5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

This section describes the reactor coolant system (RCS), and includes a process flow diagram (Figure 5.1-1) and a piping and instrumentation diagram (Figure 5.1-2).

5.1.1 Design Bases

The performance and safety design bases of the reactor coolant system (RCS) and its major components are interrelated. These design bases are listed below:

A. The RCS has the capability to transfer to the steam and power conversion system the heat produced during power operation and when the reactor is subcritical, including the initial phase of plant cooldown.

B. The RCS has the capability to transfer to the residual heat removal system the heat produced during the subsequent phase of plant cooldown and cold shutdown.

C. The RCS heat removal capability under power operation and normal operational transients, including the transition from forced to natural circulation, assures no fuel damage within the operating bounds permitted by the reactor control and protection systems.

D. The RCS provides the water used as the core neutron moderator and reflector and as a solvent for the neutron absorber used for chemical shim control.

E. The RCS maintains the homogeneity of the soluble neutron poison concentration and limits the rate of change of coolant temperature, so that uncontrolled reactivity changes do not occur.

5.1.2 Design Description

The reactor coolant system (RCS), shown in Figure 5.1-1, consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump, steam generator, and associated piping and valves. In addition, the system includes a pressurizer, pressurizer relief and safety valves, interconnecting piping, and the instrumentation necessary for operational control. All the above components are located in the containment building.

During operation, the RCS transfers the heat generated in the core to the steam generators, where steam is produced to drive the turbine generator. Borated demineralized water is circulated in the RCS at a flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector and as a solvent for the neutron absorber used for chemical shim control.

The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the plant.

The RCS pressure is controlled by the use of the pressurizer where water and steam are maintained at saturation conditions by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Spring loaded safety valves and power operated relief valves connected to the pressurizer provide for steam discharge from the RCS. Discharged steam is piped to the pressurizer relief tank (pressurizer relief discharge system), where the steam is condensed and cooled by mixing with water.

The extent of the RCS is defined as:

- The reactor vessel, including control rod, gray rod and displacer rod drive mechanism housings.

- The portions of the steam generators containing reactor coolant.

- The reactor coolant pumps.

- The pressurizer.

- The pressurizer safety and relief valves.

- The reactor vessel level instrumentation system.

- The reactor vessel head vent.

- The interconnecting piping, valves, and fittings between the principal components listed above.

- The piping, fittings, and valves leading to connecting auxiliary or support systems.

The RCS schematic flow diagram is shown in Figure 5.1-2. See Table 5.1-1 for a tabulation of principal pressures, temperatures and flow rates of the system under normal steady-state, full power operating conditions corresponding to this figure. These parameters are based on the best-estimate flow at the pump discharge. The RCS volume under these conditions is presented in Table 5.1-2.

A piping and instrumentation diagram of the RCS is shown in Figure 5.1-2. This diagram shows the extent of the systems located within the containment and the points of separation between the RCS and the secondary (heat utilization) system.

5.1.3 System Components

The major components of the reactor coolant system are as follows:

A. Reactor Vessel

The reactor vessel is cylindrical and has a welded hemispherical bottom head and a removable flanged-and-gasketed hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core.

The vessel has inlet and outlet nozzles located in a horizontal plane just below the reactor vessel flange but above the top of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

B. Steam Generators

The steam generators are vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the bottom head of the steam generator. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

C. Reactor Coolant Pumps

The reactor coolant pumps (RCPs) are single-speed centrifugal units driven by water/air-cooled, three-phase induction motors. The shaft is vertical with the motor mounted above the pump. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The flow inlet is at the bottom of the pump, and the discharge is on the side.

D. Piping

The reactor coolant piping is seamless stainless steel piping. The hot leg is defined as the piping between the reactor vessel outlet nozzle and the steam generator. The cold leg is defined as the piping between the reactor coolant pump outlet and the reactor vessel. The crossover leg is defined as the piping between the steam generator outlet and the RCP inlet.

E. Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. The pressurizer is connected to the hot leg of one of the coolant loops by a surge line. Electrical heaters are installed through the bottom head of the vessel. The spray nozzle and relief and safety valve connections are located in the top head of the vessel.

F. Safety and Relief Valves

The pressurizer safety valves are of the totally enclosed pop-type. The valves are spring loaded and self actuated with backpressure compensation. The power operated relief valves are solenoid operated valves. They are operated automatically or by remote manual control. Position indicating lights are provided in the control room for the power operated relief valves. In addition, both the safety valves and the power operated relief valves are equipped with devices which provide positive position indication (open or closed) in the control room.

Remotely operated block valves are provided to isolate the inlets to the power operated relief valves. The block valves can be operated either automatically or manually. In the automatic mode, i.e., when the plant is operating under automatic pressure control, the valves are opened whenever the pressurizer pressure is above the block valve set pressure, which is approximately 350 psi below the normal pressurizer operating pressure of 2250 psia. They are closed automatically if the pressure falls below the set pressure. These characteristics serve to protect against the consequences of a stuck open PORV when the plant is operating under automatic pressurizer pressure control. An override feature permits manual operation of the block valves. This capability would be required for certain off normal events such as a safety grade cold shutdown requiring manual PORV operation, of if excessive PORV leakage occurs.

5.1.4 System Performance Characteristics

Design and performance characteristics of the reactor coolant system are provided in Table 5.1-2.

5.1.4.1 Reactor Coolant Flows

The various reactor coolant flow rates discussed below are major parameters considered in the design of the reactor coolant system and its major components.

- Best Estimate Flow (BEF)

  The best estimate flow is considered to be the most likely value for the nominal full power operating condition. This flow is based on the best estimates of reactor vessel, steam generator and piping flow resistances, and on the best estimate of the reactor coolant pump head-flow capability, with no uncertainties assigned to either the system flow resistance or the pump head. The best estimate flow provides the basis for the establishment of the other design flow rates and defines the performance requirement for the reactor coolant pump head.

- Minimum Expected Flow Measurement (Minimum EFM)

  The minimum EFM is defined to be the lowest measured flow predicted using flow measurement and modeling uncertainties. The parameters used to define minimum EFM, particularly the flow measurement uncertainty, are selected to assure that the measured flow will be equal to or greater than the predicted minimum measured flow.

The minimum EFM, then, is the lowest design flow value in which the flow measurement uncertainty is statistically combined with hydraulic modeling uncertainties.

- Thermal Design Flow (TDF)

  The thermal design flow is specified as the conservatively low design flow wherein total flow measurement error is accounted for by algebraically adding the flow measurement uncertainty to the minimum EFM value. This method assures that the actual flow will be greater than the thermal design flow.

  The thermal design flow provides the additional flow margin needed for certain analyses such as determination of the nominal full power operating parameters. For the WAPWR the thermal design flow is defined as 96.8 percent of the best estimate flow, based on use of the transit time flow meter.

- Mechanical Design Flow (MDF)

  The mechanical design flow is the conservatively high value of flow used in the mechanical design of Reactor System components such as the reactor internals and fuel assemblies, in recognition of possibly lower system flow resistance or higher pump head capability. The mechanical design flow is defined as 4 percent greater than the best estimate flow.

Table 5.1-1 (Sheet 1 of 2)

NOTES TO RCS PROCESS FLOW DIAGRAM (Figure 5.1-1)

Steady-State, Full-Power Operation[a]

(a,

Location	Fluid	Pressure (psig)	Nominal Temperature	Flow (gal/min[b])	Flow (lb/hx10)	Volume (ft^3)
1	Reactor coolant					
2	Reactor coolant					
3	Reactor coolant					
4	Reactor coolant					
5	Reactor coolant					
6	Reactor coolant					
7-12	Reactor coolant					
13-18	Reactor coolant					
19-24	Reactor coolant					
25	Reactor coolant					
26	Reactor coolant					
27	Reactor coolant					
28	Steam					
29	Reactor coolant					
30	Reactor coolant					

TABLE 5.1-1 (Sheet 2 of 2)

NOTES TO RCS PROCESS FLOW DIAGRAM (Figure 5.1-1)

Steady-State, Full-Power Operation[a]

Location	Fluid	Pressure (psig)	Nominal Temperature	Flow (gal/min[b])	Flow (lb/hx10)	Volume (ft^3)
31	Reactor coolant					
32	Steam					
33	Reactor coolant					
34	N_2					
35	Reactor coolant					
36	N_2					
37	N_2					
38	N_2					
39	Pressurizer relief tank water					
40	Emergency Water Storage Tank					

(a,c

a. Simplifying assumption is made that letdown and charging flows are distributed over all loops. Charging, letdown, and continuous spray flows are ignored in overall balance.

b. At the conditions specified.

TABLE 5.1-2

SYSTEM DESIGN AND OPERATING PARAMETERS

Plant design life, years	40	
Nominal operating pressure, psig	2235	(a,c)
Total system volume, including pressurizer and surge line, ft^3		
System liquid volume, including pressurizer water at maximum guaranteed power, ft^3		
Pressurizer spray rate, maximum, gpm		
Pressurizer heater capacity, kW		

System Thermal and Hydraulic Data	4 Pumps Running (E.O.L. Conditions)	
NSSS power, MWt	3816	
Reactor power, MWt	3800	
Thermal design flows, gpm		(a,c)
Active loop		
Reactor		
Total reactor flow, 10^6 lb/hr		
Temperatures, °F		
Reactor vessel outlet		
Reactor vessel inlet		
Steam generator outlet		
Steam generator steam		
Feedwater		
Steam pressure, psia		
Total steam flow, 10^6 lb/hr		
Best estimate flows, gpm		
Active loop		
Reactor		
Mechanical design flows, gpm		
Active loop		
Reactor		

TABLE 5.1-2 (Sheet 2 of 2)

SYSTEM DESIGN AND OPERATING PARAMETERS

System Thermal and Hydraulic Data	4 Pumps Running (E.O.L. Conditions)
System Pressure Drops	(a,c)
Reactor vessel ΔP, psi	
Steam generator ΔP, psi	
Hot leg piping ΔP, psi	
Crossover leg piping ΔP, psi	
Cold leg piping ΔP, psi	
Pump head, ft	

* Includes core, internals, and nozzles.

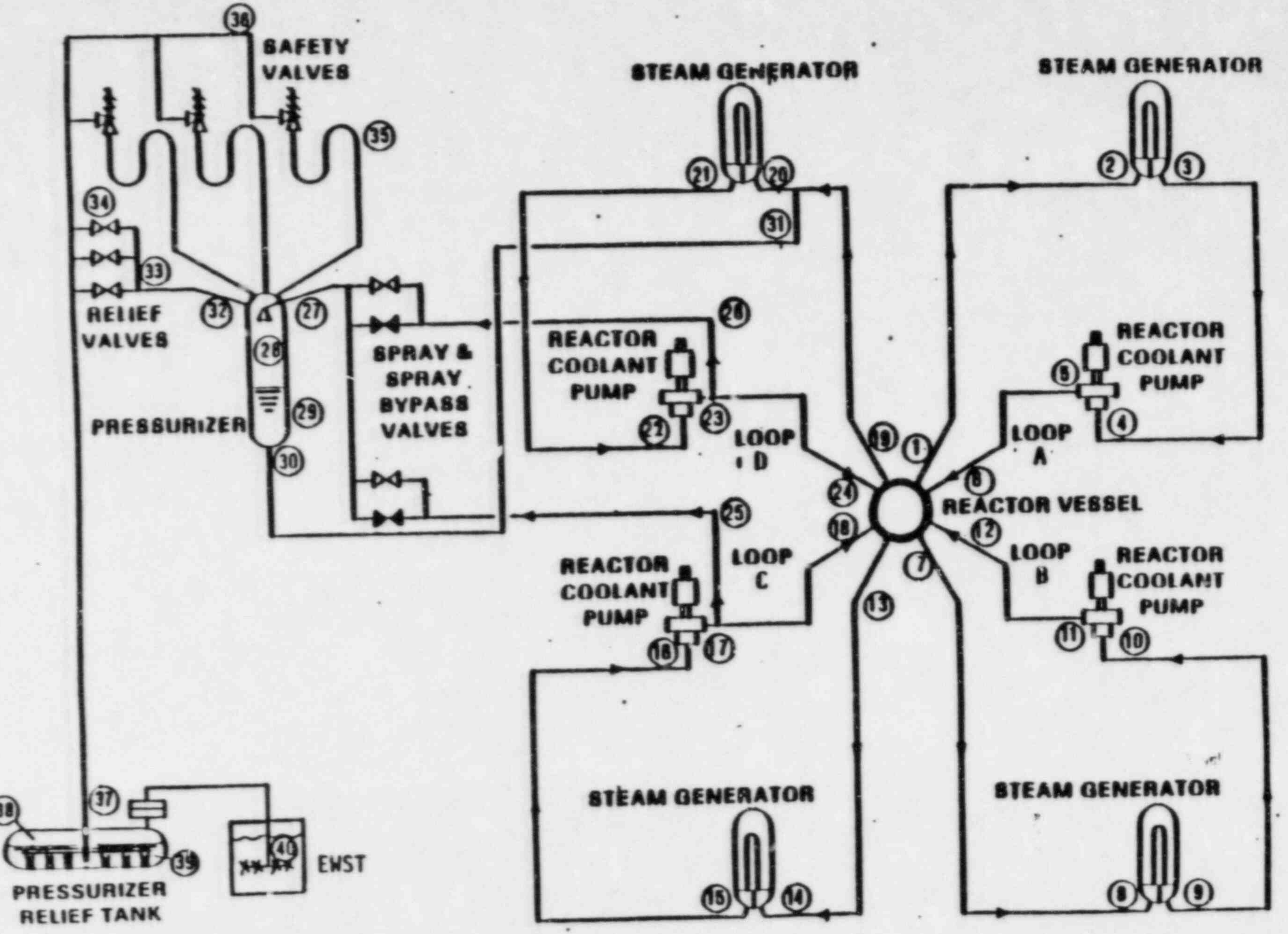

FIGURE 5.1-1 REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime. Section 50.2 of 10 CFR 50 defines the RCPB as extending to the outermost containment isolation valve in system piping which penetrates the containment and is connected to the reactor coolant system (RCS). This section is limited to a description of the components of the RCS, as defined in Section 5.1, unless otherwise noted. Components which are part of the RCPB (as defined in 10 CFR 50) but are not described in this section are described in the following sections:

A. Section 6.3 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System" - RCPB components which are part of the emergency core cooling system.

B. Subsection 9.3.4 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems" - RCPB components which are part of the chemical and volume control system.

C. Subsection 3.9.1 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" - Design loading, stress limits, and analyses applied to the ASME Code Class 1 RCS components.

D. Subsection 3.9.3 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" - Design loadings, stress limits, and analyses applied to ASME Code Class 2 and 3 components.

The abbreviation RCS, as used in this section, is as defined in Section 5.1. When the term RCPB is used in this section, its definition is that of Section 50.2 of 10 CFR 50.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

RCS components are designed and fabricated in accordance with 10 CFR 50, Section 50.55a, Codes and Standards. The addenda of the ASME code to be applied in the design of each component will be determined in connection with the specific plant application.

5.2.1.2 Applicable Code Cases

Regulatory Guides 1.84 and 1.85 document ASME code cases acceptable to the NRC on a generic basis. Code cases may be approved by the NRC on a plant specific basis if an approval request is made.

Westinghouse takes the following position with respect to controlling code cases used on RCS components.

1. Westinghouse controls its suppliers to:

    a. Limit the use of code cases to those listed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered, except as allowed in item 2 below.

    b. Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered, where use of such code cases is needed by the supplier.

    c. Permit continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

2. Westinghouse will seek NRC permission for the use of code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered. Suppliers may use these code cases only if NRC permission is obtained or is otherwise assured (e.g., a later version of the regulatory guide which includes endorsement of the code case).

5.2.2 Overpressure Protection

RCS overpressure protection is provided by the pressurizer safety valves and the steam generator safety valves; in conjunction with the action of the reactor protection system. Combinations of these systems ensure compliance with the overpressure protection requirements of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized water reactor systems.

The only portions of an auxiliary system connected to the RCS that are utilized for overpressure protection of the RCS are the liquid relief valves of the residual heat removal system (RHRS). These valves protect the RCS at low temperatures when the RHRS is in operation.

5.2.2.1 Design Bases

Overpressure protection is provided for the RCS by the pressurizer safety valves. This protection is afforded for the following events which envelope those credible events that could lead to overpressure of the RCS if adequate overpressure protection were not provided:

- o Loss of electrical load and/or turbine trip.
- o Uncontrolled rod withdrawal at power.
- o Loss of reactor coolant flow.
- o Loss of normal feedwater.
- o Loss of offsite power to the station auxiliaries.

The sizing of the pressurizer safety valves is based on the analysis of a complete loss of steam flow to the turbine with the reactor operating at 102 percent of engineered safeguards design power. In this analyses, feedwater flow is also assumed to be lost, and no credit is taken for operation of the pressurizer power-operated relief valves (PORVs), pressurizer level control system, pressurizer spray system, rod control system, steam dump system, or steam line PORVs. The reactor is maintained at full power (no credit for direct reactor trip on turbine trip), and steam relief through the steam generator safety valves is considered. The total pressurizer safety valve capacity is required to be at least as large as the maximum surge rate into the pressurizer during this transient.

This sizing procedure results in a safety valve capacity well in excess of the capacity required to prevent exceeding 110 percent of system design pressure for the events listed above.

For a discussion of cold overpressure protection of the RCS and the mitigation of potential overpressurization transients, see Subsection 5.2.2.10 of this module.

Overpressure protection for the steam system is provided by steam generator safety valves. The steam system safety valve capacity is based on providing enough relief to remove the engineered safeguards design steam flow. This must be done while limiting the maximum steam system pressure to less than 110 percent of the steam generator shell side design pressure.

Blowdown and heat dissipation systems of the nuclear steam supply system (NSSS) connected to the discharge of these pressure relieving devices are discussed in Subsection 5.4.11 of this module.

The steam generator blowdown system is discussed in Subsection 10.4.8 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion".

5.2.2.2 Design Evaluation

A description of the pressurizer safety valves' performance characteristics along with the design description of the incidents, assumptions made, method of analysis, and conclusions are discussed in Chapter 15.

The relief capacities of the pressurizer and steam generator safety valves are determined from the postulated overpressure transient conditions in conjunction with the action of the reactor protection system. An overpressure protection report specifically for the WAPWR will be prepared for the RESAR-SP/90 FDA document in accordance with Article NB-7300 of Section III of the ASME Code. The description of the analytical model used in the analysis of the overpressure protection system and the basis for its validity will be provided for the RESAR-SP/90 FDA document.

The capacities of the pressurizer safety and relief valves are discussed in Subsection 5.4.13 of this module. The setpoints and reactor trip signals which occur during overpressure transients are discussed in Subsection 5.4.10 of this module.

5.2.2.3 Piping and Instrumentation Diagrams

Overpressure protection for the RCS is provided by the pressurizer safety and relief valves shown in Figure 5.1-2. These valves discharge to the pressurizer relief tank through a common manifold.

The steam system safety valves are discussed in Section 10.3 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion".

5.2.2.4 Equipment and Component Description

The operation, significant design parameters, number and types of operating cycles, and environment conditions of the pressurizer safety valves are discussed in Subsection 3.9.1 and Section 3.11 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" and Subsection 5.4.13 of this module.

Section 10.3 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion" contains a discussion of the equipment and components of the steam system overpressure protection features.

5.2.2.5 Mounting of Pressure Relief Devices

The design and installation of the pressure relief devices for the RCS are described in Subsection 5.4.11 of this module. The design basis for the assumed loads for the primary and secondary side pressure relief devices are described in Section 3.9 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design". Subsection 10.3.2 of RESAR-SP/90 PDA Module 8, "Steam and Power Conversion" provides a discussion of the main steam safety valves and the power-operated atmospheric steam relief valves.

5.2.2.6 Applicable Codes and Classification

The requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 (Overpressure Protection Report) and NC-7300 (Overpressure Protection Analysis), are met.

Piping, valves, and associated equipment used for overpressure protection are classified in accordance with American National Standards Institute (ANSI) ANSI/ANS-51.1-1983, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. These safety class designations are delineated in Table 3.2-1 of this module and shown in Figure 5.1-2 of this module.

5.2.2.7 Material Specifications

Refer to Subsection 5.2.3 of this module for a description of material specifications.

5.2.2.8 Process Instrumentation

Each pressurizer safety valve discharge line incorporates a control board temperature indicator and alarm to notify the operator of steam discharge due to either leakage or actual valve operation. For a further discussion on process instrumentation associated with the system, refer to RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

5.2.2.9 System Reliability

The reliability of the pressure relieving devices will be demonstrated for the RESAR-SP/90 FDA document.

5.2.2.10 RCS Pressure Control During Low-Temperature Operation

Administrative controls in plant procedures are available to aid the operator in controlling RCS pressure during low-temperature operation. However, to provide a backup to the operator and to minimize the frequency of RCS overpressurization, an automatic system is provided to help mitigate inadvertent pressure excursions to within allowable limits.

This protection is provided through use of the PORVs to mitigate the effects of any potential overpressurization transients. Analyses have shown that one PORV is sufficient to prevent violation of these limits due to anticipated mass and heat input transients. The mitigation system is required to be available only during low-temperature water solid operation. It is manually armed and automatically actuated.

5.2.2.10.1 System Operation

Two of the three pressurizer PORVs are supplied with actuation logic to ensure that an independent RCS pressure control feature is available to the operator during low-temperature operations. This system provides the capability for additional RCS inventory letdown thereby maintaining RCS pressure within allowable limits. Refer to Section 7.7 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power", Subsection 5.4.7 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System", Subsections 5.4.10 and 5.4.13 of this module, and Subsection 9.3.4 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems" for additional information on RCS pressure and inventory control during other modes of operation.

The basic function of the system logic is to monitor continuously the RCS temperature and pressure conditions whenever plant operation is at low temperatures. An auctioneered system temperature is continuously converted to an allowable pressure and the actual (measured) pressure is continuously compared to this value. A main control board alarm is actuated whenever the measured pressure approaches within a predetermined amount of the allowable pressure, thereby indicating that a pressure transient is occurring. On a further increase in measured pressure, an actuation signal is transmitted to open the PORVs to mitigate the pressure transient.

5.2.2.10.2 Evaluation of Low-Temperature Overpressure Transients - Pressure Transient Analyses

ASME Section III, Appendix G, establishes guidelines and limits for RCS pressure primarily for low temperature conditions (≤350°F). The relief system discussed in Subsection 5.2.2.10 of this module satisfies these conditions.

Both heat input and mass input analyses take into account the single failure criterion and therefore only one PORV is assumed to be available for pressure relief. The evaluation will demonstrate that the allowable limits are not exceeded upon occurrence of either of these transients and therefore, these transients do not constitute an impairment to vessel integrity and plant safety.

5.2.2.10.3 Operating Basis Earthquake Evaluation

The PORVs have been designed in accordance with the ASME code to provide the integrity required for the reactor coolant pressure boundary. They are qualified in accordance with the Westinghouse valve operability program which is described in detail in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

5.2.2.10.4 Administrative Controls

Although the system described in Subsection 5.2.2.10.1 of this module is designed to maintain RCS pressure within allowable limits, administrative controls are provided in plant procedures for minimizing the potential for any transient that could actuate the cold overpressure mitigation system. The following discussion highlights these procedural controls, listed in order of their importance for mitigating RCS cold overpressurization transients.

Of primary importance is the basic method of operation of the plant. Normal plant operating procedures maximize the use of a pressurizer cushion (steam bubble) during periods of low-pressure, low-temperature operation. This cushion dampens the plants response to potential transient generating inputs, thereby providing easier pressure control with the slower response rates.

An adequate cushion substantially reduces the severity of some potential pressure transients such as reactor coolant pump heat input, and slows the rate of pressure rise for others. In conjunction with the previously discussed alarms, this provides reasonable assurance that most potential transients can be terminated by operator action before the overpressure relief system actuates.

However, for those modes of operation when water solid operation may still be possible, the following procedures further highlight precautions that minimize the potential for developing an overpressurization transient. The following specific recommendations are made:

A. The residual heat removal (RHR) inlet lines from the reactor coolant loop are not isolated unless the charging pumps are stopped. This precaution is to ensure there is a relief path from the reactor coolant loop to the RHR suction line relief valves when the RCS is at low pressure (less than 500 psi) and is water solid.

B. Whenever the plant is water solid and the reactor coolant pressure is being maintained by the low pressure letdown control valve, letdown flow must bypass the normal letdown orifices, and the valve in the bypass line must be in the full-open position. During this mode of operation, all three letdown orifices must also remain open.

C. If all reactor coolant pumps have stopped for more than 5 minutes during plant heatup and the reactor coolant temperature is greater than the charging and seal injection water temperature, no attempt is made to restart a pump unless a steam bubble is formed in the pressurizer. This precaution minimizes the pressure transient when the pump is started and the cold water previously injected by the charging pumps is circulated through the warmer reactor coolant components. The steam bubble accommodates the resultant expansion as the cold water is rapidly warmed.

D. If all reactor coolant pumps are stopped and the RCS is being cooled down by the residual heat exchangers, a nonuniform temperature distribution may occur in the reactor coolant loops. No attempt is made to restart a reactor coolant pump unless a steam bubble is formed in the pressurizer.

E. During plant cooldown, all steam generators are connected to the steam header to ensure a uniform cooldown of the reactor coolant loops.

F. At least one reactor coolant pump is maintained in service until the reactor coolant temperature is reduced to 160°F.

These special precautions back up the normal operational mode of maximizing periods of steam bubble operation so that cold overpressure transient prevention is continued during periods of transitional operations.

The specific plant configurations required for emergency core cooling system (ECCS) testing and during normal ECCS alignment also involve procedures intended to prevent development of cold overpressurization transients. During these limited periods of operation, the following administrative controls are applied:

A. To preclude inadvertent ECCS actuation during heatup and cooldown, blocking of the safety injection signal actuation logic below 1975 psia is required.

B. During RCS cooldown, closure and power lockout of the accumulator isolation valves as well as power lockout to all ECCS pumps and to the nonoperating charging pumps is required. This provides protection in addition to Step A above. These actions are taken when RCS pressure is approximately 1000 psig, shortly before RHR operation begins ($T \leq$ 350°F).

C. Periodic ECCS pump performance testing must be done during normal power operation or at hot shutdown conditions. Under these conditions (hot and with RCS pressure above the HHSI pump shutoff head) any potential for developing a cold overpressurization transient is precluded.

D. If testing of the charging or HHSI pumps under cold shutdown conditions with the reactor vessel closed is necessary, the procedures require that the pump's discharge valves be closed, to prevent potential pump input to the RCS. The procedures also require valve alignment such as to provide the benefits of the RHRS relief valves.

E. Should cold shutdown testing of the charging or HHSI pumps be desired when the reactor vessel is open to the atmosphere, overpressurization of the RCS is not possible.

F. The "S" signal circuitry testing, if performed during cold shutdown, also requires RHRS valve alignment and power lockout to the HHSI pumps and non-operating charging pumps, to preclude development of cold overpressurization transients.

The above procedures (covering normal operations with a steam bubble, transitional operations where the RCS can be water solid, followed by specific testing operations) provide in-depth cold overpressure prevention measures augmenting the installed overpressure relief system.

5.2.3 Reactor Coolant Pressure Boundary (RCPB) Materials

5.2.3.1 Material Specifications

Typical material specifications used for the principal pressure retaining in Class 1 primary components and for Class 1 and 2 auxiliary components in systems required for reactor shutdown and for emergency core cooling are listed in Table 5.2-1 of this module. Typical material specifications used for the reactor internals required for emergency core cooling, for any mode of normal operation or under postulated accident conditions, and for core structural load bearing members are listed in Table 5.2-2 of this module.

Tables 5.2-1 and 5.2-2 may not be totally inclusive of the material specifications used in the listed applications; however, the listed specifications are representative.

The materials utilized conform to the applicable American Society of Mechanical Engineers (ASME) code rules.

The welding materials used for joining the ferritic base materials of the RCPB conform or are equivalent to ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18, and 5.20. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining the austenitic stainless steel base materials of the RCPB conform to ASME Material Specifications SFA 5.4 and 5.9. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14. They are qualified to the requirements of the ASME Code, Section III.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Chemistry of Reactor Coolant

The RCS water chemistry is selected to minimize corrosion. Routinely scheduled analyses of the coolant chemical composition are performed to verify that the reactor coolant chemistry meets the specifications.

The chemical and volume control system (CVCS) provides a means for adding chemicals to the RCS which perform the following functions:

- Control the pH of the coolant during pre-startup testing and subsequent operation.
- Scavenge oxygen from the coolant during heatup.
- Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during all power operations subsequent to startup.

The normal limits for chemical additives and reactor coolant impurities for power operation are shown in Table 5.2-3 of this module.

The pH control chemical utilized is lithium hydroxide monohydrate, enriched in the lithium-7 isotope to 99.9 percent. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/inconel systems. In addition, lithium-7 is produced in solution from the neutron irradiation of the dissolved boron in the coolant. Detailed procedures for controlling the lithium-7 level in the coolant are discussed in the CVCS section of RESAR-SP/90 PDA Module 13, "Auxiliary Systems".

During reactor startup from the cold condition, hydrazine is employed as an oxygen scavenging agent. The hydrazine solution is introduced into the RCS in the same manner as described above for the pH control agent.

The reactor coolant is treated with dissolved hydrogen to control the net decomposition of water by radiolysis in the core region, and also to react with the oxygen and nitrogen wihch are introduced into the RCS as impurities. Sufficient partial pressure of hydrogen is maintained in the volume control tank so that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant.

Boron, in the chemical form of boric acid, is added to the RCS for long-term reactivity control of the core.

Suspended solids (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the CVCS.

5.2.3.2.2 Compatibility of Construction Materials With Reactor Coolant

All of the ferritic low-alloy and carbon steels which are used in principal pressure retaining applications have corrosion resistant cladding on all surfaces that are exposed to the reactor coolant. The corrosion resistance of the cladding material is at least equivalent to the corrosion resistance of Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy, martensitic stainless steel, and precipitation hardened stainless steel. These corrosion-resistant cladding materials may be subjected to the ASME Code required postweld heat treatment of the ferritic base materials.

Ferritic low-alloy and carbon steel nozzles have safe ends of either stainless steel wrought materials, stainless steel weld metal per analysis A-7 (designated A-8 in the ASME Code), or nickel-chromium-iron alloy weld metal F-Number 43. The latter buttering material requires further safe ending with austenitic stainless steel base material after completion of the post-weld heat treatment when the nozzle is larger than a 4-inch nominal inside diameter and/or the wall thickness is greater than 0.531 inches.

All of the austenitic stainless steel and nickel-chromium-iron alloy base materials with primary pressure retaining applications are used in the solution annealed condition. These heat treatments are as required by the material specifications.

During subsequent fabrications, these materials are not heated above 800°F other than locally by welding operations. The solution annealed surge line material is subsequently formed by hot bending followed by a re-solution annealing heat treatment.

Components with stainless steel sensitized in the manner expected during component fabrication and installation operate satisfactorily under normal plant chemistry conditions in pressurized water reactor (PWR) systems because chlorides, fluorides, and oxygen are controlled to very low levels.

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

In general, all of the materials listed in Table 5.2-1 which are used in principal pressure-retaining applications and are subject to elevated temperature during system operation are in contact with thermal insulation that covers their outer surfaces.

The thermal insulation used on the RCPB is either reflective stainless steel type or made of compounded materials which yield low leachable chloride and/or fluoride concentrations. The compounded materials in the form of blocks, boards, cloths, tapes, adhesives, cements, etc., are silicated to provide protection of austenitic stainless steels against stress corrosion which may result from accidental wetting of the insulation by spillage, minor leakage or other contamination from the environmental atmosphere. Section 1.8 of this module indicates the degree of conformance with Regulatory Guide 1.36, "Non-metallic Thermal Insulation for Austenitic Stainless Steel".

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is anticipated from service experience, such as valve packing, pump seals, etc., only materials which are compatible with the coolant are used. These are as shown in Table 5.2-1. Ferritic materials exposed to coolant leakage can be readily observed as part of the inservice visual and/or nondestructive inspection program to assure the integrity of the component for subsequent service.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness properties of the RCPB components meet the requirements of the ASME Code, Section III, paragraphs NB, and NC-2300, as appropriate.

The fracture toughness properties of the reactor vessel materials are discussed in Section 5.3 of this module.

Limiting steam generator and pressurizer reference temperature for a nil ductility transition (RT_{NDT}) temperatures are guaranteed at 60°F for the base materials and the weldments. These materials meet the 50 ft-lb absorbed energy and 35-mils lateral expansion requirements of the ASME Code, Section III, at 120°F. The actual results of these tests are provided in the ASME material data reports which are supplied for each component and submitted to the owner at the time of shipment of the component.

Calibration of temperature instruments and Charpy impact test machines will be performed to meet the requirements of the ASME Code, Section III, paragraph NB-2360.

Westinghouse has conducted a test program to determine the fracture toughness of low-alloy ferritic materials with specified minimum yield strengths greater than 50,000 psi to demonstrate compliance with Appendix G of the ASME Code, Section III. In this program, fracture toughness properties were determined and shown to be adequate for base metal plates and forgings, weld metal, and heat affected zone metal for higher strength ferritic materials used for components of the RCPB. The results of the program are documented in Reference 1, which has been submitted to the NRC for review.

5.2.3.3.2 Control of Welding

All welding is conducted utilizing procedures qualified according to the rules of Sections III and IX of the ASME Code. Control of welding variables, as well as examination and testing, during procedure qualification and production welding is performed in accordance with ASME Code requirements.

Westinghouse practices for storage and handling of welding electrodes and fluxes comply with ASME Code, Section III, Paragraph NB-2400.

Section 1.8 of this module indicates the degree of conformance of the ferritic materials components of the RCPB with Regulatory Guides 1.34, "Control of Electroslag Weld Properties", 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components", 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel", and 1.71, "Welder Qualification for Areas of Limited Accessibility".

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

Subsections 5.2.3.4.1 through 5.2.3.4.5 of this module address Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel", and present the methods and controls utilized by Westinghouse to avoid sensitization and prevent intergranular attack (IGA) of austenitic stainless steel components. Also, Section 1.8 indicates the degree of conformance with Regulatory Guide 1.44.

5.2.3.4.1 Cleaning and Contamination Protection Procedures

It is required that all austenitic stainless steel materials used in the fabrication, installation and testing of nuclear steam supply components and systems be handled, protected, stored, and cleaned according to recognized and accepted methods which are designed to minimize contamination which could lead to stress corrosion cracking. The rules covering these controls are stipulated in Westinghouse process specifications. As applicable, these process specifications supplement the equipment specifications and purchase order requirements of every individual austenitic stainless steel component or system which Westinghouse procures.

The process specifications which define these requirements and which follow the guidance of the American National Standards Institute (ANSI) N-45 Committee specifications include the following:

Number	Process Specification
82560HM	Requirements for Pressure Sensitive Tapes for Use on Austenitic Stainless Steels.
83336KA	Requirements for Thermal Insulation Used on Austenitic Stainless Steel Piping and Equipment.
83860LA	Requirements for Marking of Reactor Plant Components and Piping.
84350HA	Site Receiving Inspection and Storage Requirements for Systems, Material and Equipment.
84351NL	Determination of Surface Chloride and Fluoride on Austenitic Stainless Steel Materials.
85310QA	Packaging and Preparing Nuclear Components for Shipment and Storage.
292722	Cleaning and Packaging Requirements of Equipment for Use in the NSSS.

Number	Process Specification
597756	Pressurized Water Reactor Auxiliary Tanks Cleaning Procedures.
597760	Cleanliness Requirements During Storage Construction, Erection and Start-Up Activities of Nuclear Power System.

Section 1.8 includes the degree of conformance of the austenitic stainless steel components of the RCPB with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants".

5.2.3.4.2 Solution Heat Treatment Requirements

The austenitic stainless steels listed in Tables 5.2-1 and 5.2-2, are utilized in the final heat-treated condition required by the respective ASME Code, Section II materials specification for the particular type or grade of alloy.

5.2.3.4.3 Material Testing Program

Westinghouse practice is that austenitic stainless steel materials of product forms with simple shapes need not be corrosion tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as all plates, sheets, bars, pipe and tubes, as well as forgings, fittings, and other shaped products which do not have inaccessible cavities or chambers that would preclude rapid cooling when water quenched. When testing is required, the tests are performed in accordance with ASTM A 262, Practice A or E, as amended by Westinghouse Process Specification 84201MW.

5.2.3.4.4 Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels

Unstabilized austenitic stainless steels are subject to intergranular attack IGA provided that three conditions are present simultaneously. These are:

- o An aggressive environment, e.g., an acidic aqueous medium containing chlorides or oxygen.

- A sensitized steel.

- A high temperature.

If any one of the three conditions described above is not present, IGA will not occur. Since high temperatures cannot be avoided in all components in the NSSS, reliance is placed on the elimination of the other two conditions to prevent IGA on wrought stainless steel components.

This is accomplished by:

- Control of primary water chemistry to ensure a benign environment.

- Utilization of materials in the final heat-treated condition and the prohibition of subsequent heat treatments in the 800° and 1500°F temperature range.

- Control of welding processes and procedures to avoid heat-affected zone sensitization.

- Confirmation that the welding procedures used for the manufacture of components in the primary pressure boundary and the reactor internals do not result in the sensitization of heat affected zones.

Further information on each of these steps is provided in the following paragraphs.

The water chemistry in the RCS is controlled to prevent the intrusion of aggressive species. In particular, the maximum permissible oxygen and chloride concentrations are 0.005 ppm and 0.15 ppm, respectively. Table 5.2-3 of this module lists the recommended reactor coolant water chemistry specifications. The precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage are stipulated in the appropriate process specifications. The use of hydrogen overpressure precludes the presence of oxygen during operation. The effectiveness of these controls has been demonstrated by both laboratory tests and operating experience. The long-term exposure of severely sensitized stainless steels to reactor coolant environments in early Westinghouse PWRs has not resulted in any sign of IGA. Reference 2 describes the laboratory experimental findings and reactor operating experience. The additional years of operation since the issuing of reference 2 have provided further confirmation of the earlier conclusion that severely sensitized stainless steels do not undergo any IGA in Westinghouse PWR coolant environments.

Although there is no evidence that PWR coolant water attacks sensitized stainless steels, Westinghouse considers it good metallurgical practice to avoid the use of sensitized stainless steels in the NSSS components. Accordingly, measures are taken to prohibit the purchase of sensitized stainless steels and to prevent sensitization during component fabrication. Wrought austenitic stainless steel stock used for components that are part of:

- o The RCPB.
- o Systems required for reactor shutdown.
- o Systems required for emergency core cooling.
- o Reactor vessel internals relied upon to permit adequate core cooling for normal operation or under postulated accident conditions.

The wrought austenitic stainless steel stock is utilized in one of the following conditions:

- o Solution annealed and water quenched.
- o Solution annealed and cooled through the sensitization temperature range within less than approximately 5 minutes.

It is generally accepted that these practices will prevent sensitization. Westinghouse has verified this by performing corrosion tests on wrought material as it was received.

The heat-affected zones of welded components must, of necessity, be heated into the sensitization temperature range, 800° to 1500°F. However, severe sensitization, i.e., continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion, can be avoided by controlling welding parameters and welding processes. The heat input and associated cooling rate through the carbide precipitation range are of primary importance. Westinghouse has demonstrated this by corrosion testing a number of weldments.

Heat input is calculated according to the formula:

$$H = \frac{(E)(I)(60)}{S}$$

where:

H = joules/in.

E = volts.

I = amperes.

S = travel speed (in./min).

Of 25 production and qualification weldments tested, representing all major welding processes, and a variety of components, and incorporating base metal thicknesses from 0.?0 to 4.0 in., only portions of two were severely sensitized. Of these, one involved a heat input of 120,000 J, and the other involved a heavy socket weld in relatively thin walled material. In both cases, sensitization was caused primarily by high heat inputs relative to the section thickness. In only the socket weld did the sensitized condition exist at the surface, where the material is exposed to the environment. The component has been redesigned, and a material change has been made to eliminate this condition.

The heat input in all austenitic pressure boundary weldments has been controlled by:

- Prohibiting the use of block welding.
- Limiting the maximum interpass temperature to 350°F.
- Exercising approval rights on all welding procedures.

5.2.3.4.5 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures

As described in the previous section, it is not normal Westinghouse practice to expose unstabilized austenitic stainless steels to the sensitization range of 800° to 1500°F during fabrication into components. If, during the course of fabrication, the steel is inadvertently exposed to the sensitization temperature range, 800° to 1500°F, the material may be tested in accordance with ASTM A 262, as amended by Westinghouse process specification 84201MW, to verify that it is not susceptible to IGA, except that testing is not required for:

A. Cast metal or weld metal with a ferrite content of 5 percent or more,

B. Material with a carbon content of 0.03 percent or less that is subjected to temperatures in the range of 800° to 1500°F for less than 1 hour,

C. Material exposed to special processing provided the processing is properly controlled to develop a uniform product and provided that adequate documentation exists of service experience and/or test data to demonstrate that the processing will not result in increased susceptibility to intergranular stress corrosion.

If it is not verified that such material is not susceptible to IGA, the material will again be solution annealed and water quenched or rejected.

5.2.3.4.6 Control of Welding

The following paragraphs address Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal", and present the methods used, and the verification of these methods, for austenitic stainless steel welding.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking in the weld. Although published data and experience have not confirmed that fissuring is detrimental to the quality of the weld, it is recognized that such fissuring is undesirable in a general sense. Also, it has been well documented in the technical literature that the presence of delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. However, there is insufficient data to specify a minimum delta ferrite level below which the material will be prone to hot cracking. It is assumed that such a minimum lies somewhere between 0 and 3 percent delta ferrite.

The scope of these controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated or stamped in accordance with the ASME Code, Section III, Class 1, 2, and core support components. Delta ferrite control is appropriate for the above welding requirements, except where no filler metal is used or where for other reasons such control is not applicable. These exceptions include electron beam welding, autogenous gas shielded tungsten arc welding, explosive welding, and welding using fully austenitic welding materials.

The fabrication and installation specifications require welding procedure and welder qualification in accordance with Section III, and include the delta ferrite determinations for the austenitic stainless steel welding materials that are used for welding qualification testing and for production processing. Specifically, the undiluted weld deposits of the "starting" welding materials are required to contain a minimum of 5 percent delta ferrite (the equivalent ferrite number may be substituted for percent delta ferrite) as determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III. When new welding procedure qualification tests are evaluated for these applications, including repair welding, they are performed in accordance with the requirements of Section III and IX.

The results of all the destructive and nondestructive tests are reported in the procedure qualification record in addition to the information required by Section III.

The "starting" welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of Section III. The austenitic stainless steel welding material conforms to ASME weld metal analysis A-7 (designated A-8 in the 1974 Edition of the ASME Code), Type 308 or 308L for all applications. Bare weld filler metal, 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 according to Section III. Weld filler metal materials used in flux shielded welding processes conform to ASME SFA 5.4 or 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 according to Section III. Welding materials are tested using the welding energy inputs to be employed in production welding.

Combinations of approved heat and lots of "starting" welding materials are used for all welding processes. The welding quality assurance program includes identification and control of welding material by lots and heats as appropriate. All of the weld processing is monitored according to approved inspection programs which include review of "starting" materials, qualification records and welding parameters. Welding systems are also subject to:

- o Quality assurance audit including calibration of gauges and instruments.
- o Identification of "starting" and completed materials.

- o Welder and procedure qualifications.

- o Availability and use of approved welding and heat-treating procedures.

- o Documentary evidence of compliance with materials, welding parameters, and inspection requirements.

Fabrication and installation welds are inspected using nondestructive examination methods according to Section III rules.

To ensure the reliability of these controls, Westinghouse has completed a delta ferrite verification program, described in reference 3. This program has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative for conformance with the NRC Interim Position on Regulatory Guide 1.31. The Regulatory Staff's acceptance letter and topical report evaluation were received on December 30, 1974. The program results, which do support the hypothesis presented in reference 3, are summarized in reference 4.

Section 1.8 indicates the degree of conformance of the austenitic stainless steel components of the RCPB with Regulatory Guides 1.34, "Control of Electroslag Properties", and 1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

Inservice inspection and testing of Class 1 pressure-retaining components such as vessels, piping, pumps, valves, bolting, and supports within reactor coolant pressure boundary shall be performed in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Code including addenda in accordance with 10 CFR 50.55a(g)(4)(i), with certain exceptions whenever specific written relief is granted by the Nuclear Regulatory Commission (NRC) in accordance with 10 CFR 50.55a(g)(6)(i). The inservice testing of pumps and valves shall be in accordance with the requirements of Articles IWP and IWV of the code as discussed in Subsection 3.9.6 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

A preservice inspection program (nondestructive examination) and a preservice test program (pumps and valves) for the WAPWR will be defined which will comply with applicable inservice inspection provisions of 10 CFR 50.55a(2). The preservice programs will provide details of areas subject to examination, as well as the method and extent of preservice examinations. Inservice programs will detail the areas subject to examination and method, extent, and frequency of examinations after startup.

5.2.4.1 System Boundary Subject to Inspection

In addition to the reactor pressure vessel, all Class 1 components such as vessels, piping, pumps, valves, bolting, and supports shall be inspected to the extent practical, in accordance with Article IWB of ASME Code, Section XI. Class 1 pressure-retaining components and their specific boundaries are shown on individual system piping and instrumentation diagrams.

5.2.4.2 Arrangement and Accessibility

The physical arrangement of components shall be designed to allow personnel and equipment access to the extent practical to perform the required inservice examinations.

5.2.4.3 Examination Techniques and Procedures

The visual, surface, and volumetric examination techniques, procedures and special techniques shall be in accordance with the requirements of subarticle IWA-2200 and Table IWB-2500-1 of the ASME Code, Section XI. The liquid penetrant method or the magnetic particle method will be used for surface examinations. Radiography and/or ultrasonic techniques, including manual and remote, will be used for volumetric examinations. Other examination techniques may be used provided that the results are demonstrated to be equivalent or superior to the above techniques.

5.2.4.4 Inspection Intervals

Inspection intervals will be as defined in subarticles IWA-2400 and IWB-2400 of ASME Code, Section XI.

5.2.4.5 Examination Categories and Requirements

The examination categories and requirements shall be in accordance with subarticle IWB-2500 and Table IWB-2500-1 of ASME Code, Section XI. The preservice examinations comply with IWB-2200.

5.2.4.6 Evaluation of Examination Results

Examination results shall be evaluated in accordance with IWB-3000, with flaw indications evaluated in accordance with IWB-3400 and Table IWB-3410. Repair procedures will be in accordance with IWB-4000 of ASME Code, Section XI.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System pressure tests shall comply with IWA-5000 and IWB-5000 of the ASME Code, Section XI.

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary (RCPB) leakage detection systems monitor leaks from the reactor coolant and associated systems inside containment. These systems provide information which permit the plant operators to take corrective action if a leak is evaluated as detrimental to the safety of the facility.

5.2.5.1 Design Bases

The leak detection systems are designed in accordance with the requirements of 10 CFR 50 and General Design Criterion 30 to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage. The systems conform with Regulatory Guide 1.45. Main systems that monitor the environmental condition of the containment include the sump level monitoring systems and the containment fan cooler condensate measuring system. In addition, the humidity, temperature, pressure, and radiogas monitors provide indirect indication of leakage to the containment.

Associated systems and components connected to the reactor coolant system have intersystem leakage monitoring features.

These leakage detection systems are qualified for all seismic events not requiring a plant shutdown. The airborne radioactivity monitoring system is qualified for a safe shutdown earthquake (SSE).

5.2.5.1.1 Leakage Classification

RCPB leakage is classified as either identified or unidentified leakage. Identified leakage includes: (a) leakage into closed systems, such as pump seal or valve packing leakages that are captured, flow metered, and directed to a sump or collecting tank, (b) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of unidentified leakage monitoring systems or not to be from a flaw in the RCPB, or (c) leakage into auxiliary systems and secondary systems. Unidentified leakage is all other leakage.

5.2.5.1.2 Limits for Reactor Coolant Leakage

Limits for reactor coolant leakage are identified in the Technical Specifications.

5.2.5.2 Identified Intersystem Leakage Detection

Identified leakage, such as pump seal or valve packing leakage, is also directed to the reactor coolant drain tank. The tank contents are monitored for pressure, temperature and level to help assess the presence of leakage.

Identified leakage, such as leakage past the pressurizer safety valves or power-operated relief valves (PORVs), is directed to the pressurizer relief tank. This leakage is monitored by temperature instrumentation in the valve discharge piping and by tank pressure, temperature, and level instrumentation. Leakage collected in the pressurizer relief tank is directed to the reactor coolant drain tank for subsequent treatment and discharge.

An important identified leakage path for reactor coolant into other systems is through the steam generator tubes to the secondary side of the steam generator. Identified leakage through the steam generators is detected by means of the steam generator liquid sample or condenser air ejector radiation monitors.

Unidentified leakage which collects on the containment floor is directed to the reactor coolant drain tank.

Auxiliary systems connected to the RCPB incorporate design and administrative provisions that serve to limit leakage. These provisions include isolation valves designed for low seat leakage, periodic testing of RCPB check valves and inservice inspection. Leakage is detected by increasing auxiliary tank level, by temperature, and/or pressure indications or by lifting of relief valves. The latter is accompanied by increasing values of monitored parameters in the relief valve discharge path. These systems are isolated from the RCS by normally closed valves and/or check valves when not in use.

5.2.5.2.1 Description and Operation of Identified Leakage Detection Features

A. Residual Heat Removal System (RHRS) (Suction Side)

The RHRS which is part of the integrated safeguards system, is isolated from the RCS on the suction side by motor-operated valves 9000 A-D and 9001 A-D. Significant leakage past these valves (two in series) is detected by lifting of relief valves 9021 A-D, accompanied by increasing pressurizer relief tank level, pressure, and temperature indications and alarms on the main control board.

B. Safety Injection Accumulators

The accumulators are isolated from the RCS by check valves 8948 A-D and 8956 A-D. Leakage past these valves (two in series) and into the accumulator subsystem is detected by redundant control room accumulator pressure and level indications and alarms.

C. Core Reflood Tanks

The core reflood tanks are isolated from the RCS by check valves 9098 A-D and 9099 A-D (two in series). Leakage past these valves and into the core reflood tank subsystem is detected by redundant core reflood tank pressure and level indications and alarms.

D. HHSI/RHR Discharge Subsystem

The HHSI/RHR pump discharge portion of the ISS is isolated from the RCS by check valves 8808 A-D and 8809 A-D. Leakage past these valves will eventually pressurize this portion of the system and result in lifting of the relief valves 9020 A-D. Relief valve lifting is accompanied by alarms of increasing boron recycle holdup tank levels.

E. Head Gasket Monitoring Connections

The reactor vessel flange and head are sealed by two metallic O-rings. These gaskets are of the hollow self-energizing type in which pressure of the fluid being sealed enters the interior of the gasket. The O-rings are fastened to the closure head by a mechanical connection to facilitate removal when the head is off the vessel.

Seal Leakage is detected by means of two leak-off connections: one between the inner and outer O-ring, and one outside the outer O-ring. A manual isolation valve is installed just outside the missile barrier in each leak-off line. Downstream of these valves the lines are joined before being routed to the reactor coolant drain tank in the waste processing system. A solenoid operated isolation valve, actuated from the control board, is installed in the common line. During normal plant operation, the leak-off piping is aligned such that leakage across the inner O-ring passes through valves 8069A and 8032 into the drain tank. A surface mounted resistance temperature detector, installed on the bottom of the common pipe, signals leakage at an alarm setpoint. A local sample connection outside the missile barrier, containing isolation valve 8076, is provided to confirm and establish the magnitude of leakage too small to quantify by RCDT monitoring.

Once inner O-ring leakage is detected, valve 8069B should be opened and valve 8069A closed so that possible leakage across the outer O-ring would be monitored.

In addition, during plant refueling operations both the inner and outer reactor vessel flange leak-off valves are closed. This prevents possible gas leakage from the reactor coolant drain tank to the containment atmosphere. Refer to Figure 5.1-1 for the flow diagram representation.

The reactor vessel is the only flanged vessel within the RCPB that is provided with leak-off collection provisions.

Leakage past the reactor vessel head gaskets results in temperature indication and alarm in the control room.

F. Component Cooling Water System

Leakage from the RCS to the component cooling water system (CCWS), which services all RCPB associated components that require cooling, is detected by the CCWS radiation monitoring equipment and/or increasing surge tank level. Components serviced by this system includes reactor coolant pump thermal barriers, RHR heat exchangers, letdown and excess letdown heat exchangers and the reactor coolant pump seal injection heat exchangers.

5.2.5.3 Unidentified Leakage Detection

Normally, unidentified leakage from the RCPB is very low. The RCPB is all-welded with the exception of the pressurizer safety valves, reactor vessel head and pressurizer manways which are flanged.

In general, valves in the RCS that are 2 inches nominal size and under are of the packless type. All valves larger that 2 inches have dual packing with a leak-off connection between the two packing sets. Leakage past the first packing set is piped to the reactor coolant drain tank.

Primary indications of unidentified coolant leakage to the containment are provided by air particulate radioactivity monitors, gaseous radioactivity monitors, fan cooler condensate flow monitors, and containment sump level monitors. In normal operation, these primary monitors show a background level that is indicative of the normal level of unidentified leakage inside the containment. Variations in airborne radioactivity or specific humidity above the normal level signify an increase in unidentified leakage rates and signal to the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage.

Unidentified leakage from the RCPB may also be indicated by increasing charging pump flowrate compared with normal RCS inventory changes and by unscheduled increases in reactor makeup water usage.

Reactor coolant inventory monitoring provides an indication of system leakage. Net level changes in volume control tank is indicative of system leakage, since the chemical and volume control system (CVCS) is a closed loop connected to the RCS. Monitoring net makeup to the CVCS, as well as net collected leakage, provides an important method of obtaining information for use in establishing a water inventory balance. An abnormal increase in makeup water requirements or a significant change in the water inventory balance can be indicative of increased system leakage.

The above methods are supplemented by visual and ultrasonic inspections of the RCPB during plant shutdown periods in accordance with the inservice inspection program.

5.2.5.3.1 Description and Operation of Main Unidentified Leak Detection Systems

Systems employed for detecting leakage to the containment from unidentified sources are:

- Containment airborne particulate radioactivity monitor.
- Containment gaseous radioactivity monitor.
- Containment fan cooler condensate flow monitor.
- Containment sump level monitor.

Additionally; humidity, temperature, and pressure monitoring of the containment atmosphere are used for alarms and indirect indication of leakage to the containment.

A. Containment Airborne Particulate Radioactivity Monitoring System

An air sample is drawn outside the containment into a closed system by a sample pump and is then consecutively passed through a particulate filter with detectors and a gaseous monitor chamber with detector. The filter can collect approximately 99 percent of the particulate matter greater than 1 μm in size. The sample transport system includes:

- o A pump to obtain the air sample.
- o A flow control valve to provide flow adjustment.
- o A flow meter to indicate the flowrate.
- o A flow alarm assembly to provide high and low flow alarm signals.

The particulate filter is continuously monitored by a scintillation crystal with a photomultiplier tube that provides an output signal proportional to the activity collected on the filter.

Particulate activity can be correlated with the coolant fission and corrosion product activities. Any increase of more than two standard deviations above the count rate for background would indicate a possible leak. The total particulate activity concentration above background, due to an abnormal leak and natural decay, increases almost linearly with time for the first several hours after the beginning of a leak.

The activity is indicated and recorded in the main control room. An alarm is actuated on high activity. Local indicators provide the operational status of supporting equipment such as pumps, motors, and flow and pressure controllers.

The leakage flowrate can be determined from the measured increase in the count rate above the initial value. The initial count rate is established based on the specific background radioactivity present before the leakage begins. The background activity is dependent upon power level, percent failed fuel, coolant crud levels and natural radioactivity.

B. Containment Gaseous Radioactivity Monitoring System

The containment gaseous radioactivity monitor determines gaseous radioactivity in the containment by monitoring continuous air samples from the containment atmosphere. After passing through the gas monitor, the sample is returned via the closed system to the containment atmosphere. Each sample is continuously mixed in a fixed, shielded volume where its activity is monitored.

Gaseous radioactivity can be correlated with the gaseous activity of the reactor coolant. Any increase more than two standard deviations above the count rate for background would indicate a possible leak. The total gaseous activity level above background increases almost linearly for the first several hours after the beginning of the leak.

The detector outputs are transmitted to the radiation monitoring system cabinets in the control room where the activity is indicated by meters and recorded. An alarm is actuated on high activity. Local indicators provide the operational status of the supporting equipment.

The leakage flowrate can be determined from the measured increase in the count rate above the initial value. The initial value is based on the specific background radioactivity present before the leakage begins. The background activity is dependent principally on power level and percent failed fuel.

The containment purge system radioactivity monitors serve as backup to the containment air particulate and gaseous airborne radioactivity monitoring system when the purge system is in operation.

The containment purge monitors function in the same manner as the containment air particulate and gaseous radioactivity monitors, except that the purge monitors sample the containment purge exhaust line.

C. Containment Fan Cooler Condensate Monitoring System

The condensate monitoring system permits measurements of the liquid runoff from the containment fan cooler units. It consists of a containment fan cooler drain collection header, a vertical standpipe, valving, and standpipe level instrumentation. The condensation from the containment fan coolers flows via the collection header to the vertical standpipe. A differential pressure transmitter provides standpipe level signals. The system provides measurements of low leakages by monitoring standpipe level increase versus time, and of larger runoff rates by correlation of stand pipe level with runoff flow through a fixed resistance (valve or orifice) under steady state conditions.

The condensate flowrate is a function of containment humidity, fan cooler cooling water temperature, and containment purge rate.

Drainage flowrate from the units due to normal condensation is calculated for the ambient (background) atmospheric conditions present within the containment. With the initiation of an additional or abnormal leak, the containment atmosphere humidity and condensation runoff rate both begin to increase, the water level rises in the vertical stand pipe, and the high condensate flow alarm is actuated.

The rate of leakage can be determined by the operating personnel when the precise cooling water, outside air and containment air temperatures, and the outside relative humidity, are known.

D. Containment Sump Level Monitoring System

Since a leak in the primary system would result in reactor coolant flowing into the containment normal or reactor cavity sumps, leakage would be indicated by a measurable level increase in the sump. Indication of increasing sump level is transmitted to the control room level indicator by means of a sump level transmitter. The system provides measurements of low leakages by monitoring level increase versus time.

The actual reactor coolant leakage rate can be approximated from the increase above the normal rate of change of sump level. A check of other instrumentation would be required to eliminate possible leakage from non-radioactive systems as a cause of an increase in sump level.

Under normal conditions, the containment normal and reactor cavity sump pumps operate very infrequently. Gross leakage can be surmised from unusual frequency of pump operation. Sump level and pump running indication are provided in the control room to alert the operators.

5.2.5.3.2 Additional Unidentified Leakage Detection Methods

Other methods available for detecting leakage are:

A. Charging Pump Operation

If a gross increase in reactor coolant leakage occurs, the level would decrease and the flowrate of the charging pump would automatically increase to try to maintain pressurizer level. Charging pump discharge flow indication is provided in the control room.

The increase in leakage rate can be estimated by the amount that charging pump flowrate increases above the normal flowrate needed to maintain constant pressurizer level. Any significant increase in the charging flowrate is an indication of a possible leak.

B. Containment Humidity Monitoring System

The containment humidity monitoring system, utilizing temperature-compensated humidity detectors, is provided to determine the water vapor content of the containment atmosphere. An increase in the humidity of the containment atmosphere indicates release of water vapor within the containment. The humidity monitor supplements the condensate monitor. It is most sensitive under conditions when there is no condensation in the fan coolers.

C. Liquid Inventory

The operators can surmise gross leakage from changes in the reactor coolant inventory. Noticeable decreases in the volume control tank level not associated with known changes in operation are investigated. Likewise, makeup water usage information which is available from the plant computer is checked frequently for unusual makeup rates not due to plant operations.

5.2.5.4 Safety Evaluation

The leak detection system has no safety design basis; however, the containment atmosphere radioparticulate and radiogas monitors are qualified for an SSE per the recommendation of Regulatory Guide 1.45.

5.2.5.5 Tests and Inspections

Periodic testing of leakage detection systems is conducted to verify the operability and sensitivity of detection equipment. These tests include instrumentation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks.

The humidity detector and condensate measuring system are also periodically tested to ensure proper operation and to verify sensitivity.

5.2.5.6 Instrumentation Applications

The following indications are provided in the control room to allow operating personnel to monitor for leakage:

A. Containment air particulate monitor - air particulate activity.

B. Containment gaseous activity monitor - gaseous activity.

C. Containment fan cooler condensate monitoring system - standpipe level.

D. Containment humidity measuring system - containment humidity.

E. Containment normal sump level and reactor cavity sump level.

F. Gross leakage detection methods - charging pump flowrate, letdown flowrate, pressurizer level, and reactor coolant temperatures are available for the charging pump flow method. Containment sump levels and pump operation are available for the sump pump operation method. Total integrated makeup water flow is available from the plant computer for liquid inventory.

5.2.6 REFERENCES

1. Logsdon, W. A., Begley, J. A., and Gottshall, C. L., "Dynamic Fracture Toughness of ASME SA508 Class 2a and ASME SA533 Grade A Class 2 Base and Heat Affected Zone Material and Applicable Weld Metals," WCAP-9292, March 1978.

2. Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970 and WCAP-7735 (Non-Proprietary), August 1971.

3. Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975.

4. Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.

PRIMARY AND AUXILIARY COMPONENTS
TYPICAL MATERIAL SPECIFICATIONS

Reactor Vessel Components	
Shell and head plates (other than core region)	SA-533, Grade A, B or C, Class 1 or 2 (vacuum treated)
Shell plates (core region)	SA-533, Grade A or B, Class 1 (vacuum treated)
Shell, flange and nozzle forgings, nozzle safe ends	SA-508, Class 2 or 3; SA-182, Grade F304 or F316
Control rod drive mechanism (CRDM) and/or ECCS appurtenances upper head	SB-166 or SB-167 and SA-182, Grade F304
Instrumentation tube appurtenances, lower head	SB-166 or SB-167 and SA-182, Grade F304, F304L or F316
Closure studs, nuts, washers, inserts and adaptors	SA-540, Class 3, Grade B23 or B24
Core support pads	SB-166 with carbon less than 0.10 percent
Monitor tubes and vent pipe	SA-312 or SA-376, Grade TP304 or TP316 or SB-166 or SB-167 or SA-182, Grade F316
Vessel supports, seal ledge and heat lifting lugs	SA-516, Grade 70 (quenched and tempered) or SA-533, Grade A, B or C, Class 1 or 2 (vessel supports may be of weld metal buildup of equivalent strength of the nozzle material)
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Steam Generator Components	
Pressure plates	SA-533, Grade A, B, or C, Class 1 or 2
Pressure forgings (including nozzles and tube sheet)	SA-508, Class 1, 2, 2a, or 3

Nozzle safe ends	Stainless Steel Weld Metal Analysis A-8
Channel heads	SA-533, Grade A, B or C, Class 1 or 2 or SA-216, Grade WCC
Tubes	SB-163 (Ni-Cr-Fe annealed) Alloy 600 Alloy 690 (Code Case N-20)
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Closure bolting	SA-193, Grade B7
Pressurizer Components	
Pressure plates	SA-533, Grade A, Class 2
Pressure forgings	SA-508, Class 2 or 2a
Nozzle safe ends	SA-182, Grade F316L
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Closure bolting	SA-193, Grade B7
Reactor Coolant Pump	
Pressure forgings	SA-182, Grade F304, F316, F347 or F348
Pressure casting	SA-351, Grade CF8, CF8A or CF8M
Tube and pipe	SA-213; SA-376 or SA-312, Seamless, Grade TP304 or TP316
Pressure plates	SA-240, Type 304 or 316
Bar material	SA-479, Type 304 or 316
Closure bolting	SA-193; SA-320; SA-540 or SA-453, Grade 660
Flywheel	SA-533, Grade B, Class 1

TABLE 5.2-1 (Sheet 3 of 5)

<u>Reactor Coolant Piping</u>	
Reactor coolant pipe	SA-351, Grade CF3A, CF8A, CF3M, CF8M, SA 182, TP 304, 316, 304N, 316N (Code Case 1423-2)
Reactor coolant fittings, branch nozzles	SA-351, Grade CF8A and SA-182, Grade 316N
Surge line	SA-376, Grade TP304, TP316 or F304N
Auxiliary piping	SA-312 and SA-376 Grades TP304 and TP316 to ANSI B36.10 or B36.19
Socket weld fittings	ANSI B16.11
Piping flanges	ANSI B16.5
<u>Full Length CRDM</u>	
Latch housing	SA-182, Grade F304 or SA-351, Grade CF8
Rod travel housing	SA-182, Grade F304 or SA-336, Class 304
Cap	SA-479, Type 304
Welding materials	Stainless Steel Weld Metal Analysis A-8
<u>Valves</u>	
Bodies	SA-182, Grade F316 or SA-351, Grade CF8 or CF8M
Bonnets	SA-182, Grade F316 or SA-351, Grade CF8 or CF8M
Discs	SA-182, Grade F316 or SA-564 Grade 630, or SA-351, Grade CF8 or CF8M
Stems	SA-182, Grade F316 or SA-564, Grade 630

Pressure retaining bolting	SA-453, Grade 660
Pressure retaining nuts	SA-453, Grade 660 or SA-194, Grade 6
Auxiliary Heat Exchangers	
Heads	SA-240, Type 304
Nozzle necks	SA-182, Grade F304; SA-240 and 312, Type 304
Tubes	SA-213, Grade TP304; SA-249, Type 304
Tube sheets	SA-182, Grade F304; SA-240, Type 304 and SA-516 GR 70 clad with Stainless Steel Weld Metal Analysis A-8
Shells	SA-240 and SA-312, Grade TP304
Auxiliary Pressure Vessels, Tanks, Filters, etc.	
Shells and heads	SA-240, Type 304 or SA-351 Grade CFA8, or SA264 (consisting of SA-537, Class 1 with Stainless Steel Weld Metal Analysis A-8 Cladding)
Flanges and nozzles	SA-182, Grade F304 and SA-105 or SA-350, Grade LF2 and LF3 with Stainless Steel Weld Metal Analysis A-8 Cladding
Piping	SA-312 Type 304 and SA-240, Grade TP304 or TP316
Pipe fittings	SA-403, Grade WP304 Seamless
Closure bolting and nuts	SA-193, Grade B7 and SA-194, Grade 2H
Auxiliary Pumps	
Pump casing and heads	SA-351, Grade CF8 or CF8M; SA-182, Grade F304 or F316
Flanges and nozzles	SA-182, Grade F304 or F316; SA-403 Grade WP316L Seamless

TABLE 5.2-1 (Sheet 5 of 5)

Piping	SA-312, Grade TP304 or TP316 Seamless
Stuffing or packing box cover	SA-351, Grade CF8 or CF8M; SA-240, Type 304 or 304L or 316
Pipe fittings	SA-403, Grade WP316L Seamless
Closure bolting and nuts	SA-193, Grade B6, B7 or B8M; SA-194, Grade 2H or 8M; SA-453 Grade 660, and Nuts, SA-194, Grade 2H, 6 and 8 M

TABLE 5.2-2

REACTOR VESSEL INTERNALS MATERIAL SPECIFICATIONS

Forgings	SA-182, Grade F304
Plates	SA-240, Type 304
Pipes	SA-312, Grade TP304 Seamless or SA-376, Grade TP304
Tubes	SA-213, Grade TP304 or ASTM A-511 Grade MT304, Code Case 1618
Bars	SA-479, Type 304
Castings	SA-351, Grade CF8
Bolting	SA-193, Class 2 (65-90 YS/90 MTS) Code Case 1618 Inconel-750; SA-637, Grade 688, Type 2
Nuts	SA-193, Grade B8
Locking devices	SA-479, Type 304

TABLE 5.2-3 (Sheet 1 of 2)

RECOMMENDED[a] REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical conductivity	Determined by the concentration of boric acid and alkali present.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C.
Oxygen[b]	0.005 ppm, maximum
Chloride[c]	0.15 ppm, maximum
Fluoride[c]	0.15 ppm, maximum
Hydrogen[d]	15 to 50 cm^3 (STP)/kg H_2O
Suspended solids[e]	1.0 ppm, maximum
pH control agent (Li^7OH)[f]	0.7 to 2.2 ppm as Li
Boric acid	Variable from 0 to 4000 ppm as B
Silica[g]	0.2 ppm, maximum
Aluminum[g]	0.05 ppm, maximum
Calcium[g]	0.05 ppm, maximum
Magnesium[g]	0.05 ppm, maximum

NOTES:

(a) Refer to the Technical Specifications for required reactor coolant chemistry limits.

(b) Oxygen concentration must be controlled to less than 0.1 ppm in the reactor coolant at temperatures above 180°F by scavenging with hydrazine. During power operation with the specified hydrogen concentration maintained in the coolant, the residual oxygen concentration must not exceed 0.005 ppm.

TABLE 5.2-3 (Sheet 2 of 2)

(c) Halogen concentrations must be maintained below the specified values at all times regardless of system temperature.

(d) Hydrogen must be maintained in the reactor coolant for all plant operations with nuclear power above 1 MWt. The normal operating range should be 30- to 40-$cm^3/kg\ H_2O$.

(e) Solids concentration determined by filtration through filter having 0.45-μm pore size.

(f) The specified limits for lithium hydroxide must be established for pre-startup testing prior to heatup beyond 150°F. During cold hydrostatic testing and hot functional testing in the absence of boric acid, the reactor coolant limits for lithium hydroxide must be maintained to povide inhibition of halogen stress corrosion cracking. Upon plant restart, the lithium hydroxide limits should be established at 180°F.

(g) These limits are included in the table of reactor coolant specifications as recommended standards for monitoring coolant purity. Establishing coolant purity within the limits shown for these species is judged desirable with the current data base to minimize fuel clad crud deposition which affects the corrosion resistance and heat transfer of the clad.

FIGURE 5.2-1 RCS PIPING AND INSTRUMENTATION DIAGRAM
(SHEETS 1 THROUGH 3 PROPRIETARY)

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Material Specifications

Material specifications will be in accordance with the American Society of Mechanical Engineers (ASME) Code requirements and are given in Subsection 5.2.3.

The ferritic material of the reactor vessel beltline will be restricted to the following maximum limits of copper and phosphorus to reduce sensitivity to irradiation embrittlement in service:

Element	Base Metal (percent)	As Deposited Weld Metal (percent)
Copper	0.10 (ladle) 0.12 (check)	0.10
Phosphorus	0.12 (ladle) 0.17 (check)	0.020
Vanadium	0.05 (check) as residual	0.05 (as residual)

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

A. The vessel is Safety Class 1. Design and fabrication of the reactor vessel will be carried out in strict accordance with ASME Code, Section III, Class 1 requirements. The vessel shells, flanges, and nozzles will be manufactured as forgings. The cylindrical portion of the vessel is made of several shells, joined by full penetration girth weld seams. The heads will be made from forgings or dished plates. The reactor vessel parts are joined by welding, using the single or multiple wire submerged arc and the shielded metal arc processes.

B. The use of severely sensitized stainless steel as a pressure boundary material has been prohibited and has been eliminated by either a select choice of material or by programming the method of assembly.

C. At all locations in the reactor vessel where stainless steel and Inconel are joined, the final joining beads are Inconel weld metal in order to prevent cracking.

D. The stainless steel clad surfaces are sampled to ensure that composition requirements are met.

E. Freedom from underclad cracking is ensured by special evaluation of the procedure qualification for cladding applied on low-alloy steel (SA-508, Class 2).

F. Minimum preheat requirements have been established for pressure boundary welds using low-alloy material. The preheat is maintained until either an intermediate postweld heat treatment or a full postweld heat treatment is completed or until the completion of welding.

G. A field weld is made, after the reactor vessel has been set, to install the permanent reactor vessel cavity seal ring. This stainless steel filler weld joins the seal ring to the reactor vessel seal ledge. A minimum preheat is specified for this weld in compliance with the ASME Code requirements.

5.3.1.3 Special Methods for Nondestructive Examination

During the fabrication sequence, the reactor vessel and its appurtenances are subject to nondestructive examination (NDE) as specified in Section III of the ASME Boiler and Pressure Vessel Code. In addition, numerous supplementary examinations are imposed during vessel manufacture. The NDE of the reactor vessel is discussed in the following paragraphs. The reactor vessel examination requirements are summarized in Table 5.3-1.

5.3.1.3.1 Ultrasonic Examination

The ultrasonic examination requirements listed below are in addition to those specified by Section III of the ASME Boiler and Pressure Vessel Code.

A. All full penetration ferritic pressure boundary welds in the reactor vessel are ultrasonically examined. Complete examinations are performed twice during the vessel manufacturing sequence, once upon completion of welding and intermediate heat treatment and again after the vessel hydrostatic test.

B. Reactor vessel nozzle-to-safe end welds are ultrasonically examined after the vessel hydrostatic test.

5.3.1.3.2 Liquid Penetrant Examination

The liquid penetrant examination requirements listed below are in addition to those specified by Section III of the ASME Boiler and Pressure Vessel Code.

A. Partial penetration welds for the control rod drive mechanism pressure boundary and the bottom mounted instrumentation tubes are examined by liquid penetrant techniques after the root pass.

B. Core support block attachment welds are examined by liquid penetrant techniques after the first layer of weld metal and after each 1/2 inch of deposit.

C. All clad surfaces and other vessel and head internal surfaces are examined by liquid penetrant techniques after the vessel hydrostatic test.

5.3.1.3.3 Magnetic Particle Examination

The magnetic particle examination requirements listed below are in addition to those specified by Section III of the ASME Boiler and Pressure Vessel Code. Magnetic particle examinations of materials and welds prior to final post weld heat treatment are performed by prod, coil or direct contact techniques. Magnetic particle examinations of materials and welds after final post weld heat treatment are performed by the yoke method.

A. All exterior surfaces of the reactor vessel and closure head are subject to magnetic particle examination after the vessel hydrostatic test.

B. All exterior surfaces of closure studs and nuts are subject to magnetic particle examination after final machining or rolling. Continuous circular and longitudinal magnetization is used.

C. The inside diameter surfaces of all carbon and low-alloy steel products which have their properties enhanced by accelerated cooling are subject to magnetic particle examination after forming and machining and prior to cladding.

D. Welds attaching the closure head lifting lugs and refueling seal ledge to the reactor vessel are subject to magnetic particle examination after the first layer and after each 1/2 inch of deposit.

E. All pressure boundary welds are subject to magnetic particle examination after back-chip or back-grinding operations.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Welding of ferrite steels and austenitic stainless steels is discussed in Subsection 5.2.3. Subsection 5.2.3 includes discussions which indicate the degree of conformance with Regulatory Guide 1.44. Section 1.8 discusses the degree of conformance with Regulatory Guides 1.43, 1.50, 1.71, and 1.99.

5.3.1.5 Fracture Toughness

Assurance of adequate fracture toughness of ferritic materials in the reactor vessel (ASME Code, Section III, Class 1 component) is provided by compliance with the requirements for fracture toughness testing included in NB-2300 to Section III of the ASME Code and Appendix G of 10 CFR 50.

The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline (including welds) are greater than 75 ft-lb, as required by Appendix G of 10 CFR 50.

5.3.1.6 Material Surveillance

In the surveillance program, the evaluation of radiation damage is based on preirradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch, tensile, and 1/2-thickness (T) compact tension (CT) fracture mechanics test specimens. The program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach. The program conforms to American Society of Testing Materials (ASTM) E-185-42, Conducting Surveillance Tests for Light-Water-Cooled Nuclear Reactor Vessels, and 10 CFR 50, Appendix H.

The reactor vessel surveillance program uses six specimen capsules. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The six capsules contain reactor vessel steel specimens, oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting base material located in the core region of the reactor vessel and associated weld metal and weld heat-affected zone metal. The six capsules contain 54 tensile specimens, 360 Charpy V-notch specimens (which include weld metal and weld heat-affected zone material), and 72 CT specimens. Archive material sufficient for two additional capsules is retained.

Dosimeters, as described below, are placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys are included to monitor the maximum 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. Each of the six capsules contains the following specimens:

Material	Number of Charpys	Number of Tensiles	Number of CTs
Limiting base material[(a)]	15	3	4
Limiting base material[(b)]	15	3	4
Weld metal[(c)]	15	3	4
Heat-affected zone	15	-	-

The following dosimeters and thermal monitors are included in each of the six capsules:

A. Dosimeters

1. Iron
2. Copper
3. Nickel
4. Cobalt-aluminum (0.15-percent cobalt)
5. Cobalt-aluminum (cadmium shielded)
6. Uranium-238 (cadmium shielded)
7. Neptunium-237 (cadmium shielded)

B. Thermal Monitors

1. 97.5-percent lead, 2.5-percent silver, (579°F melting point)
2. 97.5-percent lead, 1.75-percent silver, 0.75-percent tin (590°F melting point)

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. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition

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a. Specimens oriented in the major rolling or working direction.

b. Specimens oriented normal to the major rolling or working direction.

c. Weld metal to be selected in accordance with ASTM E-185.

temperature shift measurements are representative of the vessel at a later time in life. Data from CT fracture toughness specimens are expected to provide additional information for use in determining allowable stresses for irradiated material.

Correlations between the calculations and measurements of the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Subsection 5.3.1.6.1. The anticipated degree to which the specimens perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure is made by the use of data on all capsules withdrawn. The schedule for removal of the capsules for postirradiation testing conforms with ASTM E-185-79 and Appendix H of 10 CFR 50.

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

In order to effect a correlation between fast neutron (E > 1.0 MeV) exposure and the radiation-induced property changes observed in the test specimens, a number of fast neutron flux monitors are included as an integral part of the reactor vessel surveillance program. In particular, the surveillance capsules contain detectors employing the following reactions:

Fe^{54} (n,p) Mn^{54}
Ni^{58} (n,p) Co^{58}
Cu^{63} (n,α) Co^{60}
Np^{237} (n,f) Cs^{137}
U^{238} (n,f) Cs^{137}

In addition, thermal neutron flux monitors, in the form of bare and cadmium-shielded Co-Al wire, are included within the capsules to enable an assessment of the effects of isotopic burnup on the response of the fast neutron detectors.

The use of activation detectors such as those listed above does not yield a direct measure of the energy dependent neutron flux level at the point of interest. Rather, the activation process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material. An accurate estimate of the average neutron flux level incident on the various detectors may be derived from the activation measurements only if the parameters of the irradiation are well known. In particular, the following variables are of interest:

- The operating history of the reactor.
- The energy response of the given detector.
- The neutron energy spectrum at the detector location.

The procedure for the derivation of the fast neutron flux from the results of 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 energy spectrum, is similar.

The Mn^{54} product of the Fe^{54} (n,p) Mn^{54} 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 interfering activities. This separation is simple and very effective, yielding sources of very pure Mn^{54} activity. In some samples, all of the interferences may be corrected for by the gamma spectrometric methods without any chemical separation.

The analysis of the sample requires that two procedures be completed. First, the Mn^{54} disintegration rate per unit mass of sample and then iron content of the sample must be measured as described above. Second, the neutron energy spectrum at the detector location must be calculated.

For this analysis, DOT two-dimensional multigroup discrete ordinates transport code[1] is employed to calculate the spectral data at the location of interest. Briefly, the DOT calculations utilize a 47 group energy scheme, an S_8 order of angular quadrature, and a P_3 expansion of the scattering matrix to compute neutron radiation levels within the geometry of interest. The reactor geometry employed here includes a description of the radial regions internal to the primary concrete (core barrel, pressure vessel, and water annuli) as well as the surveillance capsule and an appropriate reactor core fuel loading pattern and power distribution. Thus, distortions in the fission spectrum due to the attenuation of the reactor internals are accounted for in the analytical approach.

Having the measured activity, sample weight, and neutron energy spectrum at the location of interest, the calculation of the threshold flux is as follows.

The induced Mn^{54} activity in the iron flux monitors may be expressed as:

$$D = \frac{No}{A} f_i \int_E \sigma(E)\phi(E) \sum_{J=1}^{n} F_j(1-e^{-\lambda\tau_J})e^{-\lambda\tau_d}$$

where:

- D = induced Mn_{54} activity (dps/g_{Fe}).
- No = Avogadro's number (atoms/g-atom).
- A = atomic weight of iron (g/g-atom).
- f_i = weight fraction of Fe^{54} in the detector.
- $\sigma(E)$ = energy dependent activation cross section for the Fe^{54} (n,p) Mn^{54} reaction (barns).
- $\phi(E)$ = energy dependent neutron flux at the detector at full reactor power (n/cm^2-s).
- λ = decay constant of Mn^{54} (1/s).
- F_j = fraction of full reactor power during the Jth time interval, τ_J.
- τ_J = length of the Jth irradiation period (s).
- τ_d = decay time following the Jth irradiation period (s).

The parameters F_j, τ_J, and τ_d depend on the operating history of the reactor and the delay between capsule removal and sample counting.

The integral term in the above equation may be replaced by the following relation:

$$\int\sigma(E)\phi(E) = \bar{\sigma}\,\bar{\phi}_{E_{TH}} = \frac{\sum_{o}^{\infty} \sigma_S(E)\phi_S(E)}{\sum_{E_{TH}}^{\infty} \phi_S(E)}\,\bar{\phi}_{E_{TH}}$$

where:

$\bar{\sigma}$ = effective spectrum average reaction cross-section for neutrons above energy, E_{TH}.

$\bar{\phi}_{E_{TH}}$ = average neutron flux above energy, E_{TH}.

$\sigma_S(E)$ = multigroup Fe^{54} (n,p) Mn^{54} reaction cross-sections compatible with the DOT energy group structure.

$\phi_S(E)$ = multigroup energy spectra at the detector location obtained from the DOT analysis.

E_{TH} = threshold energy for damage correlation.

Thus:

$$D = \frac{No}{A} f_i\,\bar{\sigma}\,\bar{\phi}_{E_{TH}} \sum_{J=1}^{n} F_j\,(1-e^{-\lambda\tau_J})e^{-\lambda\tau_d}$$

or, solving for the threshold flux:

$$\bar{\phi}_{E_{TH}} = \frac{D}{\frac{No}{A} f_i\,\bar{\sigma} \sum_{J=1}^{n} F_j\,(1-e^{-\lambda\tau_J})e^{-\lambda\tau_d}}$$

The total fluence above energy E_{TH} is given by:

$$\Phi_{E_{TH}} = \bar{\Phi}_{E_{TH}} \sum_{J=1}^{n} F_J \tau_J$$

where:

$\sum_{J=1}^{n} F_J \tau_J$ = the total effective full power seconds of reactor operation up to the time of capsule removal.

Because of the relatively long half-life of Mn^{54}, the fluence may be accurately calculated in this manner for irradiation periods up to about 2 years. Beyond this time, the calculated average flux begins to be weighted toward the later stages of irradiation, and some inaccuracies may be introduced. At these longer irradiation times, therefore, more reliance must be placed on Np^{237} and U^{238} fission detectors with their 30-year half-life product (Cs^{137}).

No burnup correction was made in the preceding equations, since burnout of the Mn^{54} product is not significant until the thermal flux level is about 10^{14} n/cm^2-s.

The error involved in the measurement of the specific activity of the detector after irradiation is estimated to be ± 5 percent at the 1σ level.

5.3.1.6.2 Calculation of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

The energy and spatial distribution of neutron flux within the reactor geometry is obtained from the DOT two-dimensional S_N transport code.(1) The radial and azimuthal distributions are obtained from an R, θ computation, wherein the reactor core as well as the water and steel annuli surrounding the core are modeled explicitly. The axial variations are then obtained from an R,Z DOT calculation using the equivalent cylindrical core concept. The neutron flux at any point in the geometry is then given by:

$$\phi(E,R,\theta,Z) = \phi(E,R,\theta)\ F(Z)$$

where $\phi(E,R,\theta)$ is obtained directly from the R,θ calculation and F(Z) is a normalized function obtained from the R,Z analysis. The core power distributions used in both the R,θ and R,Z computations represent the expected average over the life of the station.

Having the calculated neutron flux distributions within the reactor geometry, the exposure of the capsule as well as the lead factor between the capsule and the vessel may be determined as follows:

The neutron flux at the surveillance capsule is given by:

$$\phi_c = \phi(E,R_c,\theta_c,Z_c)$$

and the flux at the location of peak exposure on the pressure vessel inner diameter is:

$$\phi_{v-max} = \phi(E,R_{v\theta v-max},Z_{v-max})$$

The lead factor then becomes:

$$LF = \frac{\phi_c}{\phi_{v-max}}$$

Similar expressions may be developed for points within the pressure vessel wall and thus, together with the surveillance program dosimetry, serve to correlate the radiation-induced damage to test specimens with that of the reactor vessel.

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure studs, nuts, and washers are designed and fabricated in accordance with the requirements of the ASME Code, Section III. The closure studs will be fabricated of SA-540, Class 3, Grade B24. The closure stud material will meet the fracture toughness requirements of the ASME Code, Section III, and 10 CFR 50, Appendix G. Compliance with Regulatory Guide 1.65, Materials and Inspections for Reactor Vessel Closure Studs, is discussed in Section 1.8. Nondestructive examinations will be performed in accordance with the ASME Code, Section III.

Refueling procedures require that the studs, nuts, and washers be removed from the reactor closure and be placed in storage racks during preparation for refueling. The storage racks are then removed from the refueling cavity and stored at convenient locations on the containment operating deck prior to removal of the reactor closure head and refueling cavity flooding. Therefore, the reactor closure studs are never exposed to the borated refueling cavity water. Additional protection against the possibility of incurring corrosion effects is ensured by the use of a manganese base phosphate surfacing treatment.

The stud holes in the reactor flange are sealed with special plugs before removing the reactor closure, thus preventing leakage of the borated refueling water into the stud holes.

5.3.2 Pressure-Temperature Limits

5.3.2.1 Limit Curves

Startup and shutdown operating limitations are based on the properties of the reactor pressure vessel beltline materials. Actual material property test data are used. The methods outlined in Appendix G of Section III of the American Society of Mechanical Engineers (ASME) Code are employed for the shell regions in the analysis of protection against nonductile failure. The initial operating curves are calculated, assuming a period of reactor operation such that the beltline material will be limiting. The heatup and cooldown curves will be provided in the Technical Specifications. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil ductility temperature, which includes a reference nil ductility temperature shift (ΔRT_{NDT}).

Predicted ΔRT_{NDT} values are derived using two curves: the effect of fluence and copper content on the shift of RT_{NDT} for the reactor vessel steels exposed to 550°F temperature curve and the maximum fluence at 1/4 thickness (T) and 3/4 T location (tips of the code reference flaw when flaw is assumed at inside diameter and outside diameter locations, respectively) curve. These curves will be provided in the Technical Specifications. For a selected time of operation, this shift is assigned a sufficient magnitude so that no unirradiated ferritic materials in other components of the reactor coolant system (RCS) will be limiting in the analysis.

The operating curves including pressure-temperature limitations are calculated in accordance with 10 CFR 50, Appendix G and ASME Code, Section III, Appendix G requirements.

The results of the material surveillance program described in Subsection 5.3.1.6 will be used to verify that the ΔRT_{NDT} predicted from the effects of the fluence and copper content curve is appropriate and to make any changes necessary to correct the fluence and copper curves, if ΔRT_{NDT} determined from the surveillance program is greater than the predicted ΔRT_{NDT}. Temperature limits for inservice leak and hydrotests will be calculated in accordance with ASME Code, Section III, Appendix G. Conformance with Regulatory Guide 1.99 is discussed in Section 1.8.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Design

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, bolted, flanged-and-gasketed hemispherical upper head. The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leakoff connections, one between the inner and outer ring and one outside the outer O-rings. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains CRDM and DRDM penetrations. These head penetrations are tubular members, attached by partial penetration welds to the underside of the closure head. Four inlet and four outlet nozzles are spaced evenly around the vessel. Outlet nozzles are arranged in adjacent pairs on opposite sides of the vessel to facilitate optimum layout of the reactor coolant system equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop. The reactor vessel is also equipped with four nozzles for introduction of safety injection flow (direct vessel injection). These nozzles are evenly spaced and each is connected to one of the four integrated safeguards system subsystems.

Eight vessel upper support pads are located symmetrically around the vessel. These pads are attached to the vessel O.D. between and below the vessel inlet and outlet nozzles. These supports withstand vertical and horizontal loads on the vessel.

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 either an Inconel or an Inconel-stainless steel composite tube. Each tube is attached to the inside of the bottom head by a partial penetration weld.

Four lower external support pads are attached to the bottom head. These pads are symmetrically located and resist any horizontal vessel motion or loads.

Internal surfaces of the vessel which are in contact with primary coolant are weld overlay with 0.125-in. minimum of stainless steel or Inconel.

The reactor vessel is designed and fabricated in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, Section III. Principal design parameters of the reactor vessel are given in Table 5.3-2. The reactor vessel is shown in Figure 5.3-1.

There are no special design features which would prohibit the in situ annealing of the vessel. If the unlikely need for an annealing operation was required to restore the properties of the vessel material opposite the reactor core because of neutron irradiation damage, a metal temperature greater than 650°F for a maximum period of 168 hours would be applied. Various modes of heating may be used, depending on the temperature required.

The reactor vessel materials surveillance program is adequate to accommodate the annealing of the reactor vessel. Sufficient specimens are available to evaluate the effects of the annealing treatment.

Thermal loadings are introduced by normal power changes, reactor trips, startup and shutdown operations and several other design transient cases. These design transients are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life. Vessel analysis results in a usage factor that is less than 1.

The design specifications require analysis to prove that the vessel is in compliance with the fatigue and stress limits of the ASME Code, Section III. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. These transients are reflected in the vessel design specification.

5.3.3.2 Materials of Construction

The materials used in the fabrication of the reactor vessel are discussed in Subsection 5.3.1.

5.3.3.3 Fabrication Methods

The fabrication methods used in the construction of the reactor vessel are discussed in Subsection 5.3.1.2.

5.3.3.4 Inspection Requirements

The nondestructive examinations performed on the reactor vessel during fabrication are described in Subsection 5.3.1.3.

5.3.3.5 Shipment and Installation

The reactor vessel is shipped in a horizontal position on a shipping sled with a vessel-lifting truss assembly. All vessel openings are sealed to prevent the entrance of moisture, and an adequate quantity of desiccant bags is placed inside the vessel. These are usually placed in a wire mesh basket attached to the vessel cover. All carbon steel surfaces except for the vessel support surfaces are painted with a heat-resistant paint before shipment.

The closure head is also shipped with a shipping cover and skid. An enclosure attached to the integrated head package support ring protects the CRDM and DRDM housings. All head openings are sealed to prevent the entrance of moisture, and an adequate quantity of desiccant bags is placed inside the head. These are placed in a wire mesh basket attached to the head cover. All carbon steel surfaces are painted with heat-resistant paint before shipment. A lifting frame is provided for handling the vessel head.

5.3.3.6 Operating Conditions

Operating limitations for the reactor vessel are presented in Subsection 5.3.2 and will be provided in the Technical Specifications.

In addition to the analysis of primary components discussed in Subsection 3.9.1.4 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design", the reactor vessel is further qualified to ensure against unstable crack growth under faulted conditions. Actuation of the emergency core cooling system (ECCS) following a loss-of-coolant accident produces relatively high thermal stresses in regions of the reactor vessel which come into contact with ECCS water. Primary consideration is given to these areas to ensure the integrity of the reactor vessel under this severe postulated transient.

For the beltline region, significant developments have recently occurred in order to address pressurized thermal shock (PTS) events. On the basis of recent deterministic and probabilistic studies, taking U.S. PWR operating experience into account, the NRC Staff concluded that conservatively calculated values of RT_{NDT} less than 270°F for plate material and axial welds, and less than 300°F for circumferential welds, present an acceptably low risk of vessel failure from PTS events. These values were chosen as the screening criterion in the proposed new §50.61 of 10 CFR Part 50. The conservative method chosen by the Staff for the calculation of the RT_{NDT} for the purpose of comparison with the screening criterion is presented in paragraph (b) (2) of the proposed §50.61.

Using the maximum limits of the reactor vessel beltline materials specified in Subsection 5.3.1.1 in conjunction with the fluence levels associated with the anticipated service life of the facility, the screening criteria will not be exceeded using the RT_{NDT} method of calculation prescribed by the NRC Staff for the vessel design lifetime.

The principles and procedures of linear elastic fracture mechanics (LEFM) are used to evaluate thermal effects in the regions of interest. The LEFM approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in LEFM is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

The elastic stress field at the crack tip in any cracked body can be described by a single parameter designated as the stress intensity factor, K. The magnitude of the stress intensity factor K is a function of the geometry of the body containing the crack, the size and location of the crack, and the magnitude and distribution of the stress.

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses perpendicular to the major plane of the crack), the stress intensity factor is designated as K_I and the critical stress intensity factor is designated for K_{IC}. Commonly called the fracture toughness, K_{IC} is an inherent material property which is a function of temperature. Any combination of applied load, structural configuration, crack geometry, and size which yields a stress intensity factor greater than K_{IC} for the material will result in crack instability.

The LEFM analysis methods in ASME XI Appendix A and ASME III Appendix G are used to perform the fracture evaluation of postulated flaws to establish that the vessel integrity is maintained. This LEFM analysis is considered accurate in the elastic range and conservative in the elastic-plastic range, and has been utilized in the evaluation of the vessel inlet nozzle and beltline region for faulted conditions.

In addition, it has been well established that the crack propagation of existing flaws in a structure subjected to cyclic loading can be defined in terms of fracture mechanics parameters. Thus, the principles of LEFM are also applicable to fatigue growth of a postulated flaw at the vessel inlet nozzle and beltline region.

Additional details on this method of analysis of reactor vessels under severe thermal transients are given in reference 2.

5.3.3.7 Inservice Surveillance

Several requirements, in excess of those specified by Section III of the ASME Boiler and Pressure Vessel Code, are imposed during vessel manufacture to facilitate meaningful nondestructive examinations during the periodic inservice examination program as specified by the applicable edition of Section XI of the ASME Boiler and Pressure Vessel Code. These include:

A. Preparation of weld deposited stainless steel cladding on the reactor vessel inside surface to allow for meaningful ultrasonic examination of regions requiring volumetric examinations per Section XI.

B. Ultrasonic examination of all weld deposited stainless steel cladding on the vessel inside surface to demonstrate adequate clad bonding.

C. Ultrasonic examination of all ferritic full penetration pressure boundary welds in the reactor vessel and closure head upon completion of welding and intermediate heat treatment.

D. Ultrasonic examination of all ferritic full penetration pressure boundary welds in the reactor vessel and closure head, as well as the primary nozzle-to-safe end welds, after the vessel hydrostatic test.

In addition, the reactor vessel, closure head, and appurtenances are designed and fabricated to allow access for periodic visual, surface, and volumetric examinations in accordance with Section XI of the ASME Boiler and Pressure Vessel Code when installed at the plant site.

All reactor internals are completely removable, providing access to the entire inside surface. In this configuration, the vessel shell is an uncluttered cylinder in which test equipment can be positioned and manipulated to all areas which require periodic examination with obstruction. With the internals in place the top of the vessel flange and the outlet nozzle bores are accessible.

The reactor vessel closure head is stored dry on the operating deck during refueling. In this location, access to both the inside and outside surfaces is provided. All reactor vessel bolting; studs, nuts, and washers can also be removed to dry storage during refueling where they are completely accessible for examination. The bottom head of the reactor vessel and all primary nozzle-to-safe end welds are accessible from the outside surface as well as the inside.

Specific information concerning the reactor vessel inservice examination program will be provided in the Technical Specifications.

5.3.4 REFERENCES

1. Soltesz, R. G., et al., "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Volume 5 - Two-Dimensional Discrete Ordinates Techniques," WANL-PR-(LL)-034, August 1970.

2. Buchalet, C., Bamford, W. H., "Method for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients," WCAP-8510, July 1976.

TABLE 5.3-1 (SHEET 1 OF 2)

REACTOR VESSEL QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]
Forgings				
Flanges		Yes		Yes
Studs and nuts		Yes		Yes
CRDM head adapter		Yes	Yes	
DRDM head adapter		Yes	Yes	
Instrumentation tube		Yes	Yes	
Nozzles		Yes		Yes
Nozzle safe ends		Yes	Yes	
Shells		Yes		Yes
Plates		Yes		Yes
Weldments				
Shell	Yes	Yes		Yes
CRDM head adapter to closure head connection			Yes	
Instrumentation tube to bottom head connection			Yes	
Nozzle	Yes	Yes		Yes
Cladding		Yes	Yes	
Nozzle to safe ends	Yes	Yes	Yes	
All full-penetration ferritic pressure boundary welds accessible after hydrotest		Yes		Yes

TABLE 5.3-1 (SHEET 2 OF 2)

REACTOR VESSEL QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]
Full-penetration nonferritic pressure boundary welds accessible after hydrotest				
a. Nozzle to safe ends		Yes	Yes	
Seal ledge				Yes
Head lift lugs				Yes
Core pad welds			Yes	

a. RT - Radiographic
UT - Ultrasonic
PT - Dye penetrant
MT - Magnetic particle

NOTE:

Base metal weld repairs as a result of UT, MT, RT, and/or PT indications are cleared by the same NDE technique/procedure by which the indications were found. The repairs will meet all Section III requirements.

In addition, UT examination in accordance with the inprocess/posthydro UT requirements is performed on base metal repairs in the core region and base metal repairs in the inservice inspection zone (1/2 T).

TABLE 5.3-2

REACTOR VESSEL DESIGN PARAMETERS

Design/maximum operating pressure (psig)	(a
Design temperature (°F)	
Overall height of vessel and closure head, bottom head outside diameter to top head outside diameter (ft-in.)	
Thickness of reactor pressure vessel head insulation, minimum (in.)	
Number of reactor closure head studs	
Diameter of reactor vessel closure studs (in.)	
Outside diameter of flange (in.)	
Inside diameter of flange (in.)	
Outside diameter at shell (in.)	
Inside diameter at shell (in.)	
Inlet nozzle inside diameter (in.)	
Outlet nozzle inside diameter (in.)	
Clad thickness, minimum (in.)	
Lower head thickness, minimum (in.)	
Vessel beltline thickness, minimum (in.)	
Closure head thickness (in.)	
Nominal water volume (ft^3)	
Safety injection nozzle inside diameter (in.)	
Number of CRDM/DRDM housing	
Number of BMI penetrations	

(a,c)

FIGURE 5.3-1 REACTOR VESSEL
(SHEET 1 OF 2)

(a,c)

FIGURE 5.3-1 REACTOR VESSEL
(SHEET 2 OF 2)

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Coolant Pump Assembly

5.4.1.1 Design Bases

The reactor coolant pump assembly ensures an adequate core cooling flowrate for sufficient heat transfer to maintain a departure from nucleate boiling ratio greater than the limiting value within the parameters of operation. The required net positive suction head (NPSH) is by conservative pump design always less than that available by system design and operation.

Sufficient pump assembly rotating inertia is provided by the motor flywheel, motor rotor, and pump rotating parts which provide adequate flow during coast-down conditions. This forced flow following an assumed loss of offsite electrical power and the subsequent natural circulation effect provides the core with adequate cooling.

The reactor coolant pump is shown in Figure 5.4-1. The reactor coolant pump design parameters are given in Table 5.4-1.

Component cooling water flow to the reactor coolant pump assembly can be provided during all modes of plant operation except postulated loss of all ac power, as the CCW pumps can be powered by the emergency diesel generators. This is described in RESAR-SP/90 PDA Module 13, "Auxiliary Systems" and meets the intent of TMI Action Item II.K.3.25.

5.4.1.2 Pump Assembly Description

5.4.1.2.1 Design Description

The reactor coolant pump is a vertical, single-stage, controlled leakage, centrifugal pump designed to pump large volumes of reactor coolant at high temperatures and pressures.

The pump assembly consists of three major sections. They are the hydraulics, the seals, and the motor.

A. The hydraulic section consists of the casing, impeller, turning vane diffuser, and diffuser adapter.

B. The seal section consists of three seals arranged in series. The first is a controlled leakage film-riding seal; the second and third are rubbing face seals. These seals are contained within seal housings, and the second and third seals are contained within the seal cartridge. The seal system provides a pressure breakdown from the reactor coolant system (RCS) pressure to ambient conditions.

C. The motor section consists of a drip-proof squirrel cage induction motor with a vertical solid shaft, an oil-lubricated, double-acting Kingsbury type thrust bearing, upper and lower oil-lubricated radial guide bearings, and a flywheel.

Additional components of the pump are the shaft, pump radial bearing, thermal barrier heat exchanger assembly, coupling, spool piece, and motor stand.

5.4.1.2.2 Description of Operation

The reactor coolant enters the suction nozzle, is pumped by the impeller through the diffuser, and exits through the discharge nozzle. The diffuser adapter limits the leakage of reactor coolant back to the suction.

Seal injection flow, under slightly higher pressure than the reactor coolant, enters the pump through a connection on the thermal barrier flange and is directed into the plenum between the thermal barrier housing and the shaft. The flow splits, with the major portion flowing down the shaft through the radial bearing and into the RCS. The remaining seal injection flow passes up the shaft through the seals.

The pump's thermal barrier heat exchanger is cooled by the CCWS. During normal operation, the thermal barrier limits the heat transfer from hot reactor coolant to the radial bearing and to the seals. In addition, if a loss of seal injection flow should occur, the thermal barrier heat exchanger cools the reactor coolant, flowing upward from the pump casing (opposite to the normal flow direction), to an acceptable level before it enters the bearing and seal area.

The reactor coolant pump motor oil-lubricated bearings are of conventional design. The radial bearings are the segmented pad type, and the thrust bearing is a double-acting Kingsbury type. Component cooling water is supplied to the external upper bearing oil cooler and to the integral lower bearing oil cooler.

The oil spillage protection system is attached to the reactor coolant pump motor and is provided to contain and channel oil to a common collection point.

The motor is a drip-proof squirrel cage induction motor with Class F thermalastic epoxy insulation, fitted with external water/air coolers. The rotor and stator are of standard construction. Six resistance temperature detectors are embedded in the stator windings to sense stator temperature. A flywheel and an antireverse rotation device are located at the top of the motor.

The internal parts of the motor are cooled by air. Integral vanes on each end of the rotor draw air in through cooling slots in the motor frame. This air passes through the motor with particular emphasis on the stator end turns. It is then routed to the external water/air coolers, which are supplied with CCW. Each motor has two such coolers, mounted diametrically opposed to each other. Coolers are sized to maintain optimum motor-operating temperature. The air is finally exhausted to the containment environment.

Each of the reactor coolant pump assemblies is equipped for continuous monitoring of reactor coolant pump shaft and frame vibration levels. Shaft vibration is measured by two relative shaft probes mounted on top of the pump seal housing; the probes are located 90° apart in the same horizontal plane and mounted near the pump shaft. Frame vibration is measured by two velocity seismoprobes located 90° apart in the same horizontal plane and mounted at the top of the motor support stand. Proximeters and converters linearize the probe output, which is displayed on monitor meters in the control room. The monitor meters automatically indicate the highest output from the relative probes and seismoprobes; manual selection allows monitoring of individual probes. Indicator lights display caution and danger limits of vibration.

The spool piece, which is a removable shaft segment, is located between the motor coupling flange and the pump coupling flange; the spool piece allows removal of the pump seals with the motor in place. The pump internals, motor, and motor stand can be removed from the casing without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

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

5.4.1.2.3 Loss of Seal Injection

Should a loss of seal injection to the reactor coolant pumps occur, the pump radial bearing and seals are lubricated by reactor coolant flowing upward from the pump casing toward the seal area. Under these conditions, the CCW continues to provide flow to the thermal barrier heat exchanger; this heat exchanger cools the reactor coolant flow before it enters the pump radial bearing and the shaft seal area. The loss of seal injection flow may result in a temperature increase in the pump bearing area, a temperature increase in the seal area, and a resultant increase in the No. 1 seal leak rate; however, pump operation can be continued (for up to 24 hours), provided these parameters remain within the allowable limits.

5.4.1.2.4 Loss of Component Cooling Water

Should a loss of CCW to the reactor coolant pumps occur, the chemical and volume control system continues to provide seal injection flow to the reactor coolant pumps; the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling. However, the loss of CCW to the motor bearing oil coolers will result in an increase in oil temperature and a corresponding rise in motor bearing metal temperature. Westinghouse has demonstrated by testing that the reactor coolant pumps will incur no damage as a result of an CCW flow interruption of 20 minutes.

Flow transmitters will be provided to monitor CCW flow for the upper and lower reactor coolant pump bearing oil coolers, as well as to monitor CCW flow for the reactor coolant pump thermal barriers. These transmitters will provide flow indication and actuate low-flow alarms in the control room.

Operating procedures are provided for a loss of CCW and/or seal injection to the reactor coolant pumps. Included in these operating procedures is the provision to trip the reactor if CCW flow, as indicated by the instrumentation discussed above, is lost to the reactor coolant pump motors and cannot be restored within 20 minutes. The reactor coolant pumps will also be manually tripped following the reactor trip.

5.4.1.2.5 Backup Seal Injection Capability

During normal plant operation, the reactor coolant pump (RCP) seals are cooled and protected by seal injection from the chemical and volume control system (see RESAR-SP/90 PDA Module 13, "Auxiliary Systems"). Should normal seal injection flow be interrupted, the seals are cooled by reactor coolant, flowing upward from the pump bowl through the thermal barrier. The thermal barrier heat exchanger is cooled by the component cooling water system. These systems will also protect the seals following most accidents.

Following a loss of all ac power, however, neither normal seal injection nor component cooling water flow will be available for seal cooling. Although the reactor coolant pumps will be deenergized, coast down and stop, the RCS will remain hot and pressurized. It is believed that the RCP seals will not fail during such an event, however, greater assurance that they will be protected is provided by the backup seal injection capability of the CVCS. This capability is provided by a small positive displacement pump which takes suction from the spent fuel pit. The pump discharges through a separate backup seal injection filter to a line penetrating containment. This line branches inside containment into four individual lines which connect to the normal seal injection lines close to the reactor coolant pumps. The backup seal injection pump is driven by a dc motor powered from batteries. Further details are provided in the CVCS portion of RESAR-SP/90 PDA Module 13, "Auxiliary Systems".

5.4.1.3 Design Evaluation

5.4.1.3.1 Pump Performance

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

The estimated performance characteristic is shown in Figure 5.4-2. The knee, at about 60-percent design flow, introduces no operational restrictions, since the pumps only operate at a speed which corresponds to full flow.

The reactor coolant pump motor is tested, without mechanical damage, at overspeeds up to and including 125 percent of normal speed. The integrity of the flywheel during a loss-of-coolant accident (LOCA) is demonstrated in Reference 1, which is undergoing generic review by the NRC.

The reactor trip features ensure that pump operation is within the assumptions used for loss-of-coolant flow analyses, which confirm 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 stationary member of the No. 1 seal (seal ring) is supported such as to allow 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.

Testing of pumps with the No. 1 seal entirely bypassed (full system pressure on the No. 2 seal) shows that relatively small leakage rates would be maintained for a period of time which is sufficient to secure the pump; even if the No. 1 seal fails entirely during normal operation, the No. 2 seal would maintain these small leakage rates if the proper action is taken by the operator. The plant operator is warned of possible No. 1 seal damage by an increase in No. 1 seal leakoff rate. The operator should then close the No. 1 seal leakoff line and secure the pump, as specified in the instruction manual.

Gross leakage from the pump does not occur if the proper operator action is taken subsequent to warning of excessive seal leakage conditions.

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 CCW for seal and bearing cooling. The emergency diesel generators are started automatically upon loss of offsite electrical power, and CCW flow and seal injection flow are automatically restored.

5.4.1.3.2 Coastdown Capability

It is important to reactor protection that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow following loss of offsite electrical power, each reactor coolant pump is provided with a flywheel. The rotating inertia of the pump, motor, and flywheel acts during the coastdown period to continue the reactor coolant flow. The pump/motor assembly is designed to withstand the effects of the safe shutdown earthquake. Thus, the coastdown capability of the pumps is maintained even under the most adverse case of loss of offsite electrical power coincident with the safe shutdown earthquake. Core flow transients and figures are provided in Subsections 15.3.1 and 15.4.4 of this module.

5.4.1.3.3 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. The surface bearing stresses are held at very low values and even under the most severe seismic transients do not begin to approach loads that 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 lube oil sumps signal alarms in the control room and require shutting down of the pump. In addition, each motor bearing contains embedded temperature detectors, so that initiation of failure, manifested as a high bearing temperature, is indicated and alarmed separately in the control room. This also requires pump shutdown. If these indications are ignored and the bearing proceeds to failure, the low melting point of Babbitt metal on the pad surfaces ensures that sudden seizure of the shaft will not occur. In this event, the motor continues to operate, since it has sufficient reserve capacity to drive the pump under such conditions. However, the high torque required to drive the pump will require high current, which leads to the motor being shutdown by the electrical protection systems.

5.4.1.3.4 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, since it is still supported on a shaft with two bearings. Flow transients are provided in the figures in Subsection 15.3.3 of this module for the assumed locked rotor.

In that the pump is designed to preclude a locked rotor accident during a safe shutdown earthquake, there are no credible sources of shaft seizure other than impeller rubs. A sudden seizure of the pump bearing is precluded by graphite in the bearing. Any seizure in the seals results in shearing of the antirotation pin in the seal ring. The motor has adequate power to continue pump operation even after occurrences of this type. Indications of pump malfunction in these conditions are initially given by high-temperature signals from the bearing temperature detector and by excessive 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.

5.4.1.3.5 Critical Speed

The reactor coolant pump shaft is designed so that its operating speed is below its first critical speed. This shaft design, even under the most severe postulated transient, gives low values of actual stress.

5.4.1.3.6 Missile Generation

Precautionary measures taken to preclude missile formation from reactor coolant pump components ensure that the pumps will not produce missiles under anticipated accident conditions.

Appropriate components of the reactor coolant pump have been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator frame. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing. Further discussion and analysis of missile generation are contained in Reference 1.

5.4.1.3.7 Pump Cavitation

The minimum NPSH required by the reactor coolant pump at best estimate flow is approximately 250 ft (approximately 111 psi). In order for the controlled leakage seal to operate correctly, a minimum differential pressure of approximately 200 psi across the No. 1 seal is required. This corresponds to a primary loop pressure which exceeds the minimum NPSH required, and no further limitation on pump operation occurs.

5.4.1.3.8 Pump Overspeed Considerations

For turbine trips actuated by either the reactor trip system or the turbine protection system, the generator and reactor coolant pumps remain connected to the external network for 30 seconds. The generator "motors" at normal speed under these conditions and no RCP overspeed occurs during this period. After 30 seconds the RCP motors are transferred to offsite power.

An external electrical fault accompanied by immediate generator trip will result in an overspeed condition. However, the turbine DEH control system operates the turbine governor valves (at the high pressure inlet) and the intercept valves (between the high and low pressure stages) to limit the overspeed to less than 110 percent of nominal speed. If the overspeed exceeds approximately 110 percent, the turbine protection system employs a mechanical overspeed trip which completely closes these valves and the turbine and reheat stop valves as well to completely isolate steam flow to the turbine. Subsequent overspeed will not exceed 120 percent. In this case the generator trip deenergizes the pump buses and the RCP motors are transferred to offsite power within 6 to 10 cycles (fast bus transfer).

Further discussion of pump overspeed considerations is contained in Reference 1.

5.4.1.3.9 Antireverse Rotation Device

Each of the reactor coolant pump motors is provided with an antireverse rotation device. This antireverse mechanism consists of pawls mounted on the underside of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and two shock absorbers.

At an approximate forward speed of 70 rpm, the pawls drop and bounce across the ratchet plate; as the motor continues to slow, the pawls drag across the ratchet plate. After the motor has come to a stop, the dropped pawls engage the ratchet plate, and if the motor tends to rotate in the opposite direction, the ratchet plate also rotates until it is stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. When the motor is started, the ratchet plate is returned to its original position by the spring return.

As the motor begins to rotate, the pawls drag over the ratchet plate. When the motor reaches sufficient speed (approximately 70 rpm), the pawls are bounced into an elevated position as the centrifugal force exceeds the gravity force acting on the pawls. Above this speed there is no contact between the pawls and ratchet plate.

Considerable plant experience with the design of the antireverse rotation device has shown high reliability of operation.

5.4.1.3.10 Shaft Seal Leakage

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. Seal injection flow is directed to each reactor coolant pump via a common seal water injection filter. It enters each pump through a connection on the thermal barrier flange. Here the flow splits; the major portion flows up the shaft and through the No. 1 seal. Most of the seal flow then leaves the pump through the No. 1 seal leakoff line. Due to the backpressure at this point afforded by the leakoff piping and volume control tank pressure, a small fraction of the flow continues along the shaft to the No. 2 seal. Leakage through the No. 2 seal is piped to the reactor coolant drain tank. Any gas or water vapor present at the No. 2 seal outlet is prevented from leaking to the atmosphere by the No. 3 seal. This seal is of the double-dam design, with both sides of the dam continuously wetted and flushed by a small continuous flow from the No. 3 seal standpipe (head tank). The flushing flow across one side of the dam entraps the water vapor and non-condensables and exits the pump with the No. 2 seal leakoff flow. The flow across the other side of the dam is clean flushing water and is discharged to the containment sump.

5.4.1.3.11 Seal Discharge Piping

The cooling flow exiting the No. 1 seal is at the backpressure developed by the leakoff piping and the volume control tank. The flow from each No. 1 seal is piped to a common manifold, then through the seal water return filter to the volume control tank. The No. 2 and 3 seal leakoff lines route No. 2 and 3 seal leakage to the reactor coolant drain tank and the containment sump, respectively.

5.4.1.4 Tests and Inspections

The reactor coolant pumps can be inspected in accordance with the ASME Code, Section XI, "Rules For Inservice Inspection of Nuclear Power Plant Components."

The design enables disassembly and removal of the pump internals for visual access to the inside of the pump casing. In addition, the support feet are cast integrally with the casing to eliminate a weld region.

The reactor coolant pump quality assurance program is given in Table 5.4-2.

5.4.1.5 Pump Flywheel

The integrity of the reactor coolant pump flywheel is assured by the following design and quality assurance procedures.

5.4.1.5.1 Design Basis

The calculated stresses at operating speed are based on stresses due to centrifugal forces. The stress resulting from the interference fit of the flywheel on the shaft is less than 2000 psi at zero speed, but this stress becomes zero at approximately 600 rpm because of radial expansion of the hub. The reactor coolant pumps run at approximately 1190 rpm and may operate briefly at overspeeds up to 110 percent (1309 rpm) during loss of offsite electrical power (generator trip - see Subsection 5.4.1.3.8). For conservatism, however, 125 percent of operating speed was selected as the design speed for the reactor coolant pumps. The flywheels are given a preoperational test at 125 percent of the synchronous speed of the motor.

5.4.1.5.2 Fabrication and Inspection

The flywheel consists of two thick plates bolted together. The flywheel material is produced by a process that minimizes flaws in the material and improves its fracture toughness properties i.e., in an electric furnace with vacuum degassing. Each plate is fabricated from SA-533, Grade B, Class 1 steel. Supplier certification reports are available for all plates and demonstrate the acceptability of the flywheel material on the basis of the requirements of NRC Regulatory Guide 1.14.

Flywheel blanks are flame cut from the plate material with at least 1/2 in. of stock left on the outer surface and bore surface for machining to final dimensions. The finished machined bores and keyways are subjected to magnetic particle or liquid penetrant examinations in accordance with the requirements of Section III of the ASME Code. The finished flywheels, as well as the flywheel material (rolled plate), are subjected to 100-percent volumetric ultrasonic inspection using procedures and acceptance standards specified in Section III of the ASME Code.

The reactor coolant pump motors are designed such that, by removing the access cover, the flywheel is available for inservice inspection program in accordance with the recommendations of Regulatory Guide 1.14, referencing Section XI of the ASME Code.

5.4.1.5.3 Material Acceptance Criteria

The reactor coolant pump motor flywheel conforms to the following material acceptance criteria:

A. The nil ductility transition temperature (NDTT) of the flywheel material is obtained by two drop weight tests which exhibit no-break performance at 20°F in accordance with ASTM E-208. The above drop weight tests demonstrate that the NDTT of the flywheel material is no higher than 10°F.

B. A minimum of three Charpy V-notch (C_V) impact specimens from each plate shall be tested at ambient (70°F) temperature in accordance with the ASME SA-370 specification. The Charpy V-notch (C_V) energy in both the parallel and normal orientation with respect to the final rolling direction of the flywheel plate material is at least 50 ft-lb with a 35-mil lateral expansion at 70°F, and, therefore, the flywheel material has a reference nil ductility temperature (RT_{NDT}) of 10°F. An evaluation of flywheel overspeed has been performed which concludes that flywheel integrity will be maintained.[1]

Thus, it is concluded that flywheel plate materials are suitable for use and can meet Regulatory Guide 1.14 acceptance criteria on the basis of suppliers certification data. The degree of compliance with Regulatory Guide 1.14 is further discussed in Section 1.8.

5.4.2 Steam Generators

5.4.2.1 Design Bases

Steam generator design data are given in Table 5.4-3. Code classifications of the steam generator components are given in Section 3.2. Although the safety classification for the secondary side is required to be Safety Class 2, the current philosophy is to design all pressure retaining parts of the steam generator, including both the primary and secondary side pressure boundaries, to satisfy the criteria specified in Section III of the ASME Code for Class 1 components. The design stress limits, transient conditions, and combined loading conditions applicable to the steam generator are discussed in Subsection 3.9.1 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design". Estimates of radioactivity levels anticipated in the secondary side of the steam generators during normal operation and the bases for the estimates are given in Chapter 11.0. The accident analysis of a steam generator tube rupture is discussed in Subsection 15.6.3 of RESAR-SP/90 PDA Module 6, "Secondary Side Safeguards System".

A design objective of the internal moisture separation equipment is that moisture carryover should not exceed 0.25 percent by weight under the following conditions:

A. Steady state operation up to 100 percent of full load steam flow with water at the normal operating level.

B. Loading or unloading at a rate of 5 percent of full power steam flow per minute in the range from 15 to 100 percent of full load steam flow.

C. Step load changes of 10 percent of full power in the range from 15 to 100 percent full load steam flow.

D. Steam pressure as much as 300 psi below its nominal full load value, coincident with the corresponding increase in volumetric steam flow rate required to maintain full load heat input to the turbine.

The primary side (reactor coolant) water chemistry is selected to minimize corrosion of RCS surfaces. Compatibility of the steam generator tubing with both the primary and secondary coolants is discussed further in Subsection 5.4.2.4.3.

The steam generator is designed with the objective to minimize tube degradation due to mechanical or flow-induced vibration. This subject is discussed in Subsection 5.4.2.3.3. The tubes and tube sheet are sized to withstand the maximum accident loading conditions as they are defined in Subsection 3.9.1 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design". Further consideration is given in Subsection 5.4.2.3.4 to the effect of tube wall thinning on accident condition stresses.

5.4.2.2 Design Description

The WAPWR steam generator is a vertical shell and U-tube evaporator with integral moisture separating equipment. Figure 5.4-3 illustrates its design features, several of which are described in the following paragraphs.

On the primary side, the reactor coolant flows through the inverted U-tubes, entering and leaving through nozzles located in the primary channel head. A

[(a,c,e,f)

] The head is divided into inlet and outlet chambers by two vertical divider plates, which are welded to the head, tubesheet, and [ ] (a,c,e,f)

Steam is generated on the shell side, flows upward and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes through a feedwater nozzle. Stratification and striping in the main nozzle region is limited by the use of heated startup feedwater during hot standby, startup, and power escalation.

During hot standby and startup, the steam generator is supplied by the startup feedwater system through the main feedwater nozzle. At about 4 percent power, the main feedwater pumps take over. At this time, the feedwater flowrate and temperature should be high enough to minimize stratification and striping conditions at the main feedwater nozzle.

During normal operation, feedwater is distributed circumferentially around the steam generator through a feedwater ring. The feedwater enters the ring through a welded thermal sleeve connection and leaves through inverted "J" tubes located at the flow holes atop the ring. The "J" tubes are arranged to distribute the feedwater uniformly into the downcomer annulus. The feed ring is designed to minimize conditions that can result in water hammer occurrences in the feedwater piping.

After feedwater is distributed into the steam generator, it mixes with recirculating water and the mixture flows downward through the annulus formed by the tube wrapper and the lower shell. At the bottom of the wrapper, the water is directed toward the center of the tube bundle by a flow distribution baffle. This baffle arrangement serves to maintain sufficient crossflow velocities adjacent to the tubesheet to minimize sludge deposition.

As the water rises through the bundle, it is converted to a steam-water mixture. The steam-water mixture then enters the steam drum section where centrifugal moisture separators remove most of the entrained water from the steam. The partially dry steam continues to the secondary separators where its moisture content is reduced to the design maximum of 0.25 percent. The moisture separators divert the separated water to the main water pool where it is combined with entering feedwater. The dry saturated steam exits from the steam generator through the outlet nozzle, which is provided with a steam flow restrictor (refer to Subsection 5.4.4).

5.4.2.3 Design Evaluation

5.4.2.3.1 Forced Convection

The effective heat transfer coefficient is determined by the physical characteristics of the WAPWR steam generator and the fluid conditions in the primary and secondary systems for the "nominal" 100 percent design case. It includes a conservative allowance for fouling and uncertainty. A designed heat transfer area is provided to permit the achievement of full design heat removal rate.

5.4.2.3.2 Natural Circulation Flow

In the event of loss of offsite power and consequential loss of forced circulation within the RCS, natural circulation removes core decay heat and permits the plant to be stabilized in the hot standby operational mode. Under this condition, reactor coolant pressure is maintained by the pressurizer, with backup heaters.

If pressurizer heaters are not available to maintain reactor coolant pressure, the RCS could be cooled through secondary side steam release. This operation would prevent pressure from dropping to saturation in the active portion of the RCS. Depending upon the circumstances, saturation conditions could occur in the upper head of the reactor vessel, leading to the formation of a steam bubble. This would not impede natural circulation flow, however, since any vapor that entered the hot legs, and subsequently the steam generators, would be condensed by heat transfer to the secondary side of the steam generators.

Vapor would be condensed in this manner as long as the steam generator tube bundle remains submerged.

5.4.2.3.3 Mechanical and Flow-Induced Vibration Under Normal Operating Conditions

In the design of the steam generators, the possibility of degradation of tubes due either to mechanical or to flow-induced vibration is thoroughly evaluated. This evaluation includes detailed analyses of the tube support systems as well as an extensive research program with tube vibration model tests.

In evaluating degradation due to vibration, sources of excitation such as those generated by primary fluid flowing within the tubes, mechanically induced vibration, and secondary fluid flow on the outside of the tubes are considered. During normal operation, the effects of primary fluid flow within the tubes and mechanically-induced vibration are considered to be negligible and should cause little concern. Thus, the primary source of tube vibration is the hydrodynamic excitation by the secondary fluid on the outside of the tubes.

In general, three vibration mechanisms have been identified:

1) Vortex shedding

2) Fluidelastic excitation

3) Turbulence.

Vortex shedding does not cause detectable tube bundle vibration. There are several reasons for this:

o Flow turbulence in the downcomer and tube bundle inlet region inhibit the formation of Von Karman's vortex trains.

- o The spatial variations of cross flow velocities along the tube preclude vortex shedding at a single frequency.

- o Both axial and cross flow velocity components exist on the tubes. The axial flow component disrupts the Von Karman vortices.

Fluidelastic excitation has been produced during testing. The maximum vibration amplitudes produced were two orders of magnitude smaller than those of the turbulent flow-induced vibrations. Therefore, fluidelastic excitation is a negligible contribution to steam generator tube bundle vibrations and is excluded.

Flow-induced vibrations due to flow turbulence result in tube stresses two orders of magnitude below the endurance limit (30,000 psi) of the tube material. Therefore, the contribution to fatigue is negligible, and fatigue degradation from flow-induced vibration is not anticipated during normal operation.

In summarizing the results of the steam generator vibration analyses and tests, an evaluation of all known modes of tube vibration mechanisms has been completed. The conclusions are that the primary source of tube vibration is fluid turbulence but that the magnitude of the vibration is so small that, when combined with its random nature, its contribution to tube fatigue is expected to be negligible. Therefore, fatigue degradation due to flow-induced vibration is not anticipated.

5.4.2.3.4 Allowable Tube Wall Thinning Under Accident Conditions

An evaluation has been performed to determine the extent of tube wall degradation that can be tolerated under accident conditions. Conservative case loading conditions are assumed to be imposed upon uniformly thinned tubes at the most critical location in the steam generator. Vibrations that occur due to such a postulated design basis accident are of such short duration that tube degradation by fatigue is not a concern. Also, the steam generator tubes are designed to provide an adequate safety margin against loss of integrity. The minimum manufacturing thickness less a conservative wall loss due to corrosion and erosion meets the ASME Code Requirements.

The results of a study made on the tubes selected for the WAPWR (0.75 inch outside diameter, 0.043 inch nominal wall thickness) under accident loadings are discussed in reference 2. These results demonstrate that a minimum wall thickness of 0.026 inches (40% degradation) would have a maximum faulted condition stress, due to combined loss-of-coolant accident LOCA and safe shutdown SSE earthquake loads, that is less than the allowable limit. This thickness (0.026 in) is 0.010 inches less than the minimum manufactured tube wall thickness of 0.039 inches, which is reduced to 0.036 inches by the assumed general corrosion and erosion loss of 0.003 inches. Thus, an adequate safety margin is exhibited.

The 3 mil corrosion rate is based on a conservative weight loss rate for Inconel tubing in 650°F, flowing, primary side reactor coolant fluid. The weight loss, when equated to a thinning rate and projected over a 40 year design objective with appropriate reduction after initial hours, is equivalent to 0.083 mils thinning. The assumed corrosion rate of 3 mils leaves a conservative 2.9 mils for general corrosion thinning on the secondary side.

5.4.2.4 Steam Generator Materials

5.4.2.4.1 Selection and Fabrication of Materials

All pressure boundary materials used in the steam generator are selected and fabricated in accordance with the requirements of Section III of the ASME Code. A general discussion of materials specifications is given in Subsection 5.2.3. Fabrication of reactor coolant pressure boundary (RCPB) materials is also discussed in Subsection 5.2.3, particularly in Subsections 5.2.3.3 and 5.2.3.4.

Testing has justified the selection of corrosion resistant thermally treated Inconel-690, a nickel-chromium-iron alloy (ASME SB-163 Code Case N-20), for the steam generator tubes. The channel head divider plate is Inconel (ASME SB-168). The interior surfaces of the reactor coolant channel head, nozzles,

(a,c,e,f) manways, and the [ ] are clad with austenitic stainless steel. The

,e,f) primary and [ ] of the tubesheet are weld clad with Inconel 600 (ASME SB-163). The tubes are then seal welded to the primary side tubesheet cladding. These fusion welds, performed in compliance with Sections III and IX of the ASME Code, are dye penetrant inspected and leak proof tested before each tube is hydraulically expanded the full depth of the tubesheet bore.

Code cases used in material selection are discussed in Subsection 5.2.1. The extent of conformance with Regulatory Guides 1.84, "Design and Fabrication Code Case Acceptability ASME Section III Division 1," and 1.85, "Materials Code Case Acceptability ASME Section III Division 1," is discussed in Section 1.8.

During manufacture, cleaning is performed on the primary and secondary sides of the steam generator in accordance with written procedures which follow the guidance of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants." Onsite cleaning and cleanliness control also follow the guidance of Regulatory Guide 1.37 as discussed in Section 1.8. Cleaning process specifications are discussed in Subsection 5.2.3.4.

The fracture toughness of the materials is discussed in Subsection 5.2.3.3. Adequate fracture toughness of ferritic materials in the RCPB is provided by compliance with 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," and paragraph NB-2300 of Section III of the ASME Code, and by meeting the requirements of General Design Criteria 1, 14, 15, and 31.

5.4.2.4.2 Steam Generator Design Effects on Materials

Several features have been introduced into the WAPWR steam generator to minimize the deposition of contaminants from the secondary side flow. Such deposits could otherwise produce a local environment in which adverse conditions could develop and result in material corrosion.

The WAPWR steam generator employs broached plate quatrefoil tube supports made of corrosion-resistant Incoloy 800 (ASME SB-409). Figure 5.4-4 illustrates the [(a,c,e

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5.4.2.4.3 Compatibility of Steam Generator Tubing with Primary and Secondary Coolants

Corrosion tests of thermally treated Inconel 690 have indicated that its corrosion rate is negligible. Accelerated stress corrosion tests in caustic and chloride aqueous solutions have indicated that thermally treated Inconel-690 resists general corrosion in aggressive water conditions.

However, operating experience with older steam generator designs has revealed areas on secondary surfaces where localized corrosion rates are significantly greater than the above mentioned low general corrosion rates. Both intergranular stress corrosion and tube wall thinning have occurred in localized areas.

The adoption of the all volatile treatment (AVT) control program minimizes the possibility for recurrence of the tube wall thinning phenomenon, which was primarily related to phosphate chemistry control. Successful AVT operation requires that the concentrations of impurities in the steam generator water be minimized. Thus, the potential for the formation of highly concentrated solutions in low flow zones is reduced. By restricting the total impurities in the steam generator and extended operation with impurities, the AVT program should minimize the possibility for the recurrence of corrosion in localized areas due to excessive levels of free caustic.

Laboratory testing has shown that the thermally treated Inconel-690 tubing is compatible with the AVT environment. Isothermal corrosion testing in high purity water has shown that, at normal stress levels, thermally treated Inconel-690 with normal microstructures has high resistance to intergranular stress corrosion cracking in extended high temperature exposure. These tests also revealed minimal general corrosion.

A comprehensive program of steam generator inspections, including the recommendations of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," with the exceptions as stated in Section 1.8, should provide for the detection of any degradation that might occur in the steam generator tubing.

Additional margin against stress corrosion cracking has been obtained by employing thermally treated Inconel-690 tubing. Thermal treatment of Inconel tubes has been shown to be particularly effective in providing resistance to caustic corrosion. Tubing used in the WAPWR steam generator is thermally treated in accordance with a verified treatment process.

Additional measures are incorporated in the WAPWR steam generator to reduce dryout and sludge accumulations. The flow distribution baffle forces the region of lowest lateral velocity to the center of the tubesheet where a 32 inch diameter untubed region is provided for sludge collection. A blowdown region is located at the center of the tubesheet.

(a,c,e,f)

[ ]

(a,c,e,

5.4.2.4.4 Secondary Side Cleaning Provisions

Several methods are employed to clean operating steam generators of secondary side deposits. Sludge lancing can be performed when necessary, as indicated by the results of steam generator tube inspection. Sludge lancing is a process in which a hydraulic jet inserted through an access port (handhole) loosens deposits which are then removed with a suction pump. Six 6-inch access ports are provided in each steam generator for sludge lancing and inspection. Four ports are located just above the tubesheet and two lie above the flow distribution baffle.

(a,c,e,f

During operation, continuous sampling is performed to monitor water chemistry. The sample connection is located near the top of the tubesheet in the region of lowest flow velocities.

5.4.2.5 Steam Generator Inservice Inspection

The steam generator is designed to permit inspection of Class 1 and 2 components, including individual tubes. The design includes a number of openings to permit access to both the primary and secondary sides of the steam generator. The specified inspection program complies with the applicable edition of the ASME Code, Division 1, Section XI as required by 10 CFR Part 50.55a.

The openings include four 18-inch manways, two for access to each chamber of the reactor coolant channel head and two in the steam drum for inspection and maintenance of the secondary side. Additional openings include the six 6-inch handholes for sludge lancing mentioned in the previous section. Also, two 3-inch access ports are located on the tubelane above the upper tube support plate to facilitate inspection of the U-bend region.

Additional access to the tube U-bend is provided through each of the three deck plates. Some of the deck plate openings are covered with welded, but removable, hatch plates.

Regulatory Guide 1.83 offers tube inspection recommendations including inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, and required actions based on findings. The steam generators are designed to permit access to tubes for inspection and repair or plugging, if necessary, according to guidelines expressed in Regulatory Guide 1.83. Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes," provides recommendations concerning tube plugging. The minimum requirements for inservice inspection of steam generators, including tube plugging criteria, are established as part of the Technical Specifications.

5.4.2.6 Quality Assurance

The steam generator quality assurance program is outlined in Table 5.4-4.

Radiographic inspection and acceptance standards shall conform to the requirements of Section III of the ASME Code.

Liquid penetrant inspection is to be performed on weld-deposited tubesheet cladding, channel head cladding, divider plate-tubesheet and divider plate-channel head weldments. Liquid penetrant inspection and acceptance standards shall conform to the requirements of Section III of the ASME Code.

Magnetic particle inspections are to be performed on the tubesheet forging, channel head [ ] nozzle forgings, and the following weldments: (a,c,e,f)

1) Nozzle to shell

2) Support brackets

3) Instrument connections (secondary)

4) Temporary attachments (after removal)

5) All accessible pressure retaining welds (after hydrostatic test).

Magnetic particle inspections and acceptance standards are to conform to the requirements of Section III of the ASME Code.

Ultrasonic tests are to be performed on the tubesheet forgings, tubesheet cladding, secondary shell and head plates, and nozzle forgings.

The heat transfer tubing is to be subjected to eddy current testing and to ultrasonic examination.

Hydrostatic tests are to be performed in accordance with Section III of the ASME Code.

5.4.3 Reactor Coolant Piping

5.4.3.1 Design Bases

The WAPWR reactor coolant system (RCS) piping is designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. Stresses are maintained within the limits of Section III, Nuclear Power Plant Components, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Code and material requirements are provided in Section 5.2.

Materials of construction are specified to minimize corrosion/erosion and ensure compatibility with the operating environment.

The piping in the RCS is Safety Class 1 and is designed and fabricated in accordance with the ASME Code, Section III, Class 1 requirements.

Stainless steel pipe conforms to American National Standards Institute (ANSI) B36.19 for sizes 1/2 in. through 12 in. and wall thickness schedules 40S through 80S. Stainless steel pipe outside the scope of ANSI B36.19 conforms to ANSI B36.10.

The minimum wall thicknesses of the loop piping and fittings, including the pressurizer surge line, are no less than those calculated using the ASME Code, Section III, Class 1 formula of Paragraph NB-3641.1(3). The nominal pipe bend radius is 5 pipe diameters, and ovality does not exceed 6 percent.

All butt welds, branch connection nozzle welds, and boss welds are of a full penetration design.

Processing and minimization of sensitization are discussed in Subsection 5.2.3.

Flanges conform to ANSI B16.5.

Socket weld fittings and socket joints conform to ANSI B16.11.

Inservice inspection is discussed in Subsection 5.2.4.

5.4.3.2 Design Description

The Class 1 RCS piping includes those sections of loop piping interconnecting the reactor vessel, steam generator, and reactor coolant pump. It also includes the following:

A. Charging line from the system isolation valve up to the branch connection on the reactor coolant loop (RCL).

B. Normal letdown line and excess letdown lines from the branch connections on the RCLs to the system isolation valves.

C. Pressurizer spray lines from two reactor coolant loop cold legs to the spray nozzle on the pressurizer vessel.

D. Residual heat removal (RHR) pump suction lines from the RCLs up to the designated isolation valve.

E. Safety injection - direct vessel injection lines from the designated check valve to the reactor vessel.

F. Accumulator lines from the designated check valve to the RCLs.

G. Core reflood tank lines from the designated check valves to the direct vessel injection lines.

H. Loop drain, sample and instrument[a] lines from the designated isolation valves to the RCLs.

I. Pressurizer surge line from one RCL hot leg to the pressurizer vessel surge nozzle.

J. Pressurizer spray scoops, sample connections[a] with scoops, reactor coolant temperature element installation bosses, and the temperature element wells.

(a) Lines with a 3/8-in. or less flow restricting orifice qualify as Safety Class 2. In the event of a break in one of these Safety Class 2 lines, the normal makeup system is capable of providing makeup flow while maintaining pressurizer water level.

K. All branch connection nozzles attached to RCLs.

L. Pressure relief lines from nozzles on top of the pressurizer vessel up to the pressurizer power operated relief valves and the pressurizer safety valves.

M. Auxiliary spray line from its isolation valve to the main pressurizer spray line.

N. Sample lines[(a)] from the pressurizer to the designated isolation valve.

O. Vent line from the reactor vessel head to the designated flow restrictor.

Principal design data for the reactor coolant piping are given in Table 5.4-5.

Details of the materials of construction and codes used in the fabrication of reactor coolant piping and fittings are discussed in Section 5.2.

The reactor coolant piping and fittings that make up the loops are austenitic stainless steel. The fittings will include "long radius elbows" with straight sections where the elbow will be welded to the next pipe section. The straight section will be sufficiently long to facilitate inservice inspection of the welded joints. Pipe fittings comply with the requirements of the ASME Code, Section II (parts A and C), Section III, and Section IX. All smaller piping that is part of the RCS, such as the pressurizer surge line, spray and relief valve lines, loop drains and connecting lines to other systems, are also austenitic stainless steel. All joints and connections are welded,

(a) Lines with a 3/8-in. or less flow restricting orifice qualify as Safety Class 2. In the event of a break in one of these Safety Class 2 lines, the normal makeup system is capable of providing makeup flow while maintaining pressurizer water level.

except for the pressurizer code safety valves, where flanged joints are used. Thermal sleeves are installed in the pressurizer inlet surge and spray line nozzles.

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

A. RHR pump suction lines, which are 45° down from the horizontal centerline. This enables the water level in the RCS to be lowered in the reactor coolant pipe while continuing to operate the RHRS, should this be required for maintenance.

B. Loop drain lines and the connection for measurement of water level in the RCS during refueling and maintenance operation.

C. The differential pressure taps for flow measurement, which are downstream of the steam generators at the first 90° elbow.

D. The pressurizer surge line, which may be attached as low as the horizontal centerline.

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

A. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the RCL flow adds to the spray driving force.

B. The RCS sample taps protrude into the main stream to obtain a representative sample of the reactor coolant.

C. The wide range and fast response temperature detectors are located in resistance temperature detector wells that extend into the reactor coolant loop piping.

5.4.3.3 Design Evaluation

Piping load and stress evaluation for normal operating loads, seismic loads, blowdown loads, and combined normal, blowdown, and seismic loads is discussed in Section 3.9 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

5.4.3.3.1 Material Corrosion/Erosion Evaluation

The water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications. (See Subsection 5.2.3.)

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 5.2-3. Maintenance of the water quality to minimize corrosion is accomplished using the CVCS and the sampling system which are described in RESAR-SP/90 PDA Module 13, "Auxiliary Systems".

5.4.3.3.2 Sensitized Stainless Steel

Sensitized stainless steel is discussed in Subsection 5.2.3.

5.4.3.3.3 Contaminant Control

Contamination of stainless steel and Inconel by copper, low melting temperature alloys, mercury, and lead is prohibited. Colloidal graphite is the only permissible thread lubricant.

Prior to application of thermal insulation, the austenitic stainless steel surfaces are cleaned and analyzed to a halogen limit of 0.0015 mg chloride/dm^2 and 0.0015 mg fluoride/dm^2.

5.4.3.4 Tests and Inspections

The RCS piping quality assurance program is outlined in Table 5.4-6.

Volumetric examination is performed throughout 100 percent of the wall volume of each pipe and fitting in accordance with the applicable requirements of Section III of the ASME Code for all pipe 27 1/2 in. and larger. All unacceptable defects are eliminated in accordance with the requirements of the same section of the code.

A liquid penetrant examination is performed on all accessible surfaces of each finished fitting, in accordance with the criteria of the ASME Code, Section III. Acceptance standards are in accordance with the applicable requirements of the ASME Code, Section III.

The pressurizer surge line conforms to SA-376, Grade 304, 304N, or 316 with supplementary requirements S2 (transverse tension tests) and S6 (ultrasonic test). The S2 requirement applies to each length of pipe. The S6 requirement applies to 100 percent of the piping wall volume.

The end of pipe sections, branch ends, and fittings are machined back to provide a smooth weld transition adjacent to the weld path.

5.4.4 Main Steam Line Flow Restrictors

5.4.4.1 Design Basis

The outlet nozzle of the steam generator is provided with a flow restrictor designed to limit steam flow in the unlikely event of a break in the main steam line. A large increase in steam flow will create a backpressure which further limits increase in flow. The flow restrictor performs the following functions:

- o Rapid rise in containment pressure is limited.

- o The rate of heat removal from the reactor is such as to keep the cooldown rate within acceptable limits.

- o Thrust forces on the main steam line piping are reduced.

- o Stresses on internal steam generator components, particularly the tube sheet and tubes, are limited.

The restrictor is configured to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

The flow restrictor consists of seven Inconel venturi inserts which are installed in the holes in an integral steam outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle and the other six equally spaced around it. After insertion into the nozzle forging holes, the Inconel venturi inserts are welded to the Inconel cladding on the inner surface of the forging.

5.4.4.3 Design Evaluation

The flow restrictor design has been analyzed to determine its structural (a,c) adequacy. The equivalent throat diameter of the steam generator outlet is [ ] inches and the resultant pressure drop through the restrictor at 100-percent (a,c) steamflow is approximately [ ] psi. This is based on a design flow rate of (a,c) [ ] lb/h. Materials of construction and manufacturing of the flow restrictor are in accordance with Code Class 1 Section III of the ASME Code. The method for seismic analysis is dynamic.

5.4.4.4 Inspections

Since the restrictor is not part of the steam system boundary, no inspections beyond those performed during fabrication are anticipated.

5.4.10 Pressurizer

5.4.10.1 Design Bases

The pressurizer provides a point in the reactor coolant system (RCS) where liquid and vapor are maintained in equilibrium under saturated conditions for control of pressure of the RCS during steady-state operations and transients.

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

A. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

B. The water volume is sufficient to prevent the heaters from being uncovered during a step load increase of 10 percent of full power, initiated at any power level above 15 percent.

C. The steam volume is large enough to accommodate the insurge resulting from a loss of all external electrical load, with automatic reactor control and 75 percent steam dump, without initiating reactor trip.

D. The steam volume is large enough to prevent water relief through the safety valves following a loss of load assuming failure of the rod control and steam dump systems.

E. Letdown flow will not be isolated on low pressurizer water level following a reactor trip.

F. A low pressurizer pressure safety injection signal will not be generated following a reactor trip and turbine trip, assuming normal operation of NSSS control systems.

The surge line is sized to maintain the pressure drop between the RCS and the safety valves within allowable limits during a design discharge flow from the safety valves.

The surge line is designed to withstand the thermal stresses resulting from volume surges of relatively hotter or colder water which may occur during operation.

5.4.10.2 Design Description

5.4.10.2.1 Pressurizer and Connected Piping

The pressurizer is a vertical, cylindrical vessel having hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all internal surfaces exposed to the reactor coolant. The surge line is constructed of stainless steel.

The general configuration of the pressurizer is shown in Figure 5.4-5. Design data for the pressurizer are given in Table 5.4-7. Applicable codes and the material requirements are provided in Section 5.2.

The pressurizer surge line connects the pressurizer to one reactor coolant hot leg, thus enabling continuous coolant volume adjustments between the RCS and the pressurizer. The surge line nozzle is located in the bottom head of the pressurizer. A retaining screen is located above the nozzle to prevent passage of any foreign matter from the pressurizer to the RCS. Baffles in the lower section of the pressurizer prevent an insurge of cold water from flowing directly to the steam/water interface and also assist in mixing.

Electrical heaters are installed through the bottom head of the pressure vessel. The heaters are removable for maintenance or replacement.

The spray line nozzle and the relief and safety valve connections are located in the top head of the pressurizer vessel. Spray flow is modulated by automatically controlled air operated valves. The spray valves also can be operated manually from the control room. A small continuous spray flow is provided through a manual bypass valve around each power-operated spray valve to minimize the boron concentration difference between the pressurizer liquid and the reactor coolant and to prevent excessive cooling of the spray piping.

During an outsurge of water from the pressurizer, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the low-pressure reactor trip setpoint. During an insurge from the RCS, the spray system, which is fed from two cold legs, condenses steam in the pressurizer to prevent the pressurizer pressure from reaching the setpoint of the PORVs. The 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 pressurizer safety valve and power operated relief valve connections are also located in the top head of the pressurizer vessel. The piping and support arrangement for these valves is designed such that the effects of thrust forces on the piping system from valve operations is minimized. The piping analysis is discussed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design."

Material specifications are provided in Table 5.2-1 for the pressurizer and the surge line. Design transients for the components of the RCS are discussed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

5.4.10.2.2 Pressurizer Spray and Relief Line Instrumentation

Refer to RESAR-SP/90 PDA Module 9, "I&C and Electric Power" for details of the instrumentation associated with measurement of pressurizer pressure, level, and temperature.

Temperatures in the spray lines from the cold legs of two loops are measured and indicated. Alarms from these signals are actuated to warn the operator of low spray water temperature which is indicative of insufficient flow in the spray lines.

Temperatures in the pressurizer safety and relief valves discharge lines are measured and indicated. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve. High temperature alarms are initiated if the leakage is abnormal.

5.4.10.3 Design Evaluation

5.4.10.3.1 System Pressure Control

RCS pressure is determined by pressurizer pressure whenever a steam volume exists in the pressurizer. At other times (i.e. when the pressurizer is water solid) RCS pressure is controlled by the chemical and volume control system. RCS pressure is essentially the same as pressurizer pressure except for elevation heads and friction pressure drop under forced circulation.

During plant heatup and cooldown operations, pressurizer pressure is controlled by the operator, exercising manual control of the pressurizer spray system and pressurizer heaters. The pressure is controlled sufficiently high to assure adequate core subcooling and adequate NPSH and No. 1 seal ΔP for the reactor coolant pumps.

During power operation, i.e. when reactor power is greater than approximately 10 percent of full power, pressurizer pressure is controlled automatically.

5.4.10.3.2 Pressurizer Level Control

(a,c) The normal operating water volume at full load conditions is approximately [ ] percent of the free internal vessel volume. Under part load conditions the water volume in the pressurizer is reduced proportionally with reductions in (a,c) plant load to approximately [ ] percent of the free internal vessel volume at the zero-power condition.

5.4.10.3.3 Pressure Setpoints

The RCS design and operating pressure, together with the safety valve, PORV, pressurizer spray and protection system pressure setpoints, are listed in Table 5.4-8. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system safety and relief valve characteristics. The low pressurizer pressure reactor trip does not require a coincident low water level signal. This is in accordance with the recommendations of Action Item II.K.3.17 of NUREG-0660.

Temperature changes which may affect the relief valve setpoints have been considered. Normal ambient air temperature variations have no significant effects. However, cold valves relieving hot fluid may show reduced setpoints; therefore, this has been considered in the design of such valves.

5.4.10.3.4 Pressurizer Spray

Two separate automatically controlled spray valves with remote manual overrides are used to control pressurizer spray. These valves are normally closed. 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 routed to the pressurizer is such as to minimize the volume of piping normally exposed to pressurizer steam. The control valve and bypass valve in each spray line are located low enough to prevent uncovering of the valves (and exposure to pressurizer steam) if the associated reactor coolant pump is stopped when the pressurizer water level is low, e.g. no-load condition. This helps to minimize thermal shock and pump re-start. The design spray rate is selected to prevent the pressurizer pressure from reaching the operating set-point of the PORVs during a step reduction in power level of 10 percent of full load, without actuating the steam dump system.

The pressurizer spray lines and valves are large enough to provide the required spray flowrate under the driving force of the pressure differential between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet extends into the cold leg piping in the form of a scoop in order to utilize the velocity head of the reactor coolant loop flow to add to the spray driving force. The spray valves and spray line connections are arranged so that the spray operates, although at a reduced capacity, when one reactor coolant pump is not operating. The spray line also assists in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

A separate auxiliary spray line from the chemical and volume control system to the main spray line is also provided. This is used to route auxiliary spray to the vapor space of the pressurizer during cooldown when the reactor coolant pumps are not operating. The pressurizer main and auxiliary spray piping is designed to withstand the thermal stresses resulting from the introduction of relatively colder spray water.

5.4.10.4 Tests and Inspections

The pressurizer is designed and constructed in accordance with the ASME Code, Section III.

To implement the requirements of the ASME Code, Section XI, the following welds are designed and constructed to present a smooth transition surface between the parent metal and the weld metal. The weld surface is ground smooth for ultrasonic inspection.

- o Support skirt to the pressurizer lower head.
- o Surge nozzle to the lower head.

- Safety, relief, and spray nozzles to the upper head.
- Nozzle to safe end attachment welds.
- All girth and longitudinal full-penetration welds.
- Manway attachment welds.

The liner within the safe end nozzle region extends beyond the weld region to maintain a uniform geometry for ultrasonic inspection.

5.4.11 Pressurizer Relief Discharge System

The pressurizer relief discharge system collects, cools, and directs the steam and water discharged from various safety and relief valves in the containment for processing. The system consists of the pressurizer relief tank (PRT), the pressurizer safety and relief valve discharge piping, the relief tank internal sparger, spray nozzles with associated header and piping, the tank nitrogen supply, the drain to the liquid waste processing system, the relief tank rupture discs and the rupture disc discharge piping.

5.4.11.1 Design Bases

The system design, including the PRT design volume, is based on the requirement to condense and cool a discharge of steam equivalent to 110 percent of the full power pressurizer steam volume, without exceeding a pressure/temperature condition of [ ] in the PRT. These values are well below the PRT design conditions of [ ] Additional design data for the tank are given in Table 5.4-9. (a,c) (a,c)

The minimum volume of water in the PRT is determined by the energy content of the steam to be condensed and cooled, by the assumed initial temperature of the water, and by the desired final temperature of the water volume. The initial water temperature is assumed to be 120°F, which corresponds to the design maximum expected containment temperature for normal conditions. Provision is made to permit cooling of the water in the tank should the water temperature rise above 120°F during plant operation. The design final temperature, following a design discharge to the tank, is 200°F, which allows the contents of the tank to be drained directly to the liquid waste processing system without cooling.

The PRT saddle supports and anchor bolt arrangement are designed to withstand the loadings resulting from the vessel seismic, static, and nozzle loadings.

The rupture disc discharge piping consists of a line from each PRT rupture disc routed to the emergency water storage tank of the Integrated Safeguards System. These lines terminate under water in the EWST; numerous small holes in the underwater end of the line form a sparger. In the event of high PRT pressure and rupture disc opening, the EWST sparger provides for uniform distribution and quenching of the steam discharge.

5.4.11.2 Design Description

The piping and instrumentation diagram for the pressurizer relief discharge system is given in Figure 5.1-2.

The steam and reactor grade water discharged from the various safety and relief valves inside the containment is routed to the PRT.

The pressurizer safety and relief valve piping and support arrangement in Figure 5.4-7 shows the valve discharge piping, as well as the piping upstream of the safety and relief valves. The piping upstream of the valves, which is not considered part of the pressurizer relief discharge system, includes the following:

A. Three lines with loop seal arrangements connecting the pressurizer nozzles to the three safety valves.

B. A line from the power operated relief valve nozzle branching to the power operated relief valves (PORVs), which have individual water seals and motor operated isolation valves.

The pressurizer safety and relief valve discharge piping consists of:

A. A common piping manifold (supported over the top of the pressurizer) into which the safety and relief valves discharge.

B. Safety valve discharge lines to the manifold.

C. Relief valve discharge lines to the manifold.

D. A manifold discharge header (downcomer).

E. Piping to the PRT.

The main support structure for the safety and relief valve piping consists of four column members, equally spaced around the common manifold, coupled to the valve support brackets on the pressurizer. No welding to the pressurizer is required. To increase the natural frequency of the system, auxiliary cross-members are provided from the common manifold to the main support columns. The safety valves are provided with a bottom saddle type support coupled to the auxiliary crossmembers. The relief valves are positioned above the manifold, and the relief valve lines are supported at various points along the manifold. The pressurizer safety and relief valve piping is constructed of austenitic stainless steel. Design data for the pressurizer safety and relief valve piping are given in Table 5.4-5.

The piping upstream of the safety and relief valves is part of the reactor coolant system (RCS) and is designed and fabricated in accordance with American Society of Mechanical Engineers (ASME) Code, Section III, Class 1 requirements. The discharge piping between these valves and the emergency water storage tank is classified as Safety Class 3 and is designed and fabricated to ASME Code, Section III, Class 3. The support structure for the pressurizer safety and relief valve piping arrangement is designed and fabricated to ASME Code, Section III, Subsection NF.

All of the above described piping is designed to Seismic Category I requirements. The principal design codes are listed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

The general configuration of the PRT is shown in Figure 5.4-6. The tank is a horizontal, cylindrical vessel with elliptical dished heads. The vessel is constructed of austenitic stainless steel and is overpressure protected by means of two safety heads with stainless steel rupture discs. Also shown are the flanged connection for the pressurizer safety and relief valve discharge header, the spray water inlet, the bottom drain connection, the gas venting N_2 addition connection, the rupture disc holders and the vessel supports. The tank is classified as Safety Class 3 and is designed and fabricated to Section III, Division 1, Class 3 of the ASME Code.

The tank normally contains water and a predominantly nitrogen atmosphere. In order to obtain effective condensing and cooling of the discharged steam, the tank is installed horizontally so that the steam can be discharged through a sparger pipe located near the bottom, under the water level. The sparger holes are designed to ensure good mixing of the discharged steam with the water initially in the tank.

A nitrogen gas blanket is used to control the atmosphere in the tank and to allow room for the expansion of the original water, plus the condensed steam discharge. The tank gas volume is sized such that the pressure following a design basis steam discharge does not exceed 50 psig, assuming an initial pressure of 3 psig. This pressure is low enough to prevent opening of the rupture discs. Provisions are made to permit the gas in the tank to be periodically analyzed to determine the concentration of hydrogen and/or oxygen.

The internal spray and bottom drain on the PRT function to cool the water when the temperature exceeds 120°F, as in the case following a steam discharge. The contents are cooled by a feed-and-bleed process, with cold reactor makeup water entering the tank through the spray water inlet and the warm mixture draining to the reactor coolant drain system. The contents may also be cooled by recirculation through a heat exchanger of the liquid waste processing system.

5.4.11.3 Design Evaluation

The pressurizer relief discharge system does not constitute part of the reactor coolant pressure boundary in accordance with 10 CFR 50.2, since all of its components are downstream of the RCS safety and relief valves; thus, General Design Criteria 14 and 15 are not applicable. Furthermore, complete failure of the auxiliary systems serving the PRT will not impair the capability for safe plant shutdown.

The design of the system piping layout and piping restraints is consistent with the hazards protection requirements discussed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design". The safety and relief valve discharge piping is restrained so that the integrity and operability of the valves are maintained in the event of a rupture. Regulatory Guide 1.67 is not applicable since the system is not an open discharge system.

The pressurizer relief discharge system is capable of handling the design discharge of steam without exceeding the design pressure and temperature of the pressurizer relief tank. The volume of nitrogen in the PRT is that required to limit the maximum pressure accompanying the design basis discharge to 50 psig, half the design pressure of the tank. The volume of water in the PRT is capable of absorbing the heat from the assumed discharge while maintaining the water temperature below 200°F.

If a discharge results in a pressure that exceeds the design, the rupture discs on the tank would burst, allowing the discharge to flow into the emergency water storage tank. The rupture discs on the relief tank have a relief capacity equal to or greater than the combined capacity of the pressurizer safety valves. The tank and rupture discs holders are also designed for full vacuum to prevent tank collapse, if the contents cool following a discharge without nitrogen being added.

The discharge piping from the pressurizer safety and relief valves to the PRT is sufficiently large to prevent backpressure at the safety valves from exceeding 20 percent of the setpoint pressure at full flow.

The recommendations of NUREG-0737, Action Items II.G.1 and II.K.3.1 are met as discussed below. The pressurizer is equipped with three Class 1E PORVs (solenoid operated) and three Class 1E PORV block valves (motor operated). The PORV and associated block valve on one line are supplied with control and motive power from train A, while the other PORV and associated block valve on the other line are powered from train B.

The PORV block valves 1HV-8000A and 1HV-8000B are powered from Class 1E buses. These buses are normally supplied from offsite power. In the event of a loss of offsite power, these buses are automatically loaded onto the diesels. PORVs are Class 1E dc solenoid valves and are powered from redundant Class 1E 125-V dc trains A and B, respectively. The train assignment for power to the PORVs and block valves is based on:

A. The ability to open one of the parallel pressurizer vent paths in conjunction with a single failure.

B. The ability to close both parallel paths in conjunction with a single failure. (Capability to isolate both parallel paths in conjunction with a single failure is based upon the fact that the solenoid-operated PORVs are qualified, dc powered, and designed to fail closed).

The PORV block valves are interlocked with pressurizer pressure such that they are opened automatically when the pressure exceeds 1900 psia and closed automatically when the pressure falls below 1900 psia. The automatic closure feature affords additional protection against the potential consequences of a stuck open PORV.

5.4.11.4 Instrumentation Requirements

The following instrumentation is provided on the main control board:

A. The PRT pressure transmitter provides a signal to an indicator. An alarm is provided to indicate high tank pressure.

B. The PRT level transmitter supplies a signal to an indicator. High and low level alarms are also provided.

C. The temperature of the water in the PRT is displayed by an indicator. An alarm actuated by high temperature informs the operator that cooling of the tank contents is required.

5.4.11.5 Inspection and Testing Requirements

The nondestructive examinations performed during fabrication of the piping from the pressurizer to the tee connection in the discharge header are identified in Table 5.4-10.

The PRT and connected piping is subject to nondestructive and hydrostatic testing during construction and after installation in accordance with Section III, Division 1, Class 3 of the ASME Code.

The downcomer piping to the PRT is subject to nondestructive and hydrostatic testing during construction.

During plant operation, periodic visual inspections and preventive maintenance are conducted on the system components.

5.4.12 Valves

5.4.12.1 Design Bases

As noted in Section 5.2, all valves out to and including the second valve that is normally closed or capable of automatic or remote closure, larger than 3/4 inch nominal size, are American Society of Mechanical Engineers (ASME) Code, Section III, Class 1 valves. All 3/4-inch or smaller valves in lines connected to the reactor coolant system (RCS) are Class 2 since the interface with the Class 1 piping is provided with suitable flow limiting orificing. Design data for the RCS valves are given in Table 5.4-11. For additional discussion on the RCS safety valves and relief valves, see Subsection 5.4.13

For a check valve to qualify as part of the RCPB, the check valve must be located inside the containment system. When the second of two normally closed check valves is considered part of the RCPB, means are provided to assess periodically any backflow leakage of the first valve.

All valves in the RCS that 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.

5.4.12.2 Design Description

All RCS valves larger than 2 inches nominal size and that normally contain radioactive fluid are provided with two sets of stem packing and a leakoff connection between the packing sets. Throttling control valves regardless of size are also provided with double-stem packing and a packing leakoff connection. In general, RCS leakoff connections are piped to a closed collection system. Leakage to the atmosphere is essentially zero for these valves.

Gate valves at the engineered safety features interface are wedge design and are essentially straight through. The wedges are flex-wedge or solid. All gate valves have backseats.

Globe valves are "T" and "Y" styles of outside screw and yoke construction.

Check valves are swing type for sizes 2-1/2 inches or larger. All check valves which contain radioactive fluid are stainless steel and do not have body penetrations other than the inlet, outlet, and bonnet. The check hinge is serviced through the bonnet. All operating parts are contained within the valve body. The disc has limited rotation to provide a change of seating surface and alignment after each valve opening.

5.4.12.3 Design Evaluation

The design requirements for Class 1 valves, as discussed in Section 5.2, limit stresses to levels which ensure the structural integrity of the valves. In addition, the testing programs described in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design", demonstrate the ability of the valves to operate, as required, during anticipated and postulated plant conditions.

Reactor coolant chemistry parameters are specified in the design specifications to ensure compatibility of valve construction materials with the reactor coolant. To ensure that this coolant/materials compatibility continues the chemical composition of the coolant is analyzed periodically, and chemistry parameters are maintained in accordance with the Technical Specifications.

5.4.12.4 Tests and Inspections

Hydrostatic shell test and seat leakage and functional tests are performed on all RCS valves. The tests and inspections discussed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" are performed to ensure the operability of the active valves.

There are no full penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection to the extent practical. The valve nondestructive examination program is given in Table 5.4-12. Inservice inspection is discussed in Subsection 5.2.4.

5.4.13 Safety And Relief Valves

5.4.13.1 Design Bases

The combined capacity of the pressurizer safety valves can accommodate the maximum pressurizer surge resulting from complete loss of load without causing RCS pressure to exceed 110 percent of the design pressure at the point of highest pressure in the system. Sizing of the pressurizer safety valves is discussed in Subsection 5.2.2.

The pressurizer power operated relief valves (PORVs) are designed to limit pressurizer pressure to a value below the high pressure reactor trip setpoint. The PORVs are designed to fail in the closed position on loss of actuating power.

5.4.13.2 Design Description

The pressurizer safety valves are of the totally enclosed pop type. They are spring loaded self actuated by direct fluid pressure and have backpressure compensation features. The set pressure of each valve is 2485 psig.

The pipe connecting each pressurizer nozzle to its safety valve inlet is shaped in the form of a loop seal. Condensate resulting from normal heat losses accumulates in the lower part of the loop. This water seal minimizes leakage of hydrogen gas and steam through the safety valve seats. If the pressurizer pressure exceeds the set pressure, the safety valves start lifting and the water from the seal discharges during the accumulation period.

The pressurizer PORVs are solenoid operated valves which respond to a signal from a pressure-sensing system or to manual control. Remotely operated block valves are provided to isolate the inlets to the PORVs if excessive leakage develops.

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 relief valves are designed to limit the pressurizer pressure to a value below the high pressure trip setpoint for all design load reduction transients up to and including a full load rejection with steam dump actuation and automatic reactor control. (It should be noted that this is the same operational event on which the design pressurizer spray rate is based). Operation of these valves also limits the undesirable opening of the spring loaded safety valves. In addition, with the programmed setpoint feature, these valves are used in connection with protection against RCS overpressurization while the reactor coolant is at low temperature. They also can be used to depressurize the RCS if plant shutdown using only safety grade equipment is required, and an alternate coolant letdown path if the pressurizer should become filled.

Remotely operated block valves are provided to isolate the inlets to the power operated relief valves. The block valves can be operated either automatically or manually. In the automatic mode, i.e., when the plant is operating under automatic pressure control the valves are opened whenever the pressurizer pressure is above the block valve set pressure, which is approximately 350 psi below the normal pressurizer operating pressure of 2250 psia. They are closed automatically if the pressure falls below the set pressure. These characteristics serve to protect against the consequences of a stuck-open PORV when the plant is operating under automatic pressurizer pressure control. An override feature permits manual operation of the block valves. This capability would be required for certain off-normal events such as a safety grade cold shutdown requiring manual PORV operation, or if excessive PORV leakage occurs.

As a safeguard against spurious operation of the power operated relief valves or of the PORV block valves, coincident high pressure signals derived from any two of the four pressurizer pressure transmitters are required to open these valves or to keep them open.

In accordance with the requirements of 10CFR50.34(f)(2)(xi), positive position indication (open or closed) is provided in the control room for the pressurizer safety valves and power operated relief valves.

Temperatures in the pressurizer safety and relief valve discharge lines are measured, and an indication and a high alarm are provided on the main control board. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve.

The PORVs provide the safety related means for reactor coolant system depressurization to achieve cold shutdown. For a discussion of the use of these valves to achieve safety grade cold shutdown, see Subsection 5.4.7 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System".

Design parameters for the pressurizer safety valves and power relief valves are given in Table 5.4-13.

5.4.13.3 Design Evaluation

The pressurizer safety valves are sized to prevent reactor coolant system pressure from exceeding 110 percent of system design pressure, in compliance with the ASME Code, Section III.

The pressurizer PORVs are sized to prevent actuation of the reactor high pressure trip for all design transients up to and including the design step load decreases with steam dump. The relief valves also limit undesirable opening of the spring loaded safety valves.

5.4.13.4 Tests and Inspections

All safety and relief valves are subjected to hydrostatic tests, seat leakage tests, operational tests, and inspections, as required. For safety valves that are required to function during a faulted condition, additional tests are performed. These tests are described in RESAR-SP/90 PDA Module 7, "Structural/

Equipment Design". There are no full penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection.

Safety and relief valves similar to those in the WAPWR have been tested under the Electric Power Research Institute safety and relief valve test program. The completion of this program addresses the requirements of 10CFR50.34(f)(2)(x), RCS Valve Testing.

5.4.14 Component Supports

5.4.14.1 Design Bases

Component supports allow unrestrained lateral thermal movement of the loop during plant operation and provide restraint to the loops and components during accident or seismic conditions. The loading combinations and design stress limits are discussed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design". Support design is in accordance with the American Society of Mechanical Engineers (ASME) Code, Section III, Subsection NF. The design maintains the integrity of the RCS boundary for normal, seismic, and accident conditions and satisfies the requirements of the ASME Code.

Conformance with Regulatory Guides 1.124 and 1.130 is discussed in Section 1.8.

5.4.14.2 Description

The support structures are welded structural steel sections. Linear-type structures (tension and compression struts, columns, and beams) are used in all cases except for the reactor vessel supports, which are plate type structures. Attachments to the supported equipment are non-integral type that are bolted to or bear against the components. The supports-to-concrete attachments are either anchor bolts or embedded fabricated assemblies.

The supports permit virtually unrestrained thermal growth of the supported systems but restrain vertical, lateral, and rotational movement resulting from seismic or pipe break loadings. This is accomplished using spherical bushings in the columns for vertical supports and girders, bumper pedestals, hydraulic (or mechanical) snubbers, and tie rods for lateral support.

Because of manufacturing and construction tolerances, sample adjustment in the support structures is provided to ensure proper erection alignment and fit-up. This is accomplished by shimming or grouting at the supports-to-concrete interface and by shimming at the supports-to-equipment interface.

The supports for the various components are described in the following paragraphs.

5.4.14.2.1 Reactor Pressure Vessel

Supports for the reactor vessel (Figure 5.4-8) consist of individual, air-cooled, rectangular box structures located beneath the vessel nozzles and lower support keys. All of these structures are bolted to the primary shield wall concrete. The box structures consist of a horizontal top plate that receives loads from the reactor vessel shoe, a horizontal bottom plate that transfers the loads to the primary shield wall concrete, and connecting vertical plates. The box structures are air-cooled to maintain the supporting concrete temperature within acceptable levels.

5.4.14.2.2 Steam Generator

As shown in Figure 5.4-9, the steam generator supports consist of the following elements:

A. Vertical support

Four individual columns provide vertical support for each steam generator. These are bolted at the top to the steam generator and at the bottom to the concrete structure. Spherical ball bushings at the top and bottom of each column allow unrestrained lateral movement of the steam generator during heatup and cooldown. The column base design permits both horizontal and vertical adjustment of the steam generator for erection and adjustment of the system.

B. Lower Lateral Support

Lateral support is provided at the generator tubesheet by fabricated steel girders and struts. These are bolted to the compartment walls and include bumpers that bear against the steam generator but permit unrestrained movement of the steam generator during changes in system temperature. Stresses in the beams caused by wall displacements during compartment pressurization are considered in the design (if applicable).

C. Upper Lateral Support

Upper lateral support of the steam generator is provided by a built-up ring plate girder at the operating deck. Two-way acting snubbers restrain sudden seismic motion but permit the normal thermal movement of the steam generator. Movement perpendicular to the thermal growth direction of the steam generator is prevented by struts.

5.4.14.2.3 Reactor Coolant Pump

Three individual columns, similar to those used for the steam generator, provide the vertical support for each pump. Lateral support for seismic loading is provided by lateral tension tie bars. The pump supports are shown in Figure 5.4-10.

5.4.14.2.4 Pressurizer

The supports for the pressurizer, as shown in Figure 5.4-11, consist of:

A. A steel ring plate between the pressurizer skirt and the supporting structure. The ring serves as a leveling and adjusting member for the pressurizer.

B. The upper lateral support consists of struts cantilevered off the compartment walls that bear against the lugs attached to the pressurizer.

5.4.14.2.5 Control Rod Drive Mechanism (CRDM) Supports

The support system for the CRDM provides lateral restraint to limit CRDM deflections due to lateral seismic loadings. The CRDM support system consists of the following:

A. Upper Seismic Support

1. A support consisting of individual plates (CRDM and DRDM RPI plates) extends across the top of the control rod drive housing columns. These plates are encompassed by an outer ring which limits the total lateral deflection of the CRDM rod travel housings.

2. Vertical support of the outer ring is provided by the Integrated Head Package which is attached to the reactor vessel head.

3. Horizontal support of the outer ring is provided by lateral tension tie rods which are pinned to the refueling cavity wall.

B. Intermediate Seismic Support

1. An intermediate seismic support plate is installed at the top of each CRDM latch housing. These plates interface with adjoining plates to provide a horizontal support grid across the entire R/V head. These plates are encompassed by an outer ring which limits the total lateral deflection of the CRDM latch housings.

2. Vertical support of the outer ring is provided by the Integrated Head Package which is attached to the reactor vessel head.

3. Horizontal support of the outer ring is provided by the Integrated Head Package structure which transmits the load into the reactor vessel head.

5.4.14.2.6 Displacer Rod Drive Mechanism (DRDM) Supports

The support system for the DRDM provides lateral restraint to limit DRDM deflections due to lateral seismic loadings. The DRDM support system consists of the following:

A. A support, consisting of individual plates (CRDM and DRDM RPI plates) extends across the top of the control rod drive housing columns. These plates are encompassed by an outer ring which limits the total lateral deflection of the DRDM rod travel housings.

B. Vertical support of the outer ring is provided by the Integrated Head Package which is attached to the reactor vessel head.

C. Horizontal support of the outer ring is provided by lateral tension tie rods which are pinned to the refueling cavity wall.

5.4.14.3 Evaluation

Detailed evaluation ensures the design adequacy and structural integrity of the reactor coolant loop and the primary equipment supports system. This detailed evaluation is made by comparing the analytical results with established criteria for acceptability. Structural analyses are performed to demonstrate design adequacy for safety and reliability of the plant in case of large or small seismic disturbance conditions. Loads which the system is expected to encounter often during its lifetime (thermal, weight, and pressure) are applied, and stresses are compared to allowable values. The modeling and analysis methods are discussed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

The reactor vessel supports are not designed to provide restraint to vertical uplift movement resulting from seismic loadings. However, RPV motion resulting from seismic events is included in the reactor coolant system analyses as discussed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

5.4.14.4 Tests and Inspections

Nondestructive examinations are performed in accordance with the procedures of the ASME Code, Section V, except as modified by the ASME Code, Section III, Subsection NF.

5.4.15 Reactor Vessel Head Vent System

The reactor vessel head vent system (RVHVS) is provided to remove potential accumulations of noncondensable gases or steam from the reactor vessel head. This system is designed to help mitigate a possible condition of inadequate core cooling or impaired natural circulation resulting from the accumulation of noncondensable gases in the reactor coolant system (RCS). The design of the RVHVS is in accordance with the requirements of 10CFR50.34 (f)(2)(vi), RCS High Point Vents, as discussed below.

5.4.15.1 Design Basis

The RVHVS is designed to remove noncondensable gases or steam from the RCS via remote manual operations from the control room and to discharge them to the pressurizer relief tank. The system is designed to vent a volume of hydrogen at system design pressure and temperature approximately equivalent to one-half of the RCS volume in one hour.

The system provides for venting the reactor vessel head using only safety grade equipment and is designed to satisfy applicable requirements and industry standards, including ASME Code classification, safety classification, single-failure criteria, and environmental qualification.

The piping and valves of the RVHVS are classified in accordance with their importance to nuclear safety. Safety classification for the various portions of the system are as follows:

- o Safety Class 1 - From the reactor vessel head connection to the "tee" at the solenoid operated isolation valve inlet lines. This portion terminates in a Class 1-Class 2 flow restrictor.

- o Safety Class 2 - From the flow restrictor to and including the downstream valve in each set of isolation valves. This portion, along with the Class 1 portion, constitutes part of the reactor coolant pressure barrier.

- o Safety Class 3 - The line from the solenoid operated isolation valves to the pressurizer safety valve discharge header. As there are no valves in this line, SC-3 designation is necessary for consistency with the SC-3 discharge header classification.

These classifications are indicated in Figure 5.1-2.

Components of the RVHVS are designed and fabricated in accordance with the applicable requirements of the ASME Code, Section III, Code Class 1, 2 and 3 as appropriate.

5.4.15.2 Design Description

The RVHVS consists of a single active failure proof flow path with redundant isolation valves. The equipment design parameters are listed in Table 5.4-14.

The active portion of the system consists of four 1 inch open/close solenoid-operated isolation valves, in a series-parallel (quad) arrangement, connected through a 1 inch vent line, to a location near the center of the reactor vessel head. The two valves in series in each flow path minimizes the possibility of reactor coolant pressure boundary leakage. The isolation valves in one flow path are powered by one vital power supply and the valves in the second flow path are powered by a second vital power supply. The isolation valves are fail-closed and are normally closed. Downstream of the valve quad a single 1 inch line routes the flow to the pressurizer safety valve discharge header.

Capability for local manual venting of the reactor vessel head during normal plant startup is provided by a short 1 inch line, with a single normally closed manual valve, connected to the RVHVS line upstream of the valve quad.

The vent system piping and valves are supported such that the resulting loads and stresses on the system equipment and on the connection to the vessel head are within acceptable limits.

5.4.15.3 Design Evaluation

The combination of valve arrangement, safety grade train assignments and valve failure modes assure capability for vessel head venting and also for venting isolation with any single active valve failure. The two valves in series in each flow path are normally deenergized. This essentially eliminates the possibility of an opened flow path due to the spurious opening of one valve. Thus, power lockout to any valve is not considered necessary to assure isolation capability. Similarly, the two parallel flow paths assure capability to open a vent path in the event that one of the valves will not open.

A break of the RVHVS line would result in a small loss-of-coolant accident (LOCA) equivalent to a 1 inch diameter break and similar to those analyzed in Reference 3. Since a break in the head vent line would behave similarly to the hot leg break case presented in Reference 3, the results presented therein are applicable to a RVHVS line break. The Reference 3 results show that the hot leg break case results in no calculated core uncovery.

5.4.15.4 Tests and Inspections

Inservice inspection is conducted in accordance with Subsection 5.2.4.

5.4.15.5 Instrumentation Requirements

The system is operated from the main control room or the shutdown panels. The open-closed position of each valve is indicated in the main control room.

5.4.16 REFERENCES

1. "Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973.

2. DeRosa, P., et al., "Evaluation of Steam Generator Tube, Tubesheet and Divider Plant Under Combined LOCA Plus SSE Conditions," WCAP-7832, December 1973.

3. Anderson, T. M. et. al., "Report on Small Break Accidents for Westinghouse NSSS Systems," WCAP-9600, Westinghouse Proprietary Class 2, June 1979.

TABLE 5.4-1 (SHEET 1 OF 2)

REACTOR COOLANT PUMP DESIGN PARAMETERS
MODEL 100A

Unit design pressure (psig)	2485	(a,c)
Unit design temperature (°F)		
Unit overall height (ft)		
Seal water injection (gal/min)		
Seal water return (gal/min)		
Component cooling water flow (gal/min) RCP and Motor		
Maximum continuous component coolant water inlet temperature (°F)		
<u>Pump</u>		
Design flow (gal/min)		
Developed head (ft)		
NPSH required (ft)		
Suction temperature, thermal design (°F)		
Pump discharge nozzle, inside diameter (in.)		
Pump suction nozzle, inside diameter (in.)		
Speed (rpm), nominal		
Water volume (ft^3)		
<u>Motor</u>		
Type	Drip-proof squirrel cage induction, with water/air coolers	
Power (hp)	8000	

TABLE 5.4-1 (SHEET 2 OF 2)

Frequency (Hz)	60
Insulation class	Class F thermalastic epoxy insulation
Pump moment of inertia, maximum (lb/ft^2)	[ ] (a,c)

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a. Design temperature of pressure retaining parts of the pump assembly exposed to the reactor coolant and injection water on the high pressure side of the controlled leakage seal is that temperature determined for the parts for a reactor coolant loop temperature of 650°F.

b. Includes reactor coolant in the casing and seal injection water in the thermal barrier region.

TABLE 5.4-2

REACTOR COOLANT PUMP QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]
Castings	Yes		Yes	
Forgings				
Main shaft (RCP, not motor)		Yes	Yes	
Main bolts		Yes		Yes
Plate				
Flywheel		Yes	Yes[b]	Yes[b]
Weldments				
Circumferential	Yes		Yes	
Instrument connections			Yes	

a. RT - Radiographic.
UT - Ultrasonic.
PT - Dye penetrant.
MT - Magnetic particle.

b. Of machined bores keyways and drilled holes within 4" of the bore (either PT or MT).

TABLE 5.4-3

STEAM GENERATOR DESIGN PARAMETERS

(a,c,e,f)

Design pressure, reactor coolant side (psia)	
Design pressure, steam side (psia)	
Design pressure, primary to secondary (psi)	
Design temperature, reactor coolant side (°F)	
Design temperature, steam side (°F)	
Total heat transfer surface area (ft^2)	
Maximum moisture carryover (weight percent)	
Overall height (in.)	
Number of U-tubes	
U-tube nominal diameter (in.)	
Tube wall nominal thickness (in.)	
Number of manways	
Inside diameter of manways (in.)	
Number of handholes	
Design fouling factor (ft^2-hr-°F/Btu)	
Steam flow (lbm/hr)	

TABLE 5.4-4 (Sheet 1 of 2)

STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]	ET[a]
Tubesheet					
Forging		yes		yes	
Cladding		yes[b]	yes		
Channel Head (if fabricated)					
Fabrication	yes[c]	yes[d]		yes	
Cladding			yes		
Secondary Shell and Head					
Plates		yes			
Tubes		yes			yes
Nozzles (Forgings)		yes		yes	
Weldments					
Shell, longitudinal	yes			yes	
Shell, circumferential	yes			yes	
Cladding (channel head-tube sheet joint)			yes		
Primary nozzles to fabrication head	yes			yes	
Manways to fabrication head	yes			yes	
Steam and feedwater nozzles to shell	yes			yes	

TABLE 5.4-4 (Sheet 2 of 2)

STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]	ET[a]
Support brackets				yes	
Tube to tubesheet			yes		
Instrument connections				yes	
Temporary attachments after removal				yes	
After hydrostatic test (all major pressure boundary welds and complete cast channel head - where accessible)				yes	
Nozzle safe ends (if weld deposit)	yes		yes		

Notes:

(a) RT - Radiographic
    UT - Ultrasonic
    PT - Dye penetrant
    MT - Magnetic particle
    ET - Eddy current
(b) Flat surfaces only
(c) Weld deposit
(d) Base material only

TABLE 5.4-5

REACTOR COOLANT PIPING DESIGN PARAMETERS

	(a,c)
Reactor Inlet Piping Inside diameter (ID) (in.)	
Reactor Inlet Piping Nominal wall thickness (in.)	
Reactor Outlet Piping and Coolant Pump Suction Piping ID (in.)	
Reactor Outlet Piping and Coolant Pump Suction Piping Nominal wall thickness (in.)	
Pressurizer Surge Line Piping Nominal pipe size (in.)	
Pressurizer Surge Line Piping Nominal wall thickness (in.)	
Nominal Water Volume, all four loops including surge line (ft^3)	
RCL Piping Design/operating pressure (psig) Design temperature (°F)	
Pressurizer Surge Line Design pressure (psig) Design temperature (°F)	
Pressurizer Safety Valve Inlet Line Design pressure (psig) Design temperature (°F)	
Pressurizer Power-Operated Relief Valve Inlet Line Design pressure (psig) Design temperature (°F)	

* At reactor coolant pump discharge - point of highest pressure in loop piping.

TABLE 5.4-6

REACTOR COOLANT PIPING QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]
Fittings and Pipe (Castings)	Yes		Yes
Fittings and Pipe (Forgings)		Yes	Yes
Weldments			
Circumferential	Yes		Yes
Nozzle to runpipe (except no RT for nozzles less than 6 in.)	Yes		Yes
Instrument connections			Yes
Castings	Yes		Yes (after finishing)
Forgings			Yes (after finishing)

(a) RT - Radiographic; UT - Ultrasonic; PT - Dye penetrant.

TABLE 5.4-7

PRESSURIZER DESIGN PARAMETERS

	(a,c)
Design pressure (psig)	
Design temperature (°F)	
Internal volume (ft^3)	
Nominal conditions at 100-percent rated load	
Steam volume (ft^3)	
Water volume (ft^3)	

TABLE 5.4-8

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	Psig (a,c)
Hydrostatic test pressure	
Design pressure	
Safety valves (begin to open)	
High pressure reactor trip	
High pressure alarm	
Power-operated relief valves	
Pressurizer spray valves (full open)	
Pressurizer spray valves (begin to open)	
Proportional heaters (begin to operate)	
Normal operating pressure	
Proportional heater (full operation)	
Backup heaters on	
Low pressure alarm	
PORV block valve setpoint	
Low pressure reactor trip	

(a) At [ ] psig, a pressure signal initiates actuation (opening) of these valves. Remote manual control is also provided. (a,c)

(b) At [ ] psig, a pressure signal initiates actuation (opening) of these valves. Remote manual control is also provided. (a,c)

TABLE 5.4-9

PRESSURIZER RELIEF TANK DESIGN PARAMETERS

	(a,c)
Design pressure (psig)	
Normal operating pressure (psig)	
Final operating pressure (psig)	
Rupture disc release pressure (psig)	
Nominal	
Range	
Normal water volume (ft^3)	
Normal gas volume (ft^3)	
Design temperature (°F)	
Initial operating water temperature (°F)[a]	
Final operating water temperature (°F)[a]	
Total rupture disc relief capacity at 100 psig (lb/h)	
Cooling time required following maximum discharge, approximately (h)	
Spray feed and bleed	
Utilizing RCDT heat exchanger	

(a) For the design basis pressurizer steam discharge.

TABLE 5.4-10

PRESSURIZER RELIEF DISCHARGE SYSTEM
NONDESTRUCTIVE TESTING PROGRAM

Components	Radiographic	Ultrasonic	Dye Penetrant
Fittings and pipe (castings)	Yes		Yes
Fittings and pipe (forgings)		Yes	Yes
Weldments			
Circumferential	Yes		Yes
Nozzle to runpipe (except no radiographic for nozzles less than 4 in.)	Yes		Yes
Instrument connections			Yes

TABLE 5.4-11

REACTOR COOLANT SYSTEM VALVE DESIGN PARAMETERS

Design pressure (psig)	2485	
Preoperational plant hydrotest (psig)	[	(a,c)
Design temperature (°F)		
Normal Operating Pressure (psig)	]	

TABLE 5.4-12

REACTOR COOLANT SYSTEM VALVES NONDESTRUCTIVE EXAMINATION PROGRAM

	RT[a]	UT[a]	PT[a]
Castings			
Larger than 4 in.	Yes		Yes
2 to 4 in.	Yes[b]		Yes
Forgings			
Larger than 4 in.	(c)	(c)	Yes
2 to 4 in.			Yes

a. RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant

b. Weld ends only

c. Either RT or Ut

TABLE 5.4-13

PRESSURIZER SAFETY AND RELIEF VALVES

DESIGN PARAMETERS

Pressurizer safety valves		
Number	3	(a,c)
Maximum relieving capacity per valve, ASME flowrate (lb/hr), at 3% accumulation		
Set pressure (psig)		
Design temperature (°F)		
Fluid		
Backpressure		
Normal (psig)		
Expected maximum during discharge (psig)		
Environmental conditions		
Ambient temperature (°F)		
Relative humidity (percent)		
Pressurizer power-operated relief valves		
Number		
Design pressure (psig)		
Design temperature (°F)		
Saturated steam-relieving capacity at 2385 psig, per valve (lb/hr)		
Saturated water-relieving capacity at 2485 psig, per valve (gal/min)		

TABLE 5.4-14

REACTOR VESSEL HEAD VENT SYSTEM EQUIPMENT

DESIGN PARAMETERS

Valves

Number (includes two manual valves)	6	(a,c)
Design pressure, psig		
Design temperatures, °F		

Piping

Vent line, nominal diameter, in.	
Design pressure, psig	
Design temperature, °F	
Maximum operating temperature	

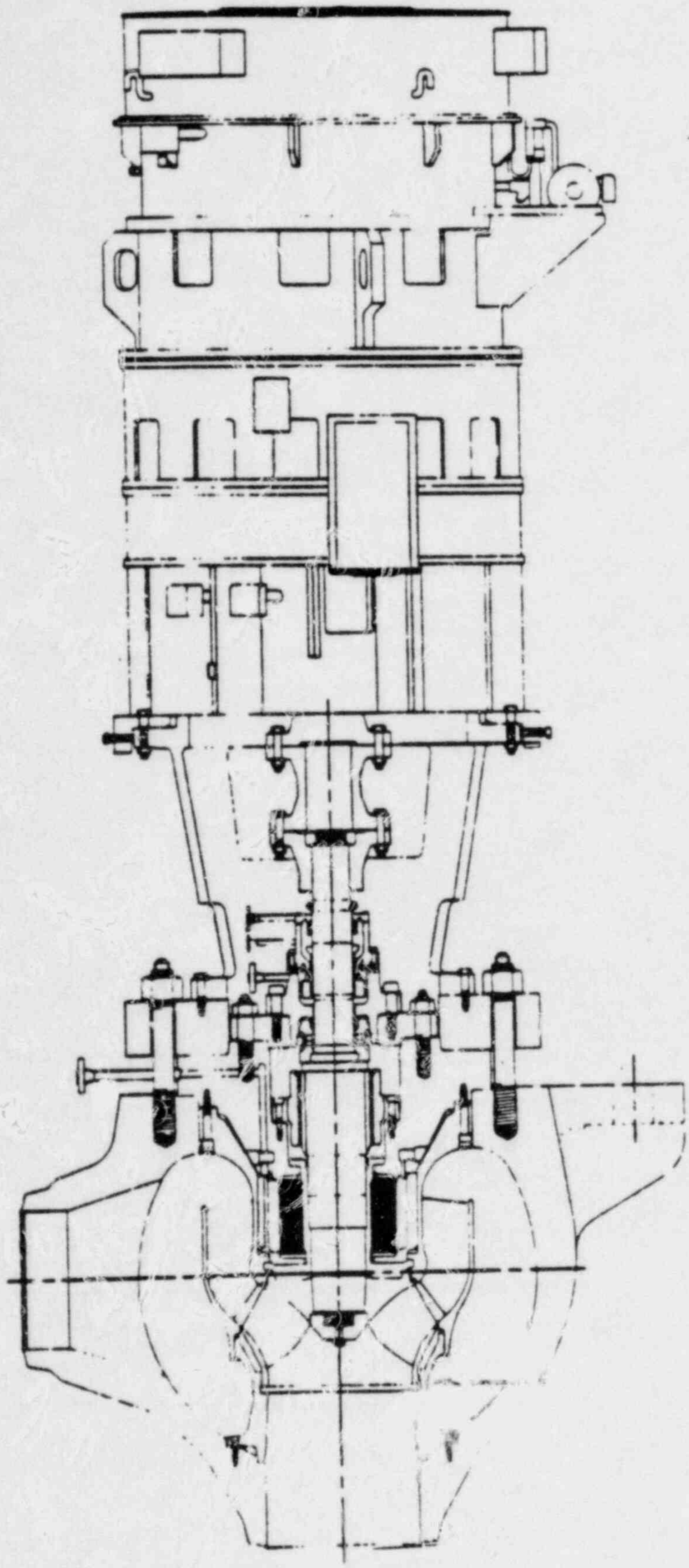

FIGURE 5.4-1 MODEL 100A REACTOR COOLANT PUMP

TOTAL HEAD (FT)

FLOW (THOUSANDS OF GPM)

FIGURE 5.4-2 REACTOR COOLANT PUMP CURVE

FIGURE 5.4-3 STEAM GENERATOR

(a,c)

FIGURE 5.4-4 STRUCTURAL BROACH SUPPORT CONFIGURATION

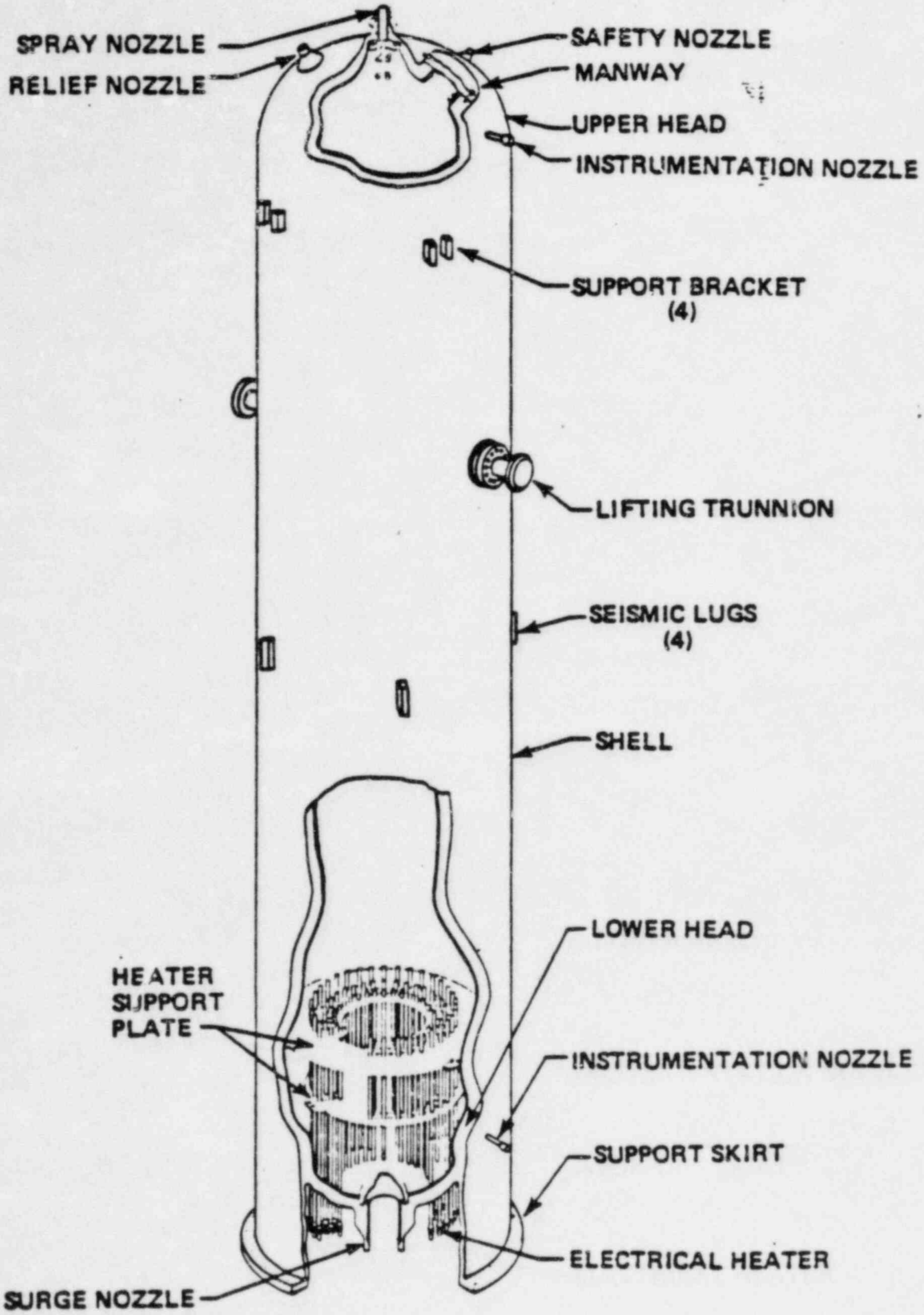

FIGURE 5.4-5 PRESSURIZER

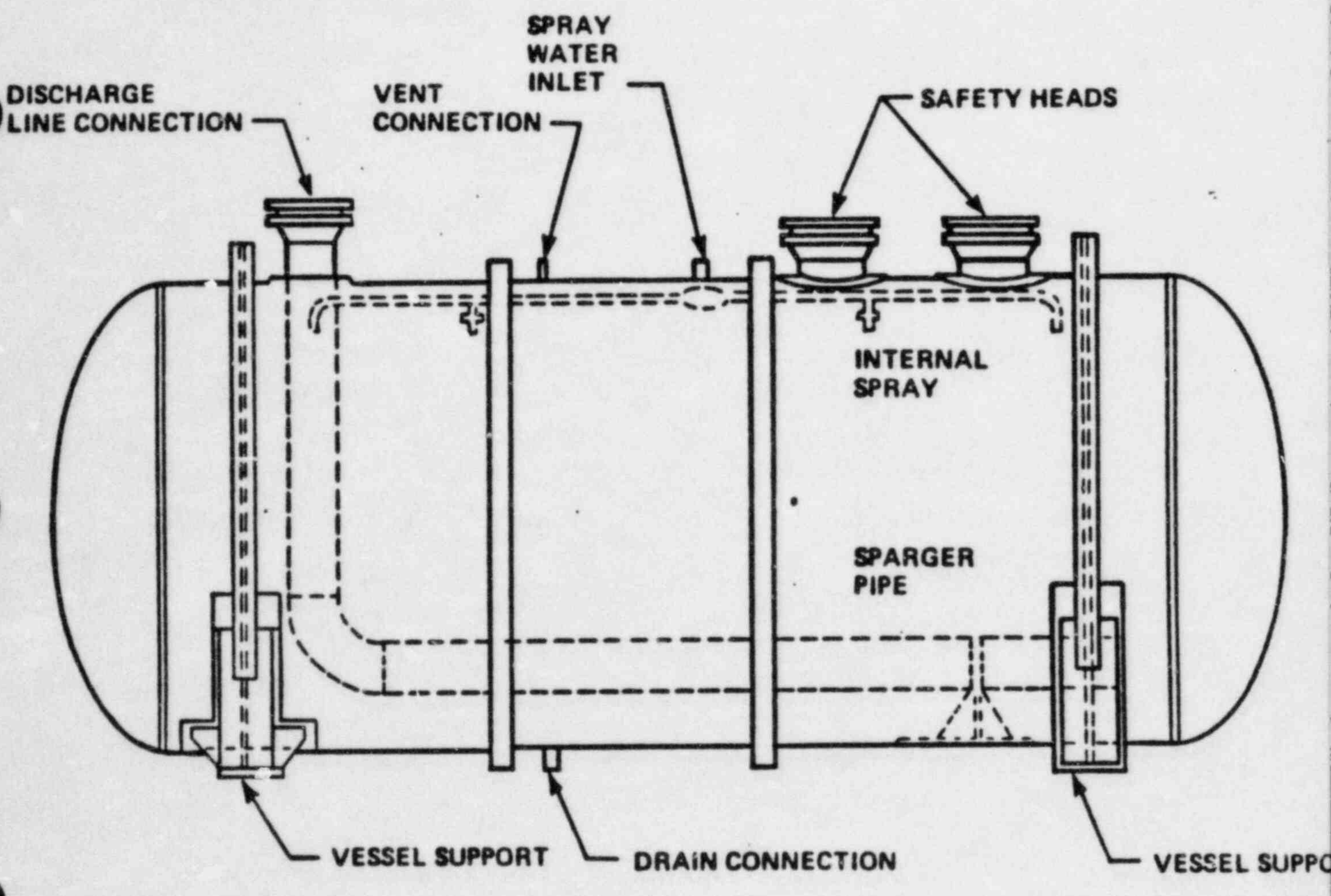

FIGURE 5.4-6 PRESSURIZER RELIEF TANK

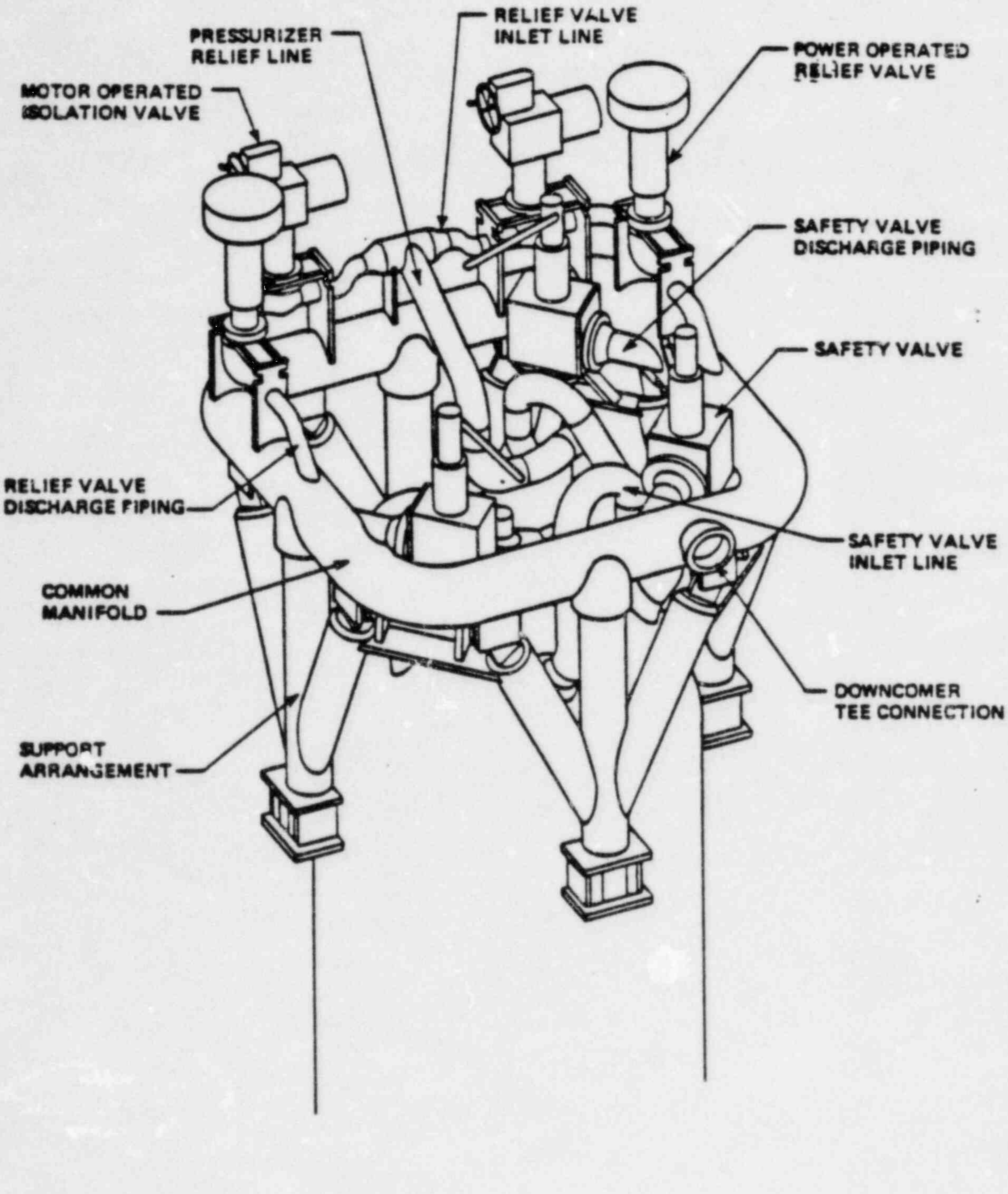

FIGURE 5.4-7 PRESSURIZER SAFETY AND RELIEF VALVE PIPING
AND SUPPORT ARRANGEMENT (TYPICAL)

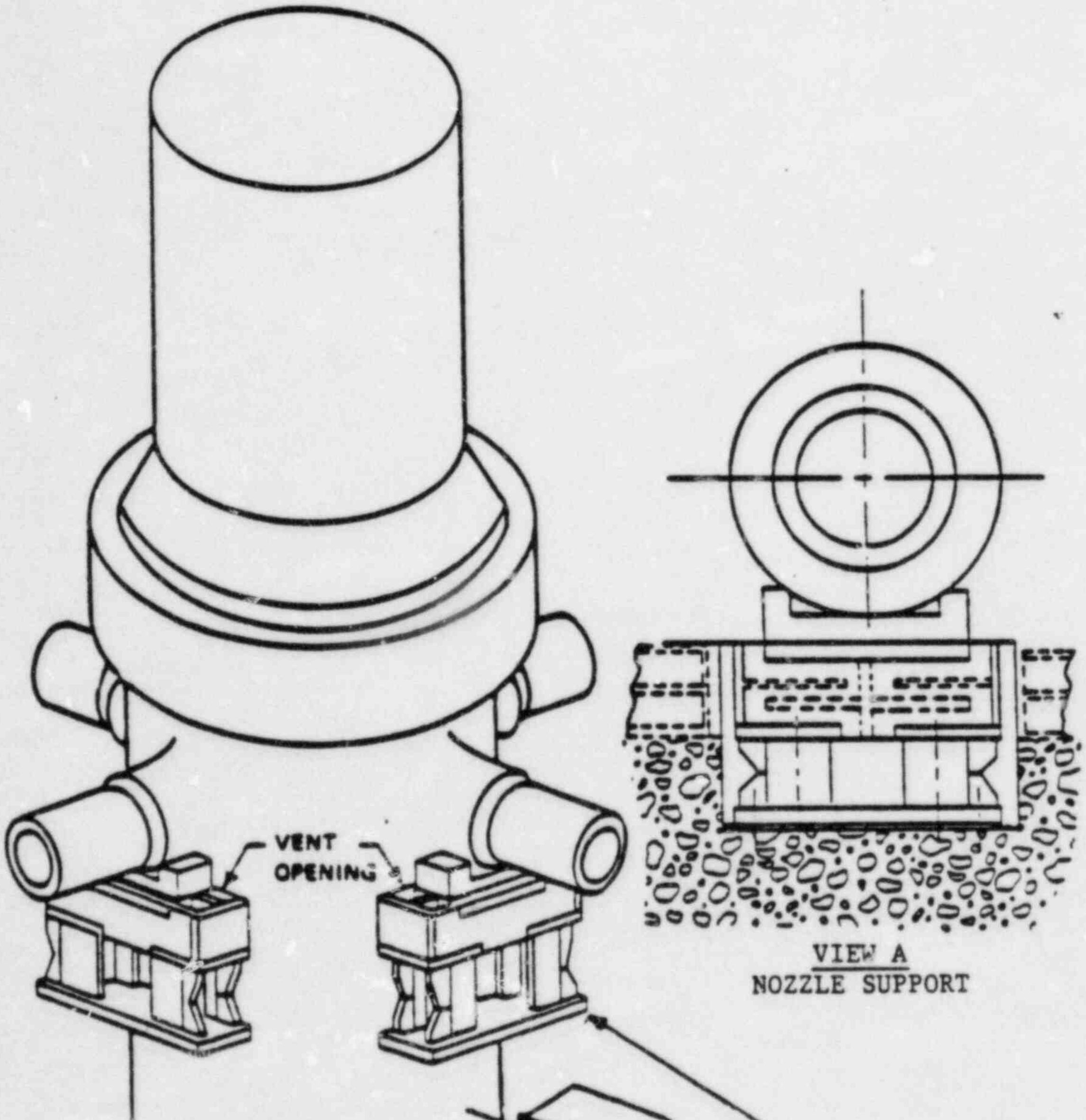

FIGURE 5.4-8 REACTOR VESSEL SUPPORTS

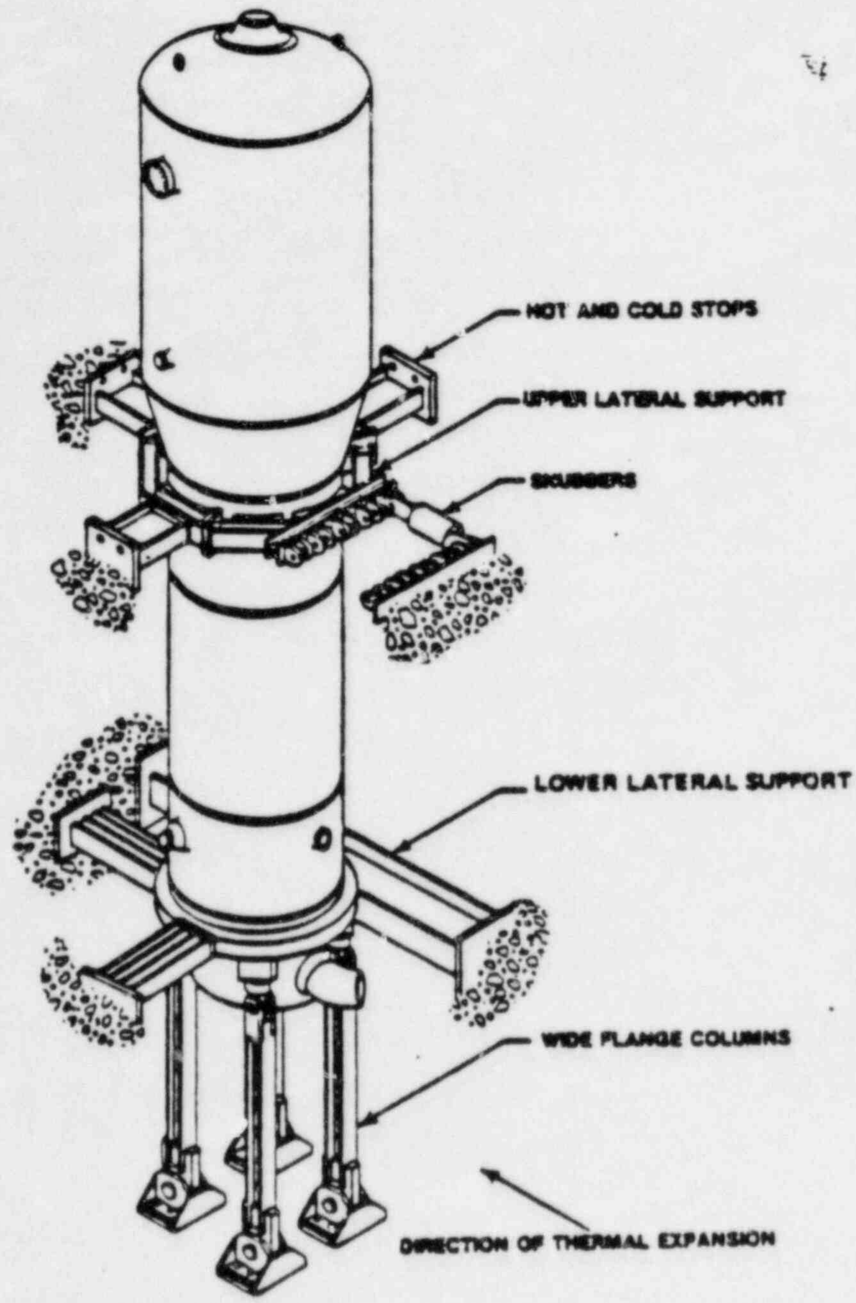

FIGURE 5.4-9 STEAM GENERATOR SUPPORTS

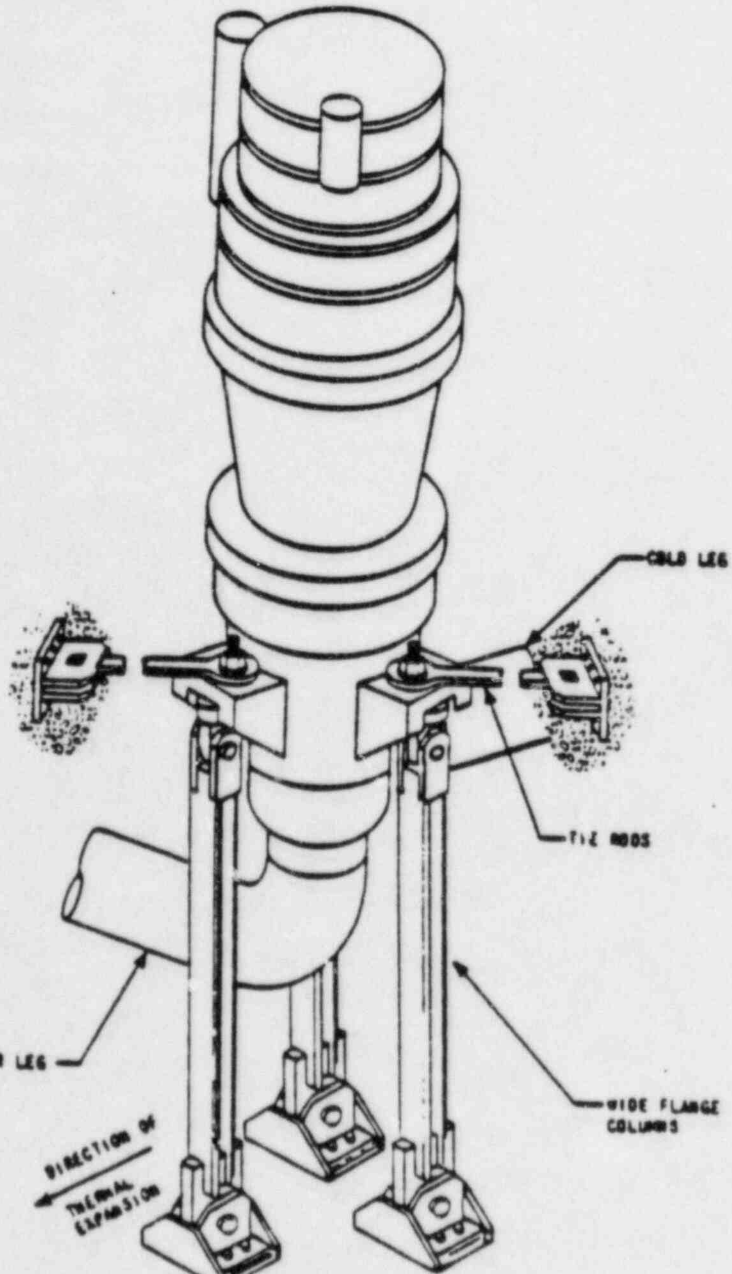

FIGURE 5.4-10 REACTOR COOLANT PUMP SUPPORTS

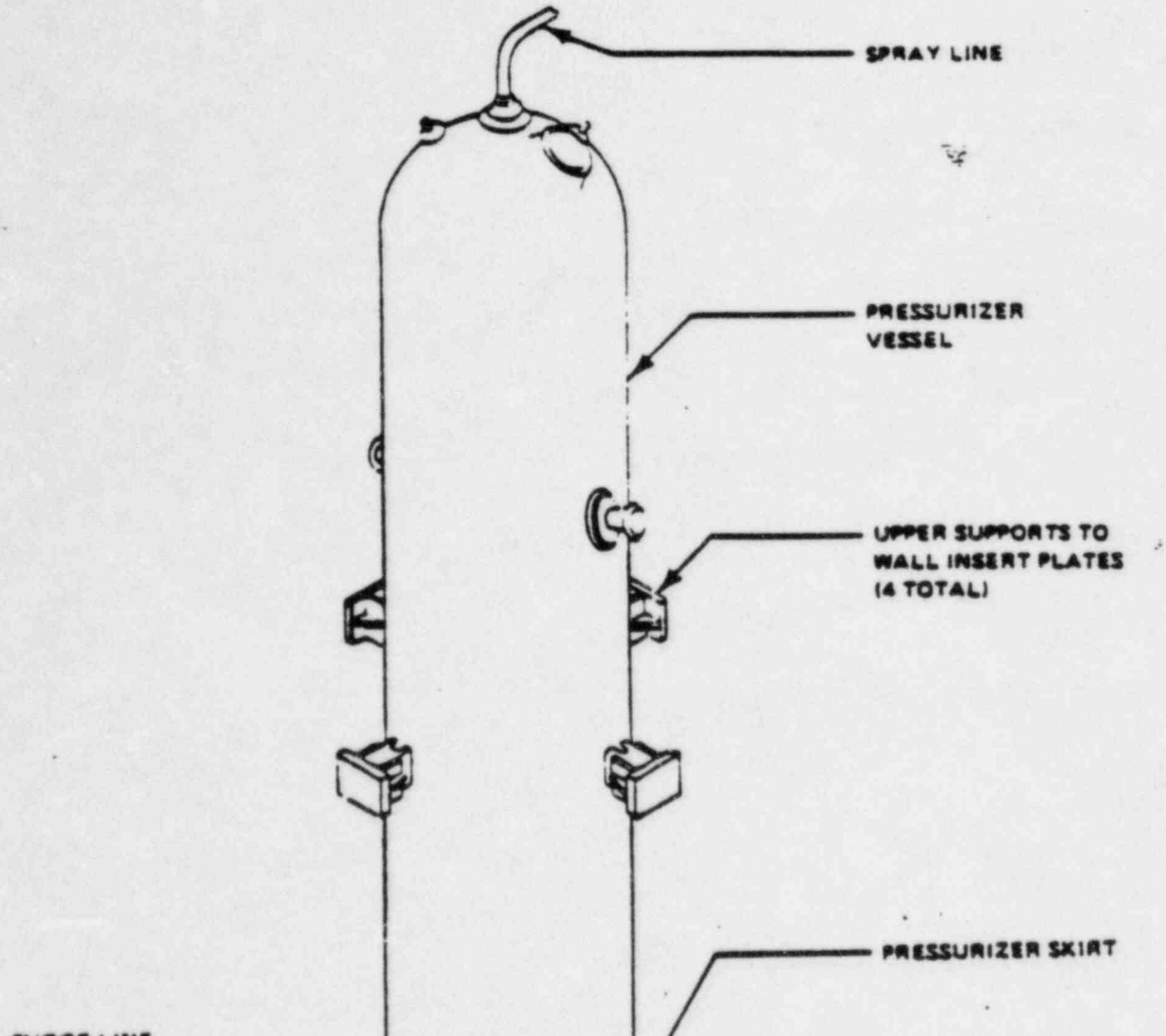

FIGURE 5.4-11 PRESSURIZER SUPPORTS