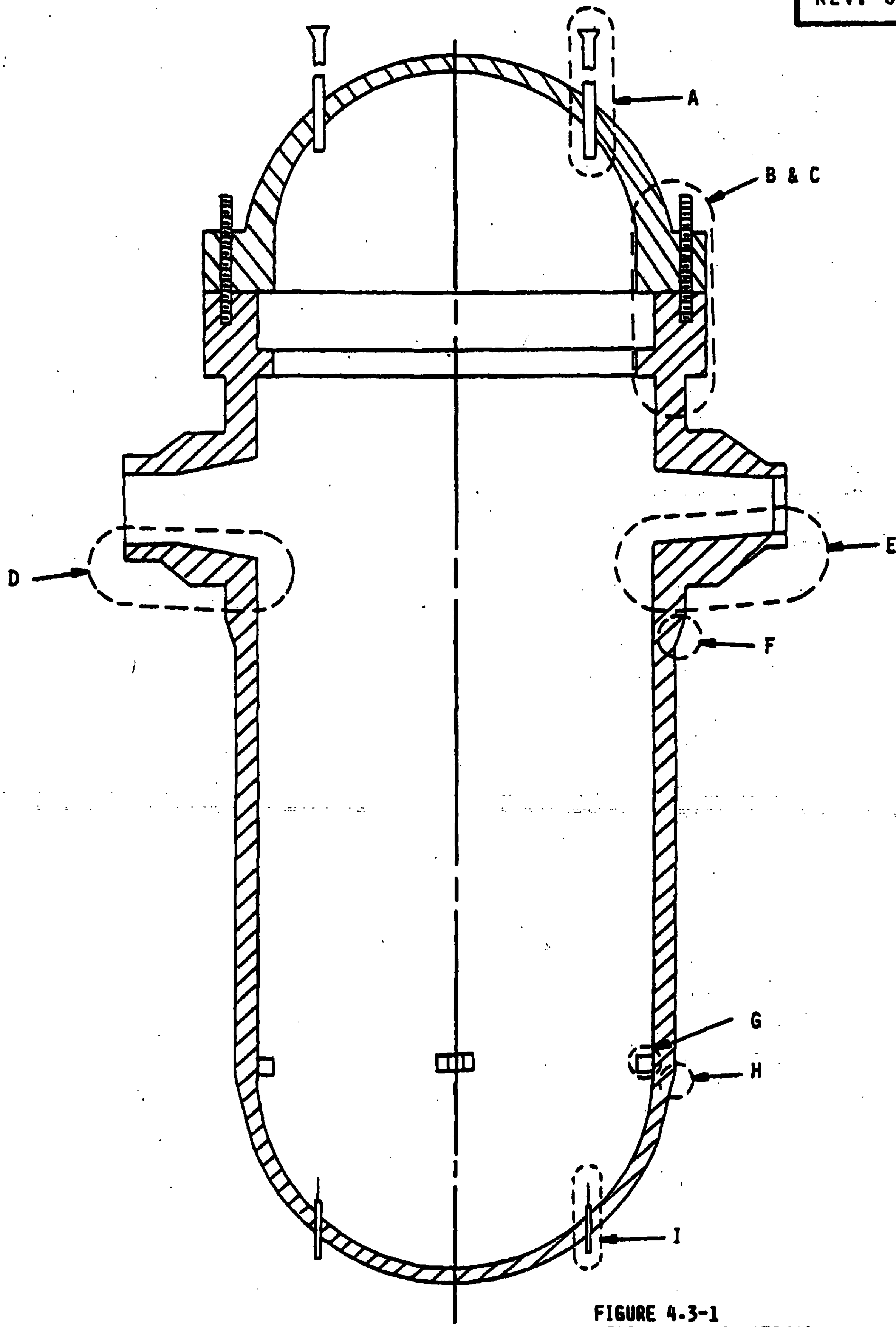



FIGURE 4.3-1
REACTOR VESSEL STRESS
ANALYSIS: AREAS EXAMINED
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

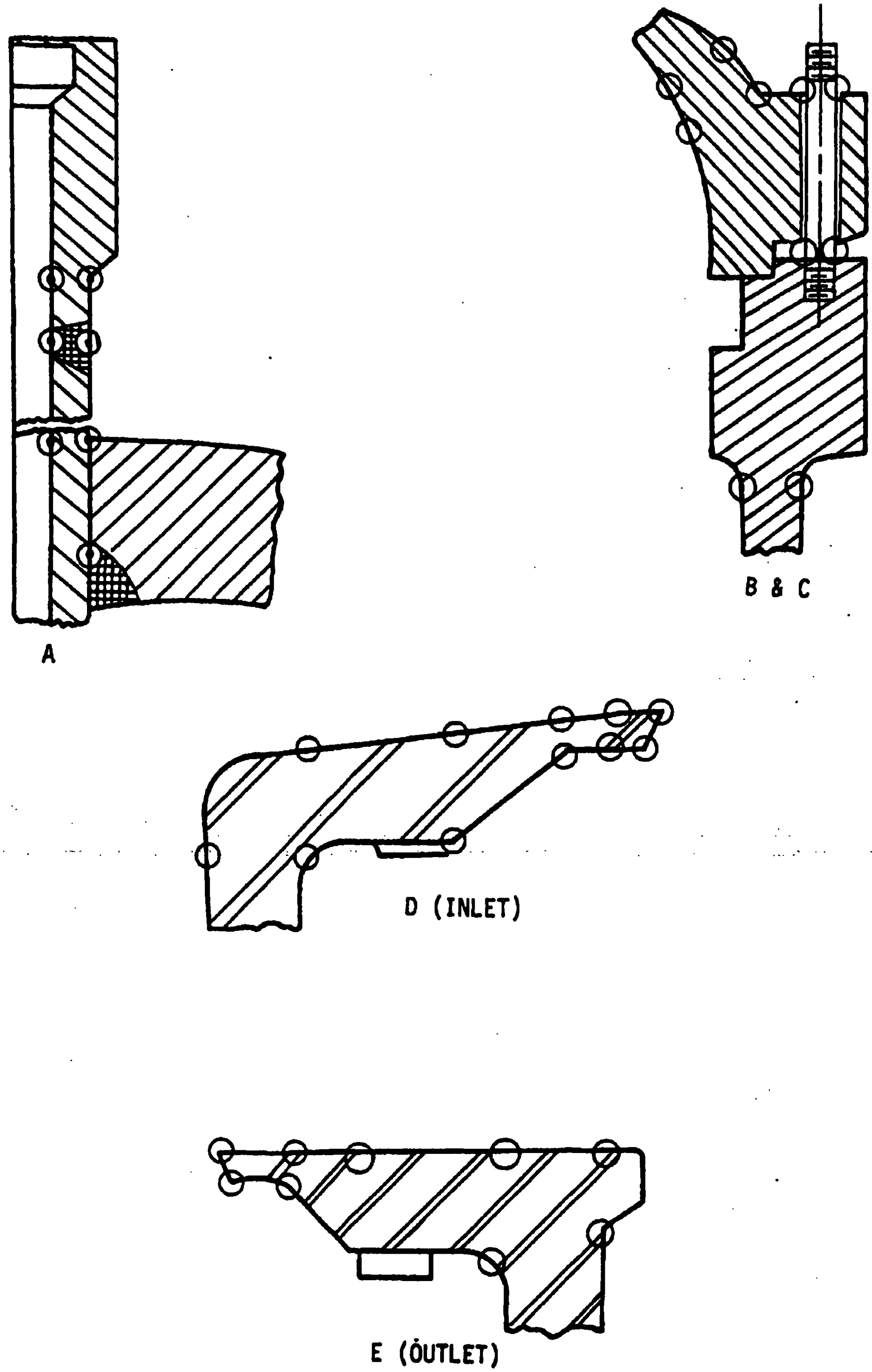
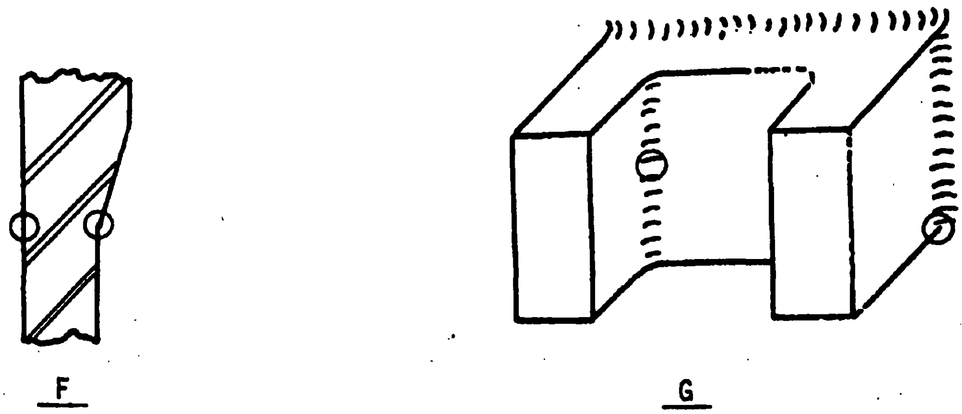
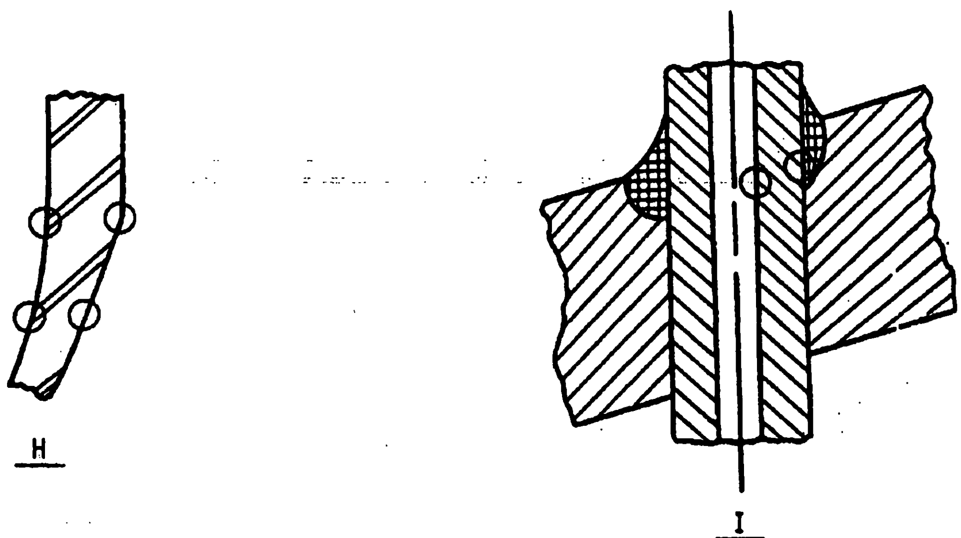



**FIGURE 4-3-2**
**REACTOR VESSEL STRESS ANALYSIS**
**DETAILS - UPPER**
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)




NOTE:

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

FIGURE 4.3-3
REACTOR VESSEL STRESS ANALYSIS
DETAILS - LOWER
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

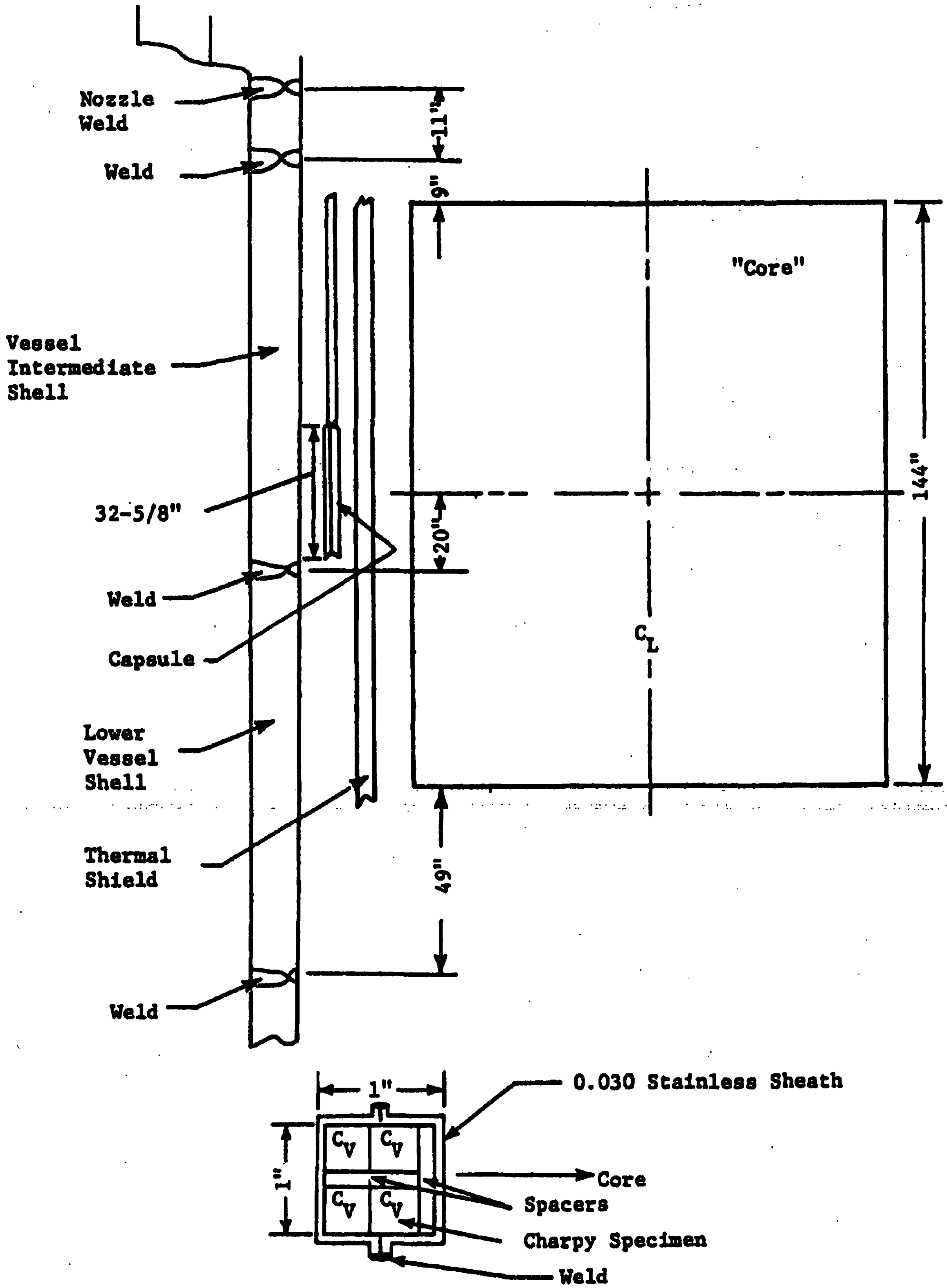
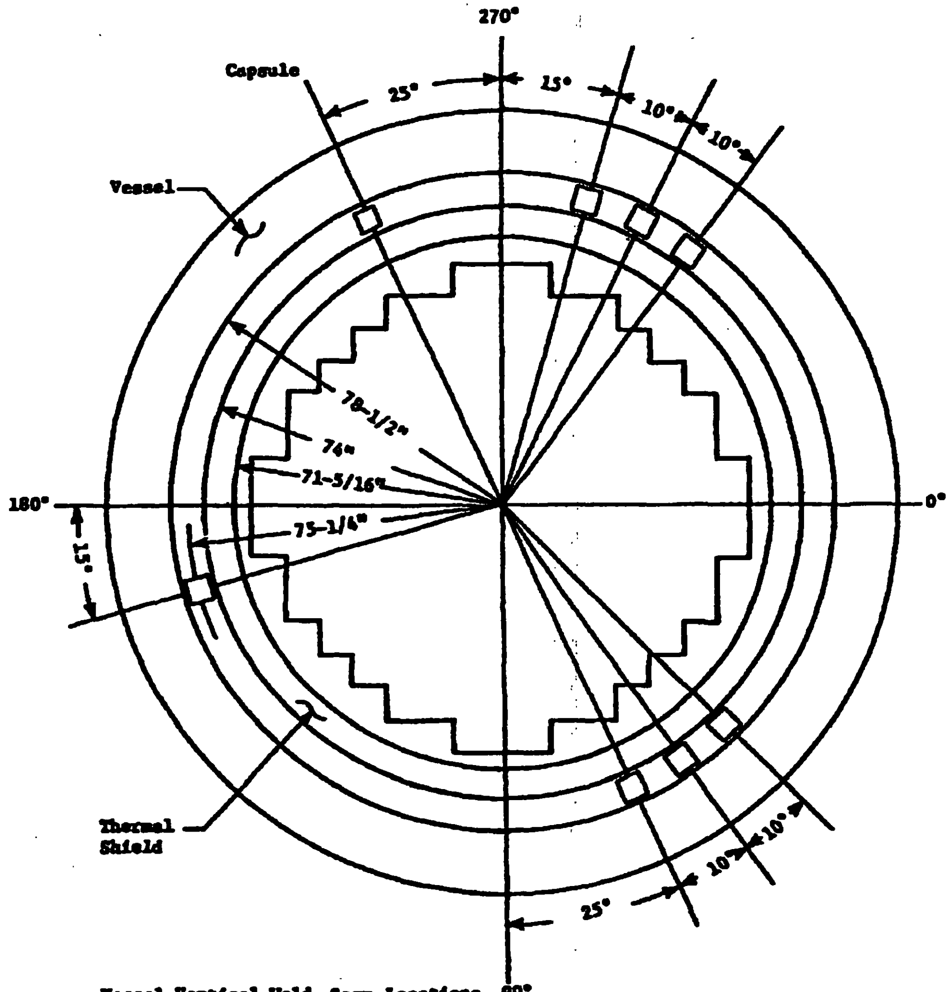



FIGURE 4.5-1
TYPICAL SURVEILLANCE
CAPSULE ELEVATION VIEW
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT




**Vessel Vertical Weld Seam Locations**

| | |
|---|---|
| Intermediate Shell | 45° and 225° |
| Lower Shell | 135° and 315° |

FIGURE 4.5-2
SURVEILLANCE CAPSULE
PLAN VIEW

BEAVER VALLY POWER STATION UNIT NO.1
UPDATED FINAL SAFETY ANALYSIS REPORT

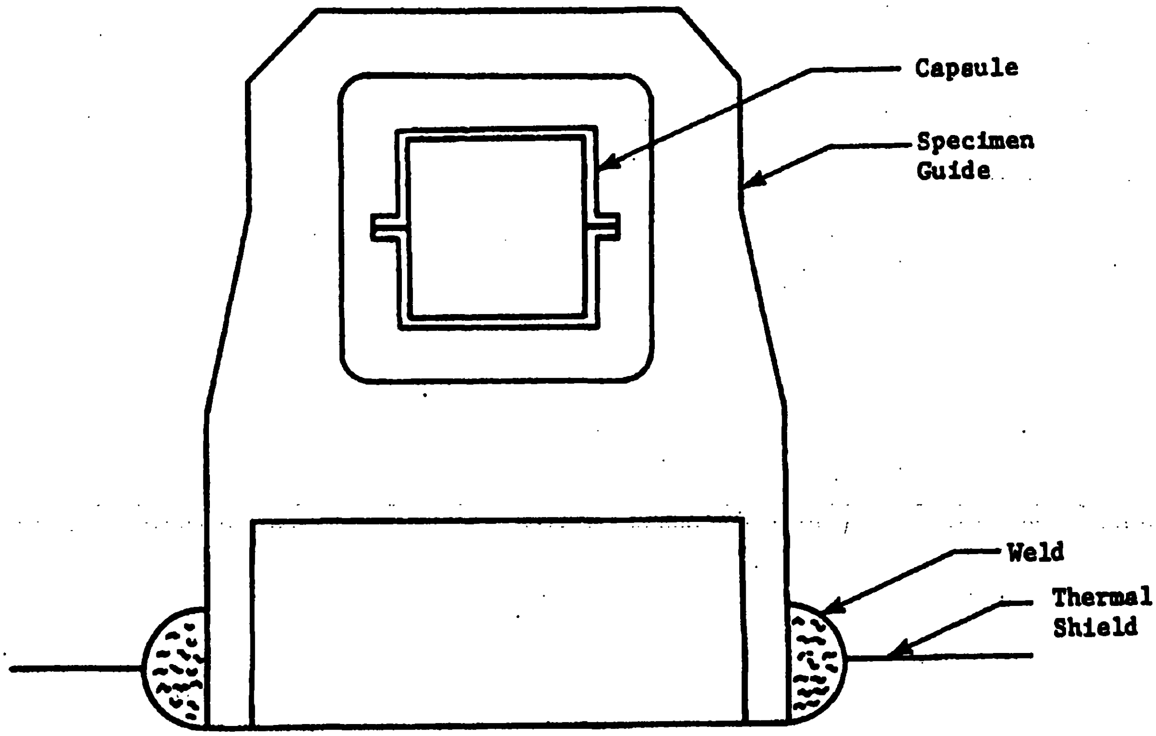



FIGURE 4.5-3
SPECIMEN GUIDE TO THERMAL
SHIELD ATTACHMENT
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

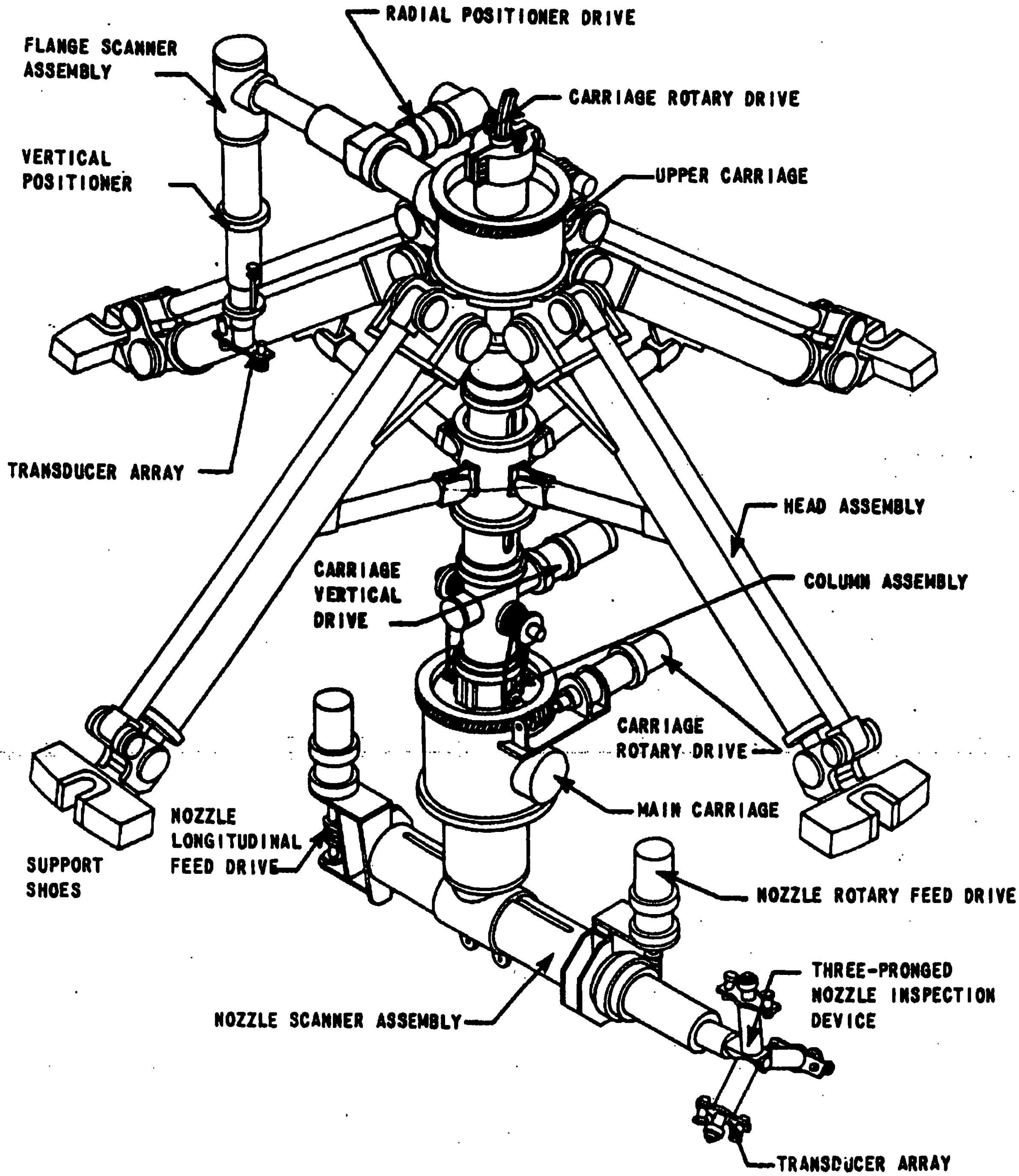



FIGURE 4.5-4
TOOL DETAILS
(NOZZLE AND FLANGE SCANNER)
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

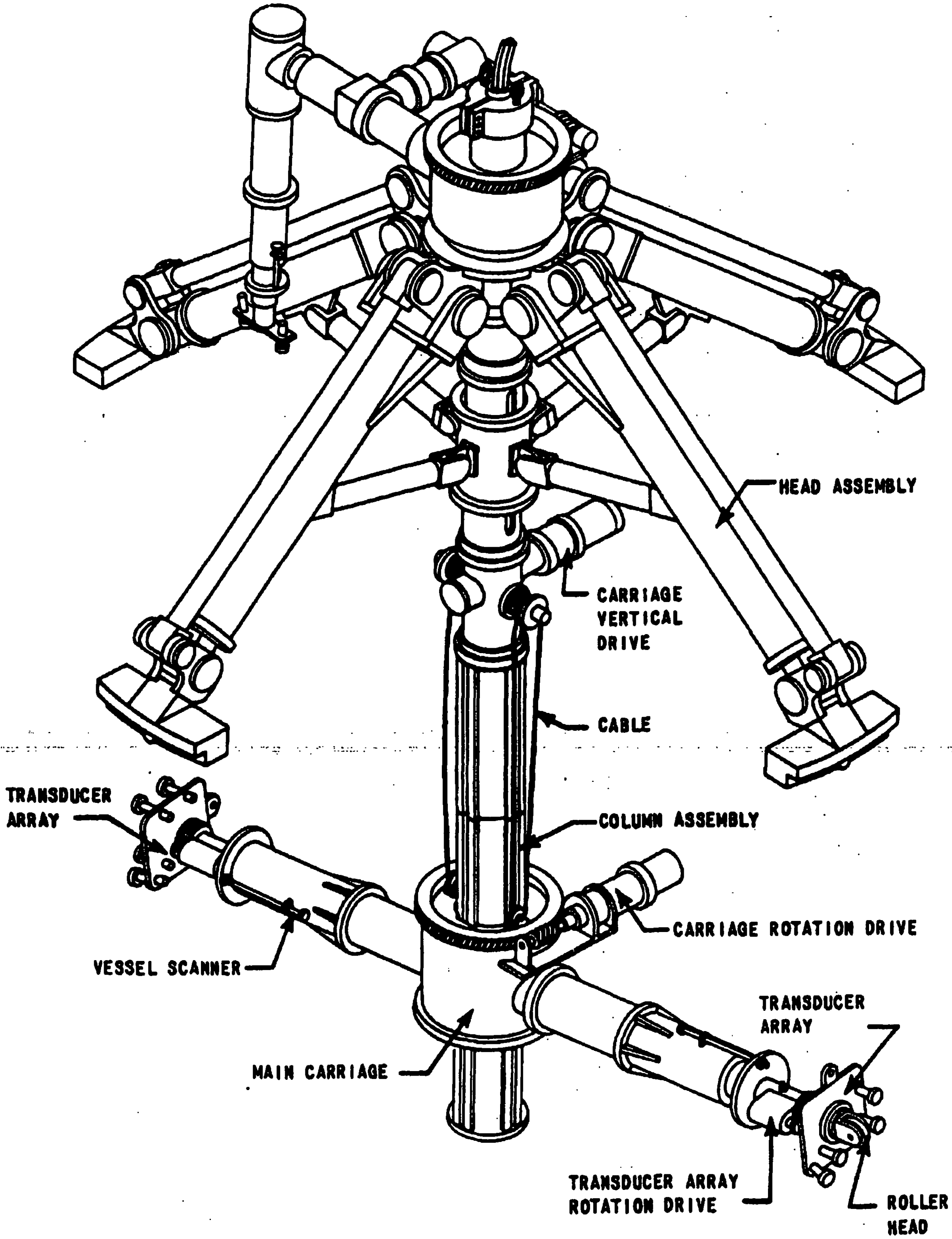



FIGURE 4.5-5
TOOL DETAILS (VESSEL SCANNER)
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

# APPENDIX 4A

# DETERMINATION OF REACTOR PRESSURE VESSEL NDTT

## 4A.1 MEASUREMENT OF INTEGRATED FAST NEUTRON (E > 1 MEV) FLUX AT THE IRRADIATION SAMPLES

Information on the spectrum of neutron fluxes at the location of the irradiation samples is obtained from the multigroup diffusion code P1MG[1]. Dosimeters including $U^{238}$, $NP^{237}$, Co-Al, Cu, Ni, Cd shielded Co-Al, and Fe from specimens are contained in the capsule assemblies.

The procedure for measurement of fast neutron flux by the $Fe^{54}$ (n, p) $Mn^{54}$ reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron spectrum, is similar.

The $Mn^{54}$ product of this reaction has a half life of 314 days and emits gamma rays of 0.84 Mev energy which are easily detected using a NaI scintillator. In irradiated steel samples, chemical separation of the $Mn^{54}$ may be performed to ensure freedom from interfacing activities. This separation is simple and very effective, yielding sources of very pure $Mn^{54}$ activity. In some samples all the interferences may be corrected for by the gamma spectrometric methods without any chemical separation. The count data is used to give the specific activity of $Mn^{54}$ per gram of iron. Because of the relatively long half life of $Mn^{54}$ the flux may be calculated for irradiation periods up to about two years. Beyond this time the dosimeter begins to reflect the later stages of irradiation. Calculation of total dose is from flux and integrated power output. The burnout of the $Mn^{54}$ produced is not significant until the thermal flux is about $10^{14}$ neutrons/$cm^2$ - sec.

The analysis of the sample requires that two steps are completed: first the measurement of $Mn^{54}$ disintegration rate per unit mass of sample; and second measurement of iron content of the sample. Having completed these analyses the calculation of the flux is as follows:

For an irradiation the activity of any activation product (A) is given by:

$$A = \text{Ø} \, \sigma \, N \, (1-e^{-\lambda t_i}) \, e^{-\lambda t_d} \qquad (4A\text{-}1)$$

Where: Ø = neutron flux, n/$cm^2$ sec

$\sigma$ = microscopic cross-section, barns, ($10^{-24} cm^2$)

N = number of target atoms

$\lambda$ = decay constant of product, $sec^{-1}$

$t_i$ = irradiation time, sec

$t_d$ = decay time from end of irradiation to counting time, sec

Then for a power reactor operating at various power levels over some long period we allow for flux changes by dividing the exposure period into several parts and normalizing the flux in each part as that fraction of full power represented. Then for $\tau$ periods:

$$A = Ø_m \sigma N \sum_{n=1}^{\tau} (1-e^{-\lambda t_{i_n}}) e^{\lambda t_{d_n}} F_n \qquad \text{(4A-2)}$$

where: $Ø_m$ = flux at maximum power, $n/cm^2 sec$

$t_{i_n}$ = irradiation time for end of nth period, sec

$t_{d_n}$ = decay time for end of nth period, sec

$F_n$ = flux normalizing factor which is:

$$\frac{\text{actual power output in nth period}}{\text{maximum possible in nth period}}$$

If now we write:

$$Ø_m \quad \sigma N = C \sum_{P1MG=1}^{55} ØP1MG(E,r)\, \sigma_{Fe}(E) \qquad \text{(4A-3)}$$

where: E = the energy

r = radial distance from core center line.

where the right hand side of Equation (4A-3) is the sum of the products of P1MG fluxes and the $Fe^{54}$ (n,p) $Mn^{54}$ cross-section(2) averaged over the P1MG energy groups, then the measured neutron flux (E > 1 Mev) is given by:

$$Ø = C \sum_{E=1}^{10} ØP1MG(E,r) \qquad \text{(4A-4)}$$

where C is a constant.

The error involved in the measurement of the specific activity of the dosimeter after irradiation is estimated to be ±5 percent.

## 4A.2 CALCULATION OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUENCE

The fast neutron (E > 1.0 MeV) fluence experienced by the test specimens contained in the surveillance capsules as well as by the pressure vessel wall is determined using discrete ordinates transport techniques. The specific methods applied follow the guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."[1]

In the application of this methodology to the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, plant-specific forward transport calculations are completed using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = [\phi(r,\theta)] * [\phi(r,z)]/[\phi(r)]$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in $r,\theta$ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the $r,\theta$ two-dimensional calculation.

All of the transport calculations are carried out using the DORT discrete ordinates code Version 3.1[2] and the BUGLE-96 cross-section library[3]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering is treated with a $P_5$ legendre expansion and the angular discretization is modeled with an $S_{16}$ order of angular quadrature. Energy and space dependent core power distributions as well as system operating temperatures are treated on a fuel cycle-specific basis.

In developing the analytical models of the reactor geometry, nominal design dimensions are employed for the various structural components. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor are taken to be representative of full power operating conditions. These coolant temperatures are varied on a cycle-specific basis. The reactor core itself is treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc. In the transport models, spatial mesh sizes are chosen to assure that proper convergence of the inner iterations is achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the calculations is set at a value of 0.001.

Core power distribution data are input to the transport calculations on a cycle-specific basis. In applying this data, the fission spectra, neutrons released per fission, and energy release per fission account for the presence of both uranium and plutonium fissioning isotopes. Specifically, the burnup dependent effects on the neutron source account for the spatial variation of the magnitude of the source as well as the spectral effects introduced by the spatial variation of the various fissioning isotopes ($^{235}U$, $^{238}U$, $^{239}Pu$, $^{240}Pu$, $^{241}Pu$, and $^{242}Pu$).

## 4A.3 EVALUATION OF SURVEILLANCE CAPSULE DOSIMETRY

Measurements obtained from individual surveillance capsules are used to validate, but not to modify, the results of the fluence calculations. That is, the measurement to calculation comparisons are used solely to demonstrate that the plant-specific fluence calculations are consistent with the methodology benchmarking and meet uncertainty requirement of 20% ($1\sigma$) specified in Regulatory Guide 1.190.

In the evaluation of these surveillance capsule measurements, the specific activity of each of the radiometric sensors is determined using established ASTM procedures as recommended by Regulatory Guide 1.190. Following sample preparation and weighing, the specific activity of each sensor is determined by means of a high purity germanium gamma spectrometer. In the case of the surveillance capsule multiple foil sensor sets, these analyses are performed by direct counting of each of the individual wires; or, as in the case of U-238 and Np-237 fission monitors, by direct counting preceded by dissolution and chemical separation of cesium from the sensor.

The irradiation history of the reactor over its operating lifetime is determined from plant power generation records. In particular, operating data are extracted on a monthly basis from reactor startup to the end of the capsule irradiation period. For the sensor sets utilized in the surveillance capsule irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation are determined from the following equation:

$$R = \frac{A}{N_0 F Y \sum_j \frac{P_j}{P_{ref}} C_j [1 - e^{-\lambda t_j}] e^{-\lambda t_d}}$$

where:

| | | |
|---|---|---|
| $A$ | = | measured specific activity provided in terms of disintegrations per second per gram of target material (dps/gm) |
| $R$ | = | reaction rate averaged over the irradiation period and referenced to operation at a core power level of $P_{ref}$ expressed in terms of reactions per second per nucleus of target isotope (rps/nucleus) |
| $N_0$ | = | number of target element atoms per gram of sensor |
| $F$ | = | weight fraction of the target isotope in the sensor material |
| $Y$ | = | number of product atoms produced per reaction |
| $P_j$ | = | average core power level during irradiation period j (MW) |
| $P_{ref}$ | = | maximum or reference core power level of the reactor (MW) |
| $C_j$ | = | calculated ratio of $\phi(E > 1.0\text{ MeV})$ during irradiation period j to the time weighted average $\phi(E > 1.0\text{ MeV})$ over the entire irradiation period |
| $\lambda$ | = | decay constant of the product isotope (sec-1) |
| $t_j$ | = | length of irradiation period j (sec) |
| $t_d$ | = | decay time following irradiation period j (sec) |

The summation is carried out over the total number of monthly intervals comprising the total irradiation period.

In the above equation, the ratio $P_j/P_{ref}$ accounts for month by month variation of power level within a given fuel cycle. The ratio $C_j$ is calculated for each fuel cycle and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. Since the neutron flux at the measurement locations within the surveillance capsules is dominated by neutrons produced in the peripheral fuel assemblies, the change in the relative power in these assemblies from fuel cycle to fuel cycle can have a significant impact on the activation of neutron sensors. For a single cycle irradiation $C_j = 1.0$. However, for multiple cycle irradiations, particularly those employing low leakage fuel management the additional $C_j$ correction must be utilized in order to provide accurate determinations of the decay corrected reaction rates for the dosimeter sets contained in the surveillance capsules.

Prior to comparing the measured reaction rates to the results of the neutron transport calculations, corrections are made to the $^{238}U$ measurements to account for the presence of $^{235}U$ impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. In addition to these corrections to the $^{238}U$ measurements, adjustments are also made to both the $^{238}U$ and $^{237}Np$ sensor reaction rates to account for gamma ray induced fission reactions occurring in these sensors over the course of the irradiation.

## References for Appendix 4A


1. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

2. RSICC Computer Code Collection CCC-650, "DOORS3.1 One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center, Oak Ridge National Laboratory, August 1996.

3. RSIC Data Library Collection DLC-185, "BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Shielding Information Center, Oak Ridge National Laboratory, March 1996.