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5.0 REACTOR COOLANT SYSTEM

5.1 SUMMARY DESCRIPTION

The Reactor Coolant System (RCS) shown on Figures 5.1-1 through 5.1-4 consists of four similar heat transfer loops connected in parallel to the reactor pressure vessel (RPV). Each loop contains a reactor coolant pump (RCP), steam generator (SG) and associated piping and valves. In addition, the system includes a pressurizer, a pressurizer relief tank (PRT), interconnecting piping, and instrumentation necessary of operational control. All the system equipment, except for the three wide-range pressure transmitters and the Containment isolation and process-actuated valves located in the lines connected to the PRT, are located in the Containment Building.

During operation, the RCS transfers the heat generated in the core to the SGs where steam is produced to drive the turbine generator (TG). 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 in chemical shim control.

The reactor coolant pressure boundary (RCPB) 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 is also serviced by auxiliary systems (Chemical and Volume Control System [CVCS], Residual Heat Removal System [RHRS] and Safety Injection System [SIS]) interconnected with the reactor coolant piping.

RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium 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 (PORVs) are connected to the pressurizer and discharge to the PRT, where the steam is condensed and cooled by mixing with water.

The extent of the RCS is defined as:

1. Reactor vessel including control rod drive mechanism (CRDM) housings

2. Reactor coolant side of the SG

3. RCPs

4. Pressurizer attached to one of the reactor coolant loops (RCL)

5. PRT

6. Safety and relief valves

7. Interconnecting piping, valves, and fittings between the principal components listed above

8. Piping, fittings, and valves leading to connecting auxiliary or support systems up to and including the second isolation valve (from the high pressure side) on each line

Reactor Coolant System Components

Reactor Vessel

The reactor vessel is cylindrical, with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core supporting 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.

The reactor vessel is carbon steel with weld-deposited austenitic stainless steel cladding on all surfaces exposed to the reactor coolant.

Steam Generators

The SGs 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 hemispherical bottom head of the SG. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

Reactor Coolant Pumps

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

Piping

The RCL is specified in sizes consistent with system requirements.

The hot leg inside diameter is 29 in. and the inside diameter of the cold leg return line to the reactor vessel is 27-1/2 inches. The mechanical stress improvement process (MSIP) has been applied to the hot and cold leg piping at the reactor vessel nozzle safe end dissimilar metal welds. A discussion of MSIP is provided in Section 3.6.2.1. The inside diameter of the piping between the SG and the pump suction is increased to 31 in. to reduce pressure drop and improve flow conditions to the pump suction.

Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief, and safety valve connections are located in the top head of the vessel.

Pressurizer Relief Tank

The PRT is a horizontal, cylindrical vessel with elliptical dished heads. Steam from the pressurizer safety and relief valves is discharged into the PRT through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature.

Safety and Relief Valves

The pressurizer safety valves are of the totally enclosed pop type. The valves are spring-loaded, self-activated with back-pressure compensation. The power-operated relief valves limit system pressure for large power mismatch. They are operated automatically or by remote manual control. Remotely operated valves are provided to isolate the inlet to the power-operated relief valves if excessive leakage occurs. Position indication lights are provided in the control room for these valves.

Reactor Coolant System Performance Characteristics

Tabulations of important design and performance characteristics of the RCS are provided in Table 5.1-1.

Reactor Coolant Flow

The reactor coolant flow, a major parameter in the design of the system and its components, is established with a detailed design procedure supported by operating plant performance data, by pump model tests and analyses, an by pressure drop tests and analyses of the reactor vessel and fuel assemblies. Data from all operating plants have indicated that the actual flow has been well above the flow specified for the thermal design of the plant. By applying the design procedure described below, it is possible to specify the expected operating flow with reasonable accuracy.

Three reactor coolant flow rates are identified for the various plant design considerations. The definitions of these flows are presented in the following paragraphs.

Best Estimate Flow

The best estimate flow is the most likely value for the actual plant operating condition. This flow is based on the best estimate of the reactor vessel, SG and piping flow resistance, and on the best estimate of the RCP head-flow capacity, with no uncertainties assigned to either the system flow resistance or the pump head. System pressure drops, based on best estimate flow, are presented in Table 5.1.-1. Although the best estimate flow is the most likely value to be expected in operation, more conservative flow rates are applied in the thermal and mechanical designs.

Thermal Design Flow

Thermal design flow is the basis for the reactor core thermal performance, the SG thermal performance, and the nominal plant parameters used throughout the design. To provide the required margin, the thermal design flow accounts for the uncertainties in reactor vessel, SG and piping flow resistances, RCP head, and the methods used to measure flow rate. The thermal design flow is confirmed when the plant is placed in operation. Tabulations of important design and performance characteristics of the RCS provided in Table 5.1-1 are based on the thermal design flow.

Mechanical Design Flow

Mechanical design flow is the conservatively high flow used in the mechanical design of the reactor vessel internals and fuel assemblies. To assure that a conservatively high flow is specified, the mechanical design flow is based on a reduced system resistance and on the maximum uncertainty of pump head capability.

Pump overspeed, due to TG overspeed of 20 percent, results in a peak reactor coolant flow of 120 percent of the mechanical design flow. The overspeed condition is applicable only to operating conditions when the reactor and turbine generator are at power.

Interrelated Performance and Safety Functions

The interrelated performance and safety functions of the RCS and its major components are listed below:

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

2. The system provides sufficient heat transfer capability to transfer the heat produced during the subsequent phase of plant cooldown and cold shutdown to the RHRS.

3. The system 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.

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

5. The system maintains the homogeneity of soluble neutron absorber concentration and rate of change of coolant temperature such that uncontrolled reactivity changes do not occur.

6. The reactor vessel is an integral part of the RCPB and is capable of accommodating the temperatures and pressures associated with the operational transients. The reactor vessel functions to support the reactor core and CRDMs.

7. The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, reactor coolant volume changes are accommodated in the pressurizer via the surge line to the pressurizer.

8. The RCPs supply the coolant flow necessary to remove heat from the reactor core and transfer it to the SGs.

9. The SGs provide high quality steam to the turbine. The tube and tubesheet boundaries are designed to prevent or control to acceptable levels the transfer of activity generated within the core to the secondary system.

10. The RCS piping serves as a boundary for containing the coolant under operating temperature and pressure conditions and for limiting leakage (and activity release) to the Containment atmosphere. The RCS piping contains demineralized borated water which is circulated at the flow rate and temperature consistent with achieving the reactor core thermal and hydraulic performance.

5.1.1 Schematic Flow Diagram

The RCS is shown schematically on Figure 5.1-5. Included on this figure is a tabulation of principal pressures, temperatures, and the flow rate of the system under normal steady state full power operating conditions. These parameters are based on the original best estimate flow at the pump discharge and are for information only. The RCS volume range reflects normal operational variations and reductions due to steam generator tube plugging. (See Table 5.1-1.)

5.1.2 Piping and Instrumentation Diagram

Piping and instrumentation diagrams of the RCS are shown on Figures 5.1-1 through 5.1-4. The diagrams show 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 Elevation Drawings

Elevation drawings of the RCS are provided on Figures 1.2-18 and 1.2-19.

TABLE 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

Plant Design Life, years	40
Nominal Operating Pressure, psig	2,235
Total System Volume, Including Pressurizer and Surge Line at No Load Conditions[1], ft^3	14,970
System Liquid Volume, Including Pressurizer Water at No Load Conditions[1], ft^3	13,346
Pressurizer Spray Rate, Maximum, gal/min	1,050
Pressurizer Heater Capacity, kW	2,100 [6]
Pressurizer Relief Tank Volume, ft^3	2,100
<u>System Thermal and Hydraulic Data</u>[1]	
NSSS Power, MWt	3,874
Reactor Power, MWt	3,853
Thermal Design Flow, gal/min	
Active Loop	98,000
Reactor Vessel	392,000
Total Reactor Flow, 10^6 lb/hr	147.39 – 145.21
Temperatures,°F	Design Range
Reactor Vessel Outlet	615.5 – 624.8
Reactor Vessel Inlet	549.8 – 560.3
Steam Generator Outlet (RCS)	549.4 – 559.9
Steam Generator Steam	539.2 – 551.4
Feedwater	390.0 – 441.8
Steam Pressure, psia	956 – 1,057
Total Steam Flow, 10^6 lb/hr	15.99 – 17.20[4]

TABLE 5.1-1 (Continued)

SYSTEM DESIGN AND OPERATING PARAMETERS

System Thermal and Hydraulic Data[1] (continued)

Best Estimate Flow, gal/min		
Active Loop	107,900[5]	107,600 [7]
Reactor Vessel	431,600[5]	430,400[7]
Mechanical Design Flow, gal/min		
Active Loop	110,000	110,000
Reactor Vessel	440,000	440,000
System Pressure Drops[1]		
Reactor Vessel ΔP, psi	61.7[5]	61.08[2,7] 59.14[3,7]
Steam Generator ΔP, psi	34.2[2,5] 38.0[3,5]	34.05[2,7] 37.82[3,7]
Hot Leg Piping ΔP, psi	1.5[5]	1.53[2,7] 1.44[3,7]
Pump Suction Piping ΔP, psi	3.9[5]	3.84[2,7] 3.61[3,7]
Cold Leg Piping ΔP, psi	2.3[5]	2.3[2,7] 2.16[3,7]
Pump Head, ft	319[5]	316.77[2,7] 325.92[3,7]

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Notes:
(1) Represent nominal design values. Operating values will vary as a result of Steam Generator tube plugging, changes in RCS flow rates and RCS full load operating temperatures. Best estimate flow and system pressure drops were calculated for Unit 1 at 0% SGTP unless otherwise noted considering the application of the Unit 1 Mechanical Stress Improvement Project (MSIP) to the Reactor Vessel inlet and outlet nozzles.
(2) At 0% tube plugging for best estimate flow.
(3) At 10% tube plugging and best estimate flow.
(4) Range affected by the feedwater temperature range.
(5) Representative of Unit 1
(6) Represent nominal design values.
(7) Best estimate flow and system pressure drops were calculated for Unit 2 at 0% SGTP unless otherwise noted prior to application of MSIP.

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NOTES TO FIGURE 5.1-5

STEADY-STATE FULL POWER OPERATION

Location	Fluid	Pressure psia	Temperature ºF	Flow Lbm/sec	Volume ft^3
1	RC	2261.3	612.0	11,226.7	-
2	RC	2254.6	612.0	11,226.7	-
3	RC	2226.0	552.3	11,226.7	-
4	RC	2222.6	552.3	11,223.2	-
5	RC	2326.4	552.6	11,223.9	-
6	RC	2324.7	552.6	11,226.7	-
10-15	RC	SEE LOOP #1 SPECIFICATIONS			
19-24	RC	SEE LOOP #1 SPECIFICATIONS			-
28-33	RC	SEE LOOP #1 SPECIFICATIONS			-
37	RC	2325.4	552.6	0	-
38	RC	2325.4	552.6	0	-
39	RC	2249.7	552.6	0	-
40	Steam	2249.7	652.7	-	923
41	RC	2250.3	652.7	-	1177
42	RC	2256.6	652.7	0	-
43	RC	2258.6	632.3	0	-
44	Steam	2249.7	652.7	0	-
45	RC	2249.7	<652.7	0	MINIMIZE
46	N_2	17.7	120	0	-
47	RC	2249.7	<652.7	0	MINIMIZE
48	N_2	17.7	120	0	-
49	RC	17.7	120	0	-
50	PRT Vapor	17.7	95	-	582
51	PRT Water	17.7	95	-	1518
52	RC	2261.3	552.6	-	-

HISTORICAL INFORMATION

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section presents a discussion of the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime. In this context, the RCPB is as defined in 10CFR50.2. In that definition, the RCPB extends to the outermost Containment isolation valve in system piping that penetrates the Containment and is connected to the Reactor Coolant System (RCS). Since other sections of this report describe the components of the auxiliary fluid systems in detail, the discussions in this section will be limited to the components of the RCS, as defined in Section 5.1, unless otherwise noted.

For additional information on the RCS and for components that are part of the RCPB (as defined in 10CFR50.2) but are not described in this section, refer to the following sections:

Section 6.3	-	For discussions of the RCPB components that are part of the Emergency Core Cooling System (ECCS).
Section 9.3.4	-	For discussions of the RCPB components that are part of the Chemical and Volume Control System (CVCS).
Section 3.9.1	-	For discussions of the design loadings, stress limits, and analyses applied to the RCS and American Society of Mechanical Engineers (ASME) Code Class 1 components.
Section 3.9.3	-	For discussions of the design loadings, stress limits and analyses applied to ASME Code Class 2 and 3 components.

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 10CFR50.2.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10CFR50.55a. RCS components are designed and fabricated in accordance with the rules of 10CFR50, Appendix B, and Section 50.55a, “Codes and Standards”. The actual addenda of the ASME Code applied in the design of each component is listed in Table 5.2-1.

5.2.1.2 Applicable Code Cases. Applicable Code Cases are discussed in Section 3.12 as part of Regulatory Guides (RGs) 1.84 and 1.85.

5.2.2 Overpressure Protection

RCS overpressure protection is accomplished by the utilization of safety valves along with the Reactor Protection System (RPS) and associated equipment. Combinations of these systems provide compliance with the overpressure requirements of the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Paragraph NB-7300, for pressurized water reactor (PWR) systems.

Auxiliary or emergency systems connected to the RCS are not utilized for prevention of RCS overpressurization protection.

5.2.2.1 Design Bases. Overpressure protection is provided for the RCS by the pressurizer safety valves, which discharge to the pressurizer relief tank (PRT) by a common header. The transient on which the design requirements are set for the primary system overpressure protection is a complete loss of steam flow to the turbine with credit taken for steam generator (SG) safety valve operation. However, for the sizing of the pressurizer safety valves, no credit is taken for reactor trip, main feedwater flow, or for operation of the following:

1. Pressurizer power-operated relief valves (PORVs)

2. Steam line PORVs

3. Steam Dump System

4. Rod Control System

5. Pressurizer Level Control System

6. Pressurizer spray valve

For this transient, the peak RCS and peak Steam System pressures must be limited to 110 percent of their respective design values.

Assumptions for the overpressure analysis include: (1) the plant is operating at the power level corresponding to the engineered safeguards design rating; and (2) the RCS average temperature and pressure are at their maximum values. There are the most limiting assumptions with respect to system overpressure.

Overpressure protection for the Steam System is provided by SG safety valves. The Steam System safety valve capacity is based on providing enough relief to remove 105 percent of the engineered safeguards design steam flow. This must be done by limiting the maximum Steam System pressure to less than 110 percent of the SG 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 Section 5.4.11.

SG blowdown systems for the balance of plant are discussed in Chapter 10.

5.2.2.2 Design Evaluation. The relief capacities of the pressurizer and SG safety valves are determined from the postulated overpressure transient conditions in conjunction with the action of the RPS.

An Overpressure Protection Report (Ref. 5.2-8), specifically for South Texas Project Electric Generating Station (STPEGS) Units 1 and 2 was prepared in accordance with Article 7300 Section III of the ASME Code. This report addressees the following events which envelope credible events which could lead to overpressurization of the RCS if adequate overpressure protection were not provided:

1. Loss of Electrical Load and/or Turbine Trip

2. Uncontrolled Rod Withdrawal at Power

3. Loss of Reactor Coolant Flow

4. Loss of Normal Feedwater

5. LOOP to the Station Auxiliaries

Description of these transients, assumptions made, methods of analysis, and conclusions are presented in Chapter 15.

Reference 5.2-1 discusses the generic methodology for sizing the pressurizer safety valves.

The description of the analytical model used in the analysis of the Overpressure Protection System and the basis for its validity is discussed in Reference 5.2-2.

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

5.2.2.3 Piping and Instrumentation Diagrams. Overpressure protections for the RCS is provided by pressurizer safety valves, shown on Figure 5.1-3. These discharge to the PRT through a common header.

The Steam System safety valves are discussed in Chapter 10 and are shown on Figure 10.3-1.

5.2.2.4 Equipment and Component Description. The operation, significant design parameters, number and types of operating cycles, and environmental qualification of the pressurizer safety valves are discussed in Section 5.4.13.

A discussion of the equipment and components of the Steam System Overpressure System is in Chapter 10.

5.2.2.5 Mounting of Pressure Relief Devices. Westinghouse provides mounting brackets on the pressurizer which are used to support the pressurizer safety valves. Westinghouse is responsible for the design and manufacture of the supports for these valves, and for determining reactions on the pressurizer mounting brackets. Mounting of pressure relief devices is discussed in Section 3.9.3.3.

5.2.2.6 Applicable Codes and Classifications. The requirements of ASME Code, Section III, NB-7300 (“Overpressure Protection Report”) and NC-7300 (“Overpressure Protection Analysis”), are followed and complied with for PWR systems.

Piping, valves, and associated equipment used for overpressure protection are classified in accordance with American National Standards Institute (ANSI) N18.2, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants”. These safety class designations are delineated in Section 3.2 and shown on Figures 5.1-1 through 5.1-4.

For further information, refer to Section 3.9, “Mechanical Systems and Components”.

5.2.2.7 Material Specifications. Refer to Section 5.2.3, "Reactor Coolant Pressure Boundary Material".

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 of process instrumentation associated with the system, refer to Section 7.2.2.3.3.

5.2.2.9 System Reliability. The reliability of the pressure-relieving device is discussed in Section 4 of Reference 5.2-1.

5.2.2.10 Testing and Inspection. Testing and inspection of the overpressure protection components are discussed in Section 5.4.13.4 and Chapter 14.

5.2.2.11 RCS Pressure Control During Low Temperature Operation. Administrative procedures are structured to aid the operator in controlling RCS pressure during low temperature operation. However to provide a backup to the operator, an automatic system is provided to maintain pressures within allowable limits.

Analyses have shown that one pressurizer PORV is sufficient to prevent violation of these limits due to anticipated mass and heat input transients. Redundant protection against a low temperature overpressure event is provided through the use of two pressurizer PORVs to mitigate potential pressure transients.

The automatic system is required only during low temperature water solid operation when it is manually armed and automatically actuated.

As described in Section 5.4.13, the STPEGS PORVs are safety-related and Class 1E powered. They are designed in accordance with the ASME Code and are qualified via the Westinghouse pump and valve operability program which is described in Section 3.9.3.2.1, are seismically qualified as described in Section 3.10N, and environmentally qualified as described in Section 3.11N.

Offsite power is not required for the system to function. The actuation logic in the system is testable on line. The PORVs are not exercised with the reactor at power, however, they are capable of being tested as required by the ASME Code and the STPEGS Technical Specifications. Section 7.6 provides details of the design of the PORV interlocks for low temperature operation.

5.2.2.11.1 System Operation: Each of the two pressurizer PORVs is supplied with actuation logic to ensure that a completely automatic and independent RCS pressure control back-up feature is provided for 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 Sections 5.4.7, 5.4.10, 5.4.13, 7.6, 7.7, and 9.3.4 for additional information on RCS pressure and inventory control during other modes of operation.

The basic function of the system logic is to continuously monitor RCS temperature and pressure conditions, with the logic manually armed whenever plant operation is at a temperature below 350°F. An auctioneered system temperature is continuously converted to an allowable pressure and then compared to the actual RCS pressure. The system logic will first annunciate a main control board alarm whenever the measure RCS pressure approaches the allowable pressure. Upon further increase in the RCS pressure, the system will provide an actuation signal to the PORVs when required, to prevent pressure-temperature conditions from exceeding the allowable limits.

See Section 7.6 for a further discussion of system logic.

5.2.2.11.2 Evaluation of Low Temperature Overpressure Transients: ASME Section III, Appendix G, established guidelines and limits for RCS pressure primarily for low temperature conditions (≤ 350°F).

Transients analyses were performed to determine the maximum pressure for the postulated (credible) worst case mass input and heat input events.

The mass input transient is divided into two parts for plant operation in Mode 4 (> 200°F) and Mode 5 (≤ 200°F). In Mode 4, the mass input transient assumes the combined maximum deliverable flow from one high-head safety injection (HHSI) pump, and one centrifugal charging pump (CCP) with letdown isolated and charging control valve fully open. In Mode 5, the mass input transient assumes the operation of one CCP with inadvertent isolation of letdown flow and charging control valve fully open.

The heat input analysis was performed for an inadvertent RCP start assuming that the RCS was water solid at the initiation of the event and that a 50°F mismatch existed between the RCS and the secondary side of the SGs. (At lower temperatures, the mass input case is the limiting transient condition).

Both heat input and mass input analyses took into account the single failure criteria and therefore, only one PORV was assumed to be available for pressure relief. The evaluation of the transient results concludes that the allowable limits will not be exceeded and therefore will not constitute an impairment to vessel integrity and plant safety.

5.2.2.11.3 Administrative Procedures: Although the system described in Section 5.2.2.11.1 is provided to maintain RCS pressure within allowable limits; administrative procedures minimize the potential for any transient that could actuate the overpressure relief system. The following discussion highlights these procedural controls.

Of primary importance is the basic method of operation of the plant. Normal plant operating procedures will specify that the use of a pressurizer steam bubble during periods of low pressure, low temperature operation is preferred. This steam bubble will dampen the plant's response to potential transient generating inputs, providing easier pressure control with slower response rates.

A steam bubble substantially reduces the severity of potential pressure transients, such as RCP induced heat input, and slows the rate of pressure rise for others. In conjunction with the alarms discussed in Section 7.6, 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, procedures will further highlight precautions that minimize the severity of, or the potential for, an overpressurization transient. The following precautions of measure are considered in developing operating procedures:

a. Whenever the plant is water solid and the reactor coolant pressure is being maintained by the low pressure letdown control valve, letdown flow normally bypasses the normal letdown orifices.

b. If all reactor coolant pumps have stopped for more than 5 minutes during plant heatup after filling and venting has been completed, and the reactor coolant temperature is greater than the charging and seal injection water temperature, a steam bubble will be formed in the pressurizer prior to restarting a RCP. 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 will accommodate the resultant expansion as the cold water is rapidly warmed.

c. If the RCPs are stopped and the RCS is being cooled down by the residual heat exchangers (HXs), a nonuniform temperature distribution may occur in the reactor coolant loops (RCLs). Prior to restarting an RCP, a steam bubble will be formed in the pressurizer or an acceptable temperature profile will be demonstrated.

d. During plant cooldown, all SGs will normally be connected to the steam header to assure a uniform cooldown of the RCL.

These special precautions backup the normal operational mode of maximizing periods of steam bubble operation so that cold overpressure transient prevention is continued during periods of transitional operations. These precautions do not apply to reactor coolant system hydrostatic testing.

The specific plant configurations of ECCS testing and alignment will also highlight procedural recommendations to prevent developing cold overpressurization transients. During these limited periods of plant operation, the following precautions/measures are considered in developing the procedures:

a. To preclude inadvertent ECCS actuation during heatup and cooldown, procedures required blocking the pressurizer pressure and low compensated steam line pressure SI signal actuation logic below the P-11 setpoint.

b. During further cooldown, closure and power lockout of the accumulator isolation valves will be performed at 1,000 psig. When the RCS temperature is reduced to or below 350°F, a maximum of one CCP and one HHSI pump is allowed operable by Technical Specifications. The low-head safety injection (LHSI) pump does not impact the Cold Overpressure Mitigation System (COMS) analysis because of the low shutoff head (approximately 315 psi).

c. The recommended procedure for periodic ECCS pump performance testing will be to test the pumps during normal power operation or at hot shutdown conditions. This precludes any potential for developing a cold overpressurization transient.

   Should cold shutdown testing of the pumps be desired, the test will be done when the vessel is open to atmosphere, again precluding overpressurization potential.

   If cold shutdown testing with the reactor vessel closed is necessary, the procedures require ECCS pump discharge valves to be closed.

d. Safety Injection System (SIS) circuitry testing, if done during cold shutdown, requires the nonoperating SI pumps that are not being tested to be power locked out to preclude cold overpressurization transients.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications. Material specifications used for the principal pressure-retaining applications in each component constituting the RCPB are listed in Table 5.2-2 for ASME Class 1 primary components and Table 5.2-3 for ASME Class 1 and 2 auxiliary components. These tables also include the unstabilized austenitic stainless steel material specifications used for components in systems required for reactor shutdown and for emergency core cooling.

The unstabilized austenitic stainless steel material for the reactor vessel internals which are 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.-5.

All of the materials utilized conform with the material specification requirements and include the special requirements of the ASME Code, Section III, plus addenda and code cases as are applicable and appropriate to meet Appendix B of 10CFR50. The listed specifications in Table 5.2-3 are representative of those materials utilized.

The welding materials used for joining the ferritic base materials of the RCPB conform to the requirements of ASME Code, Section II. They are tested and qualified to the requirements of ASME Code, Section III. In addition, the ferritic materials of the reactor vessel belt line are restricted to the following maximum limits of copper, phosphorous, and vanadium to reduce sensitivity to irradiation embrittlement in service.

Element	Base Metal (%)	Weld Metal (%)
Copper	0.10 (Ladle) 0.12 (Check)	0.10
Phosphorous	0.012 (Ladle) 0.017 (Check)	0.015
Vanadium	0.05 (Check)	0.05 (as residual)

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 tested and qualified according to the requirements of 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 tested and qualified to the requirements of ASME Code, Section III. The Mechanical Stress Improvement Process, MSIP®, has been performed on th RPV inlet and outlet nozzle safe end dissimilar metal welds to mitigate the potential formation of cracks from primary water stress corrosion cracking by placing compressive residual stresses on the inside surface of the weld region by compressing the pipe through controlled plastic deformation. CN-3284

All weld filler materials for RCPB piping field erection welds are in accordance with the applicable ASME Code Section II, SFA specification and ASME Code, Section III, Subsection NB-2400. The specific materials are:

- SFA-5.4, Type E308-16/E308L-16
- SFA-5.4, Type E316-16/E316L-16
- SFA-5.9, Type ER308/ER308L and ER316/ER316L
- SFA-5.11, Type ENiCrFe-7
- SFA-5.14, Type ERNiCr-7

5.2.3.2 Compatibility with Reactor Coolant.

5.2.3.2.1 Chemistry of Reactor Coolant: RCS chemistry specifications and guidelines recommended by the NSSS supplier for power operation are given in Table 5.2-4.

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 specifications.

The CVCS provides a means for adding chemicals to the RCS that control the pH of the coolant during pre-startup testing and subsequent operation, that scavenge oxygen from the coolant during heatup, and that control radiolysis reactions involving hydrogen, oxygen, and nitrogen during all power operations subsequent to startup. The parameters for chemical additives and reactor coolant impurities for power operation are shown in Table 5.2-4.

The pH control chemical specified is lithium hydroxide monohydrate, enriched in 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. The lithium-7 hydroxide is introduced into the RCS and the concentration of lithium-hydroxide is maintained in the range specified for pH control. If the concentration exceeds this range, the cation bed demineralizer is employed in the letdown line in series operation with the mixed-bed demineralizer.

During reactor startup from the cold condition, hydrazine is added to the coolant 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.

Oxygen concentration must be controlled to less than 0.1 ppm in the reactor coolant at temperatures above 250°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.

The reactor coolant is treated with dissolved hydrogen to control the net decomposition of water by radiolysis in the core region. The hydrogen also reacts with oxygen and nitrogen introduced into the RCS as impurities under the impetus of core radiation. 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. A self-contained pressure control valve maintains a minimum pressure in the vapor space of the volume control tank (VCT). This can be adjusted to provide the correct equilibrium hydrogen concentration.

Boron, in the chemical from of boric acid, is added to the RCS to accomplish long-term reactivity control of the core. The mechanism for the process involves the absorption of neutrons by the boron-10 isotope of naturally occurring boron.

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 though the CVCS mixed-bed demineralizer.

Zinc may be added to the reactor coolant system to reduce radiation fields and to reduce primary water stress corrosion cracking of Inconel-600 components. The zinc used may be either natural zinc or depleted of ^{64}Zn. When used, the target system zinc concentration is normally maintained to a concentration no greater than 40 ppb.

5.2.3.2.2 Compatibility of Construction Materials With Reactor Coolant: Except for the bottom mounted instrumentation penetration annular spaces exposed to primary coolant as a result of half-nozzle repair, all ferritic low-alloy and carbon steels which are used in principal pressure-retaining applications are provided with corrosion-resistant cladding on all surfaces that are exposed to the reactor coolant. This cladding material has a chemical analysis which 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. The cladding on ferritic-type base materials receives a postweld heat treatment, as required by the ASME Code.

Ferritic low alloy and carbon steel nozzles are safe-ended with either stainless steel wrought materials, stainless steel weld metal analysis A-7 (designated A-8 in the 1974 edition of 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 postweld heat treatment when the nozzle is larger than a 4-in. nominal inside diameter and/or the wall thickness is greater than 0.531 inches.

For weld joints provided by Bechtel Energy Corporation (BEC) between stainless steel and ferritic materials that require postweld heat treatment, the ferritic material is buttered with stainless steel or a nickel-chrome weld filler, heat treated, then welded to the stainless material. If a transition piece is used, the material will be low-carbon grade stainless steel.

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

During subsequent fabrication, these materials are not heated above 800°F other than instantaneously and 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 will operate satisfactorily under normal plant chemistry conditions in PWR systems because chlorides, fluorides, and oxygen are controlled to very low levels.

A new internals disconnect device (IDD) is included in the replacement reactor vessel head. The new IDD replaces the original upper Conoseal with a Grafoil seal, and the lower seal weld has been eliminated and replaced with a bimetallic weld between the IDD housing and penetration tube. The material for this new design is Alloy 690. The welding material used to join Ni-Cr-Fe alloys IDD penetration tubing with austenitic base materials for the IDD housing is compatible with the Ni-Cr-Fe base metal and complies with ASME Section II-C Material Specifications SFA 5.11 or 5.14, and ASME Code Cases 2142-1, 2142-2, or 2143-1. The IDD penetration nozzle weld buildup material is SA-508, Class 3. The crank handle assembly is SA-479 TP204, carbon content .045% -.06%, cobalt content 0.200 max. The material complies with ASME Code NB-2000 requirements. The seal assembly seal plug is SA-479 Type 304 stainless. The seal nut is SA-479 Grade UNS-S21800 (Niotronic 60).

5.2.3.2.3 Compatibility With External Insulation and Environmental Atmosphere: In general, all of the materials listed in Tables 5.2-2 and 5.2-3 which are used in principal pressure-retaining applications and which 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 Westinghouse practice meets recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel", but is more stringent in several respects, as discussed below.

The nonmetallic thermal insulation supplied by Westinghouse on the RCPB is specified to be made of compounded materials which yield low leachable chloride and/or fluoride concentrations. The compound materials, in the from 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. Each lot of insulation material is qualified and analyzed to assure that all of the materials provide a compatible combination for the RCPB.

The tests for qualification of Westinghouse-supplied nonmetallic insulation, as specified by the guide (ASTM C692-71 or RDT M12-IT), allow use of the tested insulation materials if no more than one of the metallic test samples cracks. Westinghouse rejects the tested insulation material if any of the test samples crack.

The Westinghouse procedure is more specific than the procedures suggested by RG 1.36 in that the Westinghouse specification requires determination of leachable chloride and fluoride ions from a sample of the insulating material. The procedures in the guide (ASTM D512 and ASTM D1179) do not differentiate between leachable and unleachable halogens.

In addition, Westinghouse experience indicates that only one of the three methods allowed under ASTM D512 and ASTM D1179 for chloride and fluoride analysis is sufficiently accurate for reactor applications. This is the “referee” method, which is used by Westinghouse. These requirements are defined in Westinghouse Process Specification PS-83336KA.

The thermal insulation used on the RCPB, not supplied by Westinghouse, is a low-density, semi-rigid fibrous glass material. This insulation protects the austenitic stainless steel against stress corrosion. It yields low concentrations of leachable fluoride and/or chloride in compliance with RG 1.36. Each lot of insulation material is analyzed and qualified to assure compatible combination for the RCPB.

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 Tables 5.2-2 and 5.2-3. 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.

The following controls are exercised to assure that nonmetallic thermal insulations employed do not contribute significantly to stress corrosion of stainless steel.

1. All nonmetallic thermal insulation for use on austenitic stainless steel components is manufactured and processed in a way that limits contamination by chloride and fluoride compounds.

2. All insulation materials are packaged in suitable containers which provide protection of the insulation during shipment and storage. All containers are marked as required by the ANSI N45.2.2.

3. All insulation materials for austenitic stainless steel are stored in a clean, dry storage area as required by ANSI N45.2.2 Level C.

4. Nonmetallic thermal insulation for application to austenitic stainless steel components complies with the requirements of RG 1.36.

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 ASME Code, Section III, Paragraphs NB, NC, and ND-2300 as appropriate.

Limiting SG and pressurizer RT_{NDT} temperatures are guaranteed at 60°F for the base materials and the weldments. These materials will 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 STPEGS at the time of shipment of the component.

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

Westinghouse had 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 5.2-7, which has been submitted to the Nuclear Regulatory Commission (NRC) for review. The conclusions of WCAP-9292 apply to SA 508 Class 1a and 3a, and SA 533 Grade B Class 2 materials, which are used in the construction of the SGs and the pressurizers.

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.

Where electorslag welding is used in fabricating nuclear plant components, the Westinghouse procurement procedure requires vendors to meet the guidelines of RG 1.34.

Westinghouse meets the intent of RG 1.43 by requiring qualification of any high-heat input process, such as the submerged-arc wide-strip welding process and the submerged-arc-6-wire process used on SA-508 Class 2 material, with a performance test as described in regulatory position 2 of the guide. No qualifications are required by the RG for SA-533 material and equivalent chemistry for forging grade SA-508 Class 3 material.

Westinghouse considers that RG 1.50 applies to ASME Code Section III Class 1 (BEC includes Class 2 and 3) components within their scope of responsibility.

The Westinghouse practice for Class 1 components is in agreement with the requirements of RG 1.50 except for regulatory positions C.1(b) and C.2. For Class 2 and 3 components, Westinghouse does not apply any of the RG 1.50 recommendations.

In the case of regulatory position C.1(b), the welding procedures are qualified within the preheat temperature ranges required by Section IX of the ASME Code. Westinghouse experience has shown excellent quality of welds using the ASME qualification procedures.

In the case of regulatory position C.2, the Westinghouse position described in WCAP-8577, "The Application Preheat Temperatures after Welding of Pressure Vessel Steel" (Ref. 5.2-6), has been found acceptable by the NRC. This WCAP establishes the guidelines which permit the component manufacturer to either maintain the preheat until a postweld heat treatment or allow the preheat to drop to ambient temperature.

In the case of reactor vessel main structural welds, the practice of allowing preheat to drop to ambient has been followed by Westinghouse. In either case, the welds have shown high integrity.

Westinghouse meets regulatory position C.4 in that, for Westinghouse components, the examination procedures required by ASME Code Section III and the in-service inspection requirements of ASME Code Section XI are met.

Westinghouse considers the RG 1.66, “Nondestructive Examination of Tubular Products,” position regarding angle beam scanning in the axial direction as technically unnecessary, since any flaws which might be developed by the processes employed in tubular product manufacture are invariably oriented in the axial direction, and the probability of developing metallurgical flaws of other than axial orientation is virtually nil. Flaws of transverse or circumferential orientation which might be developed would normally be mechanically-induced surface defects which should be detected by surface nondestructive examination procedures.

The primary pressure boundary and safety-related tubular products within the Westinghouse scope of supply and the nondestructive examinations applied are tabulated in Table 5.2-6. In all cases, the volumetric nondestructive examination is designed to detect the flaws inherent to the manufacturing process or processes employed. In those few cases where the guide is not followed, Westinghouse believes that the nondestructive examinations performed in the normal procurement of the tubular products covered by the guide achieve the same purpose as the guide.

The hardware items identified in Table 5.2-6 as not being in agreement with RG 1.66 are constructed in accordance with the applicable ASME Code Section III rules, as a minimum. However, the ASME Code Section III rules do not require the suggested provisions of the Guide for tubular products smaller than 2.5 in. outside diameter, which have been examined by the Eddy Current Method or for tubing larger than 6.75 in. outside diameter.

Westinghouse practice does not require qualification or requalification of welders for areas of limited accessibility, as described by RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” and has provided welds of high quality.

Westinghouse believes that limited accessibility qualification or requalification, which are additional to ASME Code Section III and IX requirements, is an unduly restrictive requirement for shop fabrication, where the welder’s physical position relative to the welds is controlled and presents no significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding is closely supervised.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel. Sections 5.2.3.4.1 through 5.2.3.4.5 address RG 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 of austenitic stainless steel components. Westinghouse compliance with the separate positions of the guide is as follows.

The use of processing, packaging, and shipping controls and preoperational cleaning to preclude adverse effects of exposure to contaminants on all stainless steel materials is in accordance with regulatory position C.1 of the RG.

Austenitic stainless steel starting materials are utilized in the final heat-treated condition required by the respective ASME Code Section II material specification for the particular type or grade of alloy, in accordance with regulatory position C.2 of the RG.

Westinghouse meets the intent of regulatory position C.4. However, the RG’s exception, (a), to regulatory position C.4, established 200°F as the upper limit for dissolved oxygen concentration above 0.1 ppm. This temperature limit should be increased to 250°F, which provides for much quicker reduction of the oxygen concentration by reaction with hydrazine. Startup operations provide for adding hydrazine after the temperature is at about 225°F. Oxygen scavenging at this temperature is rapid and complete; below 200°F considerable time can be encountered in the oxygen removal operations.

The Westinghouse position is that the significant chemistry control associated with the above limits on oxygen is the control of chloride and fluoride ion concentrations at less than 0.15 ppm each at all times.

Field Erection Welds

To meet the requirements of RG 1.44, welding procedures which were qualified in accordance with ASME Code, Section III, are used during field erection. These welding procedures also meet the following requirements:

1. All austenitic stainless steel filler metal was required to produce 5 to 25 percent delta ferrite.

2. Heat input (as determined by amperage, voltage, and travel speed, or by bead volume) and interpass temperatures are specified in the welding procedures so as to minimize propensity for hot cracking (fissuring) and severe sensitization.

To meet the requirements of RG 1.37, process controls were employed during installation to provide a degree of surface cleanliness and to minimize exposure to contaminants capable of causing stress corrosion cracking.

1. Only new hand tools (such as files, wire brushes, grinding wheels) or hand tools which had not been previously used on other than austenitic materials were used on austenitic stainless steel and nickel alloys. In addition, certain hand tools met the following requirements:

    a. Lead or brass hammers were not used

    b. Stainless steel bristle wire brushes were used

    c. Aluminum oxide grinding wheels were used

2. Liquid penetrate used for the inspection of stainless steel or nickel alloy met the halogen and sulfur requirements of Article 6 of ASME Section V 1974 including the Winter Addenda 1975.

3. Procedures were implemented to minimize the possibility of welding fluxes becoming trapped in crevices and the possibility of spent welding materials becoming trapped in a component or system.

4. During shipping, storage, and erection, all austenitic stainless and nickel alloy components were protected in a way that minimized contamination from elements which could cause stress corrosion cracking.

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 cold lead to stress corrosion cracking. The rules covering these controls are stipulated in the 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 for the STPEGS NSSS, regardless of the ASME code classification. The process specifications are also given to the responsible site organization for use within their scope of supply and activity.

The process specifications which define these requirements and which follow the guidance of the ANSI N45 Committee specifications are as follows:

Process Specification Number	
82560HM	Requirements for Pressure Sensitive Tapes for use on Austenitic Stainless Steel Reactor Components and Systems
83336KA	Requirements for Thermal Insulation Used on Austenitic Stainless Steel Reactor Plant Piping and Equipment
83860LA	Requirements for Marking of Reactor Plant Components and Piping
84350HA	Site Receiving Inspection and Storage Requirements for Nuclear Steam Supply Systems, Material and Equipment
84351NL	Determination of Surface Chloride and Fluoride Contamination on Austenitic Stainless Steel Materials
85310QA	Packaging Nuclear Components and Spare Parts for Shipment
292722	Cleaning and Packaging Requirements of Equipment fur Use in the NSSS
597756	Pressurized Water Reactor Auxiliary Tanks Cleaning Procedures
97760	Cleanliness Requirements During Storage, Construction, Erection and Start-Up Activities of Nuclear Power System
833128LW, Rev. E	Cleanliness Control and Cleaning Requirements (Steam Generators)

5.2.3.4.2 Solution Heat Treatment Requirements: The austenitic stainless steels listed in Tables 5.2-2, 5.2-3, and 5.2-5 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 Inspection Program: The Westinghouse practice is that austenitic stainless steel materials of product forms with simple shapes need not be corrosion tested provided 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 A262-70, Practice A or E, as amended by Westinghouse Process Specification PS-84201MW.

5.2.3.4.4 Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels: Unstabilized austenitic stainless steels are subject to intergranular attack provided that three conditions are present simultaneously. These are:

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

2. A sensitized steel

3. A high temperature

If any one of the three conditions described above is not present, intergranular attack will not occur. Since high temperatures cannot be avoided in all components in the NSSS, Westinghouse relies on the elimination of conditions 1 and 2 to prevent intergranular attack on wrought stainless steel components.

The water chemistry in the RCS of a Westinghouse PWR is rigorously 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. Reference 5.2-3 describes the precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage. 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-time exposure of severely sensitized stainless steels to PWR coolant environments in early plants has not resulted in any sign of intergranular attack. Reference 5.2-3 describes the laboratory experimental findings and the Westinghouse operating experience. The additional years of operations since the issuing of Reference 5.2-3 have provided further confirmation of the earlier conclusions. No evidence exists to imply that severely sensitized stainless steels undergo intergranular attack in Westinghouse PWR coolant environments.

Even though there has never been any 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 is used for components that are part of: (1) the RCPB, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals relied upon to permit adequate core cooling for normal operation or that are part of (a) the RCPB, (b) systems required for reactor shutdown, (c) systems required for emergency core cooling, and (d) reactor vessel internals relied upon to permit adequate core cooling for normal operation or under postulated accident conditions. Such stock is utilized in one of the following conditions:

1. Solution annealed and water quenched

2. 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 (ASTM 393) on as-received wrought material.

Westinghouse recognizes that the heat-affected zones of welded components must, of necessity, be heated into the sensitization temperature range, 800°F to 1,500°F. However, severe sensitization (i.e., continuous grain boundary precipitates of chromium carbide with adjacent chromium depletion) can still be avoided by control of 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.

Of 25 production and qualification weldments tested, representing all major welding processes and a variety of components, and incorporating base metal thickness from 0.10 to 4.0 in., only portions of two weldments were severely sensitized. One involved a heat input of 120,00 joules 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. However, in only the socket weld did the sensitized condition exist at the surface, where the material was exposed to the environment. The component has been redesigned and a material change has been made to eliminate this condition.

Westinghouse controls the heat input in all austenitic pressure boundary weldments by (1) prohibiting the use of block welding without engineering approval; (2) limiting the maximum interpass temperature to 350°F; and (3) exercising approval rights on all welding procedures for fabrication and installation for components within Westinghouse's scope of responsibility.

To further assure that these controls are effective in preventing sensitization, Westinghouse will, if necessary, conduct additional intergranular corrosion tests of qualification mock-ups of primary pressure boundary and core internal component welds, including the following:

- Reactor vessel safe ends
- Pressure safe ends

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* Heat input is calculated according to the formula:

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

where: H = joules/in.; E = volts; I = amperes; and S = travel speed in in./min.

- Surge line and RCP nozzles
- Control rod extensions
- Lower instrumentation penetration tubes

To summarize, Westinghouse has a four-point program designed to prevent intergranular attack of austenitic stainless steel components:

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

2. Utilization of materials in the final heat-treated coition and the prohibition of subsequent heat treatments in the 800°F to 1,500°F temperature range.

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

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

Both operating experience and laboratory experiments in primary water have conclusively demonstrated that this program is 100 percent effective in preventing intergranular attack in Westinghouse NSSSs utilizing unstabilized austenitic stainless steel.

5.2.3.4.5 <u>Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures</u>: It is not normal Westinghouse practice to expose unstabilized austenitic stainless steels to the sensitization range of 800°F to 1,500°F, during fabrication into components. If, during the course of fabrications, the steel is inadvertently exposed to the sensitization range, 800°F to 1,500°F, the material may be tested in accordance with ASTM A262 as amended by Westinghouse Process Specification PS-84201MW to verify that it is not susceptible to intergranular attack, except that testing is not required for:

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

2. Material with a carbon of 0.03 percent or less that is subjected to temperatures in the range of 800°F to 1,500°F for less than one hour.

3. 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 intergranular attack, the material will again be solution-annealed and water-quenched or will be rejected.

5.2.3.4.6 <u>Control of Welding</u>: The following paragraphs address RG 1.31, "Control of Stainless Steel Welding", and present the methods used and 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 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 the controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated, or stamped in accordance with ASME Code, Section III, Class 1 and 2 and core support components. Delta ferrite control is appropriate for the above welding requirements except where no filler metal is used or for other reasons for which 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 ASME Code, 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* as determined by chemical analysis and calculation, using the appropriate weld metal constitution diagrams. When new welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, they are performed in accordance with the requirements of ASME Code, Sections 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 ASME Code, Section III.

The "starting" welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of ASME Code, 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, for all applications with the exception that Type 308L weld metal analysis may be substituted for consumable inserts when used for weld root closures. Bare weld filler metals, 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 SFA-5.9 and are procured in a wire-flux combination to be capable of providing not less than 5 percent delta ferrite in the deposit, according to Section III. Welding materials are tested using the welding energy inputs to be employed in production welding, for components within Westinghouse scope of responsibility.

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 quality assurance audit including calibration of gages and instruments, identification of "starting" and completed materials, welder and procedure qualifications, availability and use of approved welding and heat treating procedures, and documentary evidence of compliance with materials, welding parameters, and inspection

* The equivalent ferrite number may be substituted for percent delta ferrite.

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

To assure the reliability of these controls, Westinghouse has completed a delta ferrite verification program, described in Reference 5.2-4, which has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative for conformance with RG 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 5.2-4, are summarized in Reference 5.2-5.

Section 5.2.3.3 includes discussions which indicate the degree of conformance of the austenitic stainless steel components of the RCPB with RGs 1.34 – "Control of Electroslag Properties", 1.66 – "Welder Qualification for Areas of Limited Accessibility".

1. <u>Field Erection Welds</u>

   The fabrication and installation specifications require welding procedures and welder qualification in accordance with ASME Code Section IX and Section III. Additionally, heat input (as determined by amperage, voltage, and travel speed, or by bead volume) and interpass temperatures are specified in the welding procedures.

2. <u>Production Welding</u>

   a. Filler materials used during production welding meet the 5 to 25 percent delta ferrite requirement.

   b. Welding parameters are monitored during production welding.

   c. The sampling and testing program described in the STPEGS Preliminary Safety Analysis Report (PSAR) has been deleted pursuant to Revision 2 or RG 1.31, as discussed in a letter from HL&P to the NRC dated August 25, 1977.

   d. The requirements of RG 1.71, are satisfied as discussed below:

      1) Performance qualifications for personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX. Additionally, responsible site supervisors are required to assign only the most highly skilled welders to limited access welding. Of course, welding conducted in areas of limited access is subjected to the required nondestructive testing and no waiver or relaxation of examination methods or acceptance criteria because of the limited access is permitted. Reference: Paragraph C.1 of the RG.

      2) Requalification is required when any of the essential variables of ASME Section IX are changed, or when any authorized inspector questions the ability of the welder to perform satisfactorily the requirements of ASME Sections III or IX. Reference: Paragraph C.2 of the RG.

3) Production welding is monitored and welding qualifications are certified in accordance with (1) and (2) above. Reference: Paragraph C.3 of the RG.

5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

Preservice and inservice inspections (ISI) of ASME Code Class 1 components shall be performed in accordance with Section XI of the ASME B&PV Code and applicable addenda as defined by 10CFR 50.55a(b) and 10 CFR 50.55a(g).

Units 1 and 2 of the STPEGS are configured to satisfy the Section XI Code. Unit 1 and 2 Preservice examinations were performed to the requirements set forth by the 1980 edition of Section XI with addenda through the Winter 1981.

The first 10 years of inservice examination and inservice testing was performed to the requirements of the 1983 edition of Section XI with addenda through the Summer 1983 Addenda. The inservice examination and inservice testing programs are upgraded to later editions and addenda of the Section XI Code at the start of each 10-year inspection interval in accordance with 10CFR50.55a(g) (4) (ii), except for those exempted in accordance with UFSAR section 13.7.3.

5.2.4.1 System Boundary Subject to Inspection. The ASME Section III Class 1 components (and their supports) that make up the RCPB are subject to preservice inspection and testing by rules of the ASME Section XI Code. The system boundary includes all Class 1 pressure-retaining components such as pressure vessels, piping, pumps, and valves that are part of or are connected to the RCS, up to and including the following.

1. The outermost Containment isolation valve in system piping that penetrates the primary Reactor Containment

2. The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary Reactor Containment

3. The RCS safety and relief valves

The Bottom Mounted Instrumentation flux thimbles form part of the RCPB and are classified as ASME III Class 2. The periodic inspection of the flux thimbles is described in UFSAR section 7.7.1.9.2.

5.2.4.2 Access Provisions. Access is provided for the inspector and for examination personnel and equipment in accordance with Subarticle IWA-1500 of Section XI of the ASME B&PV Code, 1974 edition including the Summer 1975 Addenda. Access for some systems and parts thereof is designed in accordance with the requirements of later editions and addenda up to the code used for preservice examination. Provisions for the removal and storage of structural members, shielding, insulation materials, etc., that would restrict access for examination, are included in the plant design and operating procedures. More specifically, access is provide for visual, surface, and volumetric examinations of welds and their adjacent base metal by means of removable insulation, removable shielding, and installation of permanent tracks for remote inspection devices in areas where personnel access is restricted by space, temperature, and/or high-radiation environments.

Also, working platforms are provided at strategic locations in the plant to permit access to those areas of the RCPB that are designated as inspection points in the ISI program.

5.2.4.2.1 Reactor Vessel Access Design Features: The vessel is designed to permit compliance with ASME B&PV Code Section XI, assuming that all volumetric examinations of pressure-containing welds will be performed primarily from internal surfaces of the vessel. However, certain areas of the closure head and the lower head-to-shell course welds are accessible only from the outer surface of the vessel.

The following areas of the reactor vessel will be available for nondestructive in-service examinations:

- Full-penetration pressure-retaining welds in the following areas: vessel shell inside surface; reactor vessel nozzle inside surfaces; closure head inside surfaces; bottom head inside surfaces; nozzle safe ends and adjacent piping from inside and outside surfaces

- Closure studs, nuts, and washers

- Vessel flange stud hole threads, flange ligaments between threaded stud holes, and flange sealing surface

- Accessible pressure-retaining partial penetration welds

- Peripheral control rod drive housing welds

- Upper internal support housing welds to closure head

The following are special features incorporated into the vessel design and into the station facilities to achieve quality inservice examinations in an expeditious manner.

1. The reactor vessel cladding finish is ground to a 250 root mean square finish (or better) for an appropriate distance on either side of all weld centerlines to enable ultrasonic examinations of the weld metal and base metal for one-half plate thickness beyond the edge of the weld.

2. The reactor vessel closure head will be stored dry in an accessible area to provide direct access to welds from the outer surface of the head.

3. The closure head will be stored dry with the flange seal surface at least 18 in. above the floor to provide access for a visual examination of the closure head flange.

4. The reactor vessel studs, nuts, and washers will be removed to dry storage. Suitable handling equipment is provided for storing the studs on racks for cleaning and examination.

5. Access is provided for mechanized and manual type examinations of the nozzle-to-safe-end and safe-end-to-piping butt welds.

6. Working platforms or temporary scaffolds are provided to facilitate access to examination areas.

5.2.4.2.2 Piping System Access Design Features: ASME Code Section III Code Class 1 piping system welds, subject to volumetric or surface examinations, are designed to facilitate ultrasonic, liquid penetrate, and/or magnetic particle examinations. Weld profiles and finishes and component configurations and arrangements utilized to assure the examinability of these systems are described below.

5.2.4.2.2.1 ASME Section III Code Class 1 Piping System Configurations – Code Class 1 piping 4 in. and larger nominal pipe size is counterbored a minimum of two pipe-wall thicknesses (2T) from the weld end preparation for circumferential butt welds. This 2T counterbore requirement applies to piping butt welds only (both shop and field).

Branch pipe or fitting connections located near circumferential butt welds subject to ultrasonic examination are located so that a minimum axial clearance of 2T + 2 in. exists between the toe of the branch weld and the centerline of the circumferential butt weld, where T = nominal pipe wall thickness.

Straight sections of pipe or "spool pieces" are, whenever possible, located between pipe fittings or between fittings and pumps and valves on piping subjected to volumetric examination. These spool pieces have a minimum length of 2T + 2 inches.

All bolted or welded pipe hangers and pipe whip restraints are located at least 2T + 2in. from any welds that require ultrasonic examination; however, bolted hangers that can be removed to allow the examination may be located nearer than 2T + 2 in. from the weld.

In general, welds requiring ultrasonic examinations are not located in the reactor shield wall penetration holes.

5.2.4.2.2.2 Weld Profile and Finish for Class 1 Piping Systems Pipe-to-Pipe Circumferential Butt Welds – For welds of the same thickness as their interfacing base material thicknesses (pipe-to-pipe), the weld crown is contoured and finished insofar as practicable.

1. Pipe-to-Fitting (or Component) Circumferential Welds

For welds connecting pipe to fittings or components (pipe-to-valve, pipe-to-tee, pipe-to-reducer, pipe-to-nozzle, etc.), the weld crown will be contoured and finished insofar as practicable.

2. Longitudinal Welds

The weld crown will be contoured and finished insofar as practicable.

5.2.4.2.2.3 Access Provisions for Class 1 Piping Systems – Sufficient space around Class 1 piping systems and components requiring manual examinations (visual, surface, and/or volumetric) will be provided for the examiner as defined in ASME Section XI. Access provisions for the examinations of various types of welds are described below.

1. Removable insulation is provided for circumferential weld joints in seamless pipe for a minimum distance of 2T + 2 in., on each side of the weld. However, where the circumferential weld joins a pipe to a fitting, valve, pump, or any other type component, the above access is provided only on the pipe side of the weld.

2. Where seam-welded pipe or fittings are used, access is provided to the longitudinal seam welds for a distance of 15 in. from the adjacent circumferential weld (12 in. for code requirements plus 3 in. for work access).

3. Working platforms will be provided to facilitate the examination.

5.2.4.2.3 Pressurizer: The external surface is accessible for visual, surface or volumetric examination by removing the external insulation. A manway is provided to allow access for internal visual examination. The permanent insulation around the pressurizer heaters will be provided with a means to identify component leakages during system hydrostatic and leakage testing.

5.2.4.2.4 Steam Generators: The external surface is accessible for visual, surface, or volumetric examination by removing portions of the vessel insulation. Manways in the SG channel head provide access for internal visual examinations and eddy current tests of SG tubing.

5.2.4.2.5 Pump Pressure Boundaries: The internal pressure-retaining surfaces of the pumps are accessible for visual examination by removing the pump internals. External surfaces of the pump casing are accessible for visual, surface, or volumetric examination by removing component insulation. Visual examination of interior surfaces or pump casings will be performed when the pumps are disassembled for maintenance purposes.

5.2.4.2.6 Valve Pressure Boundaries: The internal pressure boundary surfaces of Class 1 valves over 4-in. nominal size are accessible for visual examination by disassembly. External surfaces of valve bodies over 1-in. nominal size are accessible for visual, surface, or volumetric examinations by removing component insulation. Visual examinations of the internal valve surfaces will be performed when the valves are disassembled for maintenance purposes.

5.2.4.3 Examination Techniques and Procedures. Examination techniques and procedures as applicable to each Class 1 component are in conformance with the requirements of Article IWA-2000 and Tables IWB-2500-1 of Section XI.

5.2.4.4 Inspection Intervals. The inspection interval, as defined in ASME Section XI is 10 years, per IWA-2400. These inspection intervals represent calendar years after the reactor facility has been placed into commercial service. The scheduling of inspection programs is in accordance with Paragraph IWA-2400 and Tables IWB-2500-1 of ASME Section XI. The frequency and extent of examinations within each inspection interval are defined in Table IWB-2500-1 of Section XI.

5.2.4.5 Examination Categories and Requirements. The examination categories, methods, and requirements appropriate for each examination area are in accordance with those set forth by Tables IWB-2500-1 of Section XI.

5.2.4.6 Evaluation of Examination Results. Evaluation of examination results for Class 1 components will be conducted in accordance with Articles IWA-3000 and IWB-3000 of Section XI. Unacceptable indications are repaired or replaced in accordance with the requirements of Article IWA-4000.

5.2.4.7 System Leakage Tests. System leakage tests of the reactor pressure vessel and RCPB are conducted in accordance with the requirements of Articles IWA-5000 and IWB-5000 of

Section XI. System leak tests are conducted prior to startup following each reactor refueling outage in accordance with IWB-5000.

Examinations performed during these tests may be conducted without the removal of insulation. The system leakage test program is consistent with the operating limitations during heatup and cooldown, as provided in Chapter 16.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection System

The RCPB Leakage Detection System provides a means of detecting significant leakage from the RCS. The system was designed to conform with NRC General Design Criterion (GDC) 30 and RG 1.45. identifiable leakage from the reactor pressure vessel (RPV) flange leakoff, valve leakoffs, RCP leakoffs, and drain line leakage is collected and measured in the reactor coolant drain tank (RCDT). Unidentified leakage is determined by measuring the increase in Containment sump level, Containment air particulate radioactivity, and Containment humidity. Leakage from the RCPB to auxiliary systems is monitored by the measurement of radioactivity and water inventory balances. Measurement of RCPB leakages is sufficiently sensitive to assure that small increases in leakage can be detected while total leakage remains below a value consistent with safe plant operation.

5.2.5.1 Reactor Coolant Pressure Boundary Leakage.

5.2.5.1.1 Normal Expected Leakage: Unidentifiable leakage from the RCPB into the Containment is expected to be less than 0.02 gal/min. Identifiable leakage into the RCDT is expected to be less than 0.3 gal/min, primarily from the RCPs seals.

5.2.5.1.2 Limits for Reactor Coolant Leakage: RCS leakage will be limited to the following by the STPEGS Technical Specifications:

a. No pressure boundary leakage

b. One gal/min unidentified leakage

c. 150 gal/day primary-to-secondary leakage through any one steam generator

d. Ten gal/min identified leakage from the RCS

e. 0.5 gal/min leakage per nominal inch of valve size up to a maximum of 5 gal/min at an RCS pressure of 2,235 ± 20 psig from any RCS Pressure Isolation Valve specified in Table 5.2-7. Test pressures less than 2,235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2,235 psig, assuming the leakage to be directly proportional to pressure differential to the one-half power.

5.2.5.1.3 Collection of Identified Leakage: Leakage from the RPV flange, valves, reactor coolant pump seals, and equipment drains is collected in the RCDT. The collected fluid is recirculated and cooled. As the level in the tank reaches a preset level, the discharge valve is opened, allowing the excess fluid to be discharged to the Boron Recycle System (BRS) holdup tanks. The discharge flow is measured by an integrating flow meter. Increased identified leakage from the

RCPB is indicated by an increase in the discharge flow. In addition, leakage from the RPV flange is identified by an increase in temperature that is monitored in the flange leakoff line.

5.2.5.1.4 Unidentified Leakage to Containment: Unidentified leakage to the Containment results in increased Containment air humidity, and sump level and flow. Leakage from the Main Steam, Feedwater, and Component Cooling Water (CCW) Systems and other nonradioactive sources results in increased quantities of condensate without an associated increase in background radioactivity. RCPB leakage is identifiable by a simultaneous increase in condensate and radioactivity. Leakage to the Reactor Containment Building (RCB) is collected and monitored, and the flow rate is determined to an accuracy of one gal/min or its equivalent.

5.2.5.2 Leakage Detection Methods.

5.2.5.2.1 Containment Airborne Particulate Radioactivity Monitor: The Containment airborne radioactivity monitor is described in Section 11.5. The particulate channel of that monitor detects a selected isotope which quantitatively relates the activity increase to the magnitude of RCPB leakage to the Containment.

5.2.5.2.2 Containment Sump Level and Flow Monitoring: Leakage from sources other than those collected in the RCDT is collected in the Containment normal and secondary normal sumps. Level is measured in each sump by a pressure-type level device. The plant computer calculates the level increase rate (inflow rate) for each sump and alarms for an increasing rate exceeding one gal/min. Since the secondary normal sump is pumped into the normal sump, the computer calculation is normalized for the increased level in the normal sump that is transferred from the secondary normal sump. Level recorders are provided in the control room and permit manual calculation of leakage rates based on the rise in level associated with leakage as a function of time. Flow from the normal sump is totalized and displayed for operator use in determining leakage rates.

5.2.5.2.3 Deleted

5.2.5.2.4 Secondary Leakage Monitoring: The following secondary monitoring methods supplement the primary monitoring methods discussed above.

5.2.5.2.4.1 Containment Humidity – Containment air humidity is measured using a temperature-compensated dew cell. The dew cell operates on the principle of water vapor adsorption by a hygroscopic salt, such as lithium chloride. When heated, the salt loses water vapor by evaporation. Therefore, at a given temperature, there is an equilibrium between adsorbed water vapor and evaporation. This equilibrium temperature is the dew point. Determination of relative humidity requires measurement of the dry bulb temperature in addition to the dew point temperature.

The Containment humidity monitor, ME/MT-9682 shown on Figure 9.4.5-1, measures the dew point temperature. The dew cell range is 0°F to 160°F, with an accuracy of ± 1°F in the relative humidity range of 12 percent to 99 percent. The dry bulb temperature is measured by TE/TI-9681, also shown on Figure 9.4.5-1. The dry bulb temperature range is 65°F to 120°F, with an accuracy of ± ½ percent.

An increase in the humidity of the Containment atmosphere indicates release of water into the Containment. A rapid increase of humidity over the background level by more than 10 percent can be taken as a probable indication of a leak.

5.2.5.2.4.2 Reactor Coolant Drain Tank Level and Flow – Identifiable reactor coolant leakage is collected in the RCDT. A flow transmitter in the discharge line transmits a signal to the plant computer. The flow signal is integrated by the computer and can be printed out for operator information. Tank level is displayed in the main control room to serve as a diverse means of determining leakage flow into the tank. High and low alarms are provided to alert the operator in the event there are abnormal conditions.

No continuous leakage is expected from the RPV flange during operation. Any leakage, however, is detected by a temperature sensor mounted on the flange leakoff line.

5.2.5.2.4.3 Condenser Vacuum Pump Monitor – The condenser vacuum pump monitor is described in Section 11.5. The monitor is an off-line radioactive gaseous type that measures radioactivity in the condenser off-gas indicative of steam generator tube leaks.

5.2.5.2.4.4 Component Cooling Water Monitor – The CCW radioactivity monitor is described in Section 11.5. The monitor is an off-line liquid monitor that measures radioactivity in component cooling water. Sources of radioactivity are the letdown HX, excess letdown HX, residual heat removal HX, RCDT HX, reactor coolant sample HX, seal water HX, and Spent Fuel Pool Cooling and Cleanup System HXs.

5.2.5.2.4.5 N-16 Adjacent to Line Primary to Secondary Leak Monitors. The N-16 Primary to Secondary Leak Monitoring Subsystem is described in section 11.5. These detectors are located adjacent to the main steam lines in the TGB. They measure high energy gamma activity in the main steam lines. High energy gamma radiation in the main steam lines in indicative of steam generator tube leaks.

5.2.5.3 Intersystem Leakage. The condenser vacuum pump monitor and CCW monitor are the primary means of detecting intersystem leakage. However, the operator can detect leakage on the basis of water inventory in the auxiliary systems. Suspected intersystem leakage can be verified by laboratory radiosotopic analysis of grab samples from the suspect systems.

5.2.5.4 System Sensitivity and Response Time.

5.2.5.4.1 Containment Airborne Particulate Radioactivity Monitor: The particulate monitor sensitivity and response time have been determined by analysis, with the conclusion that the monitor is capable of measuring the equivalent of a 1-gallon-per-minute RCPB leakage to the Containment within 1 hour. Measurement interferences due to external ambient gamma background, naturally occurring airborne radioactivity, and normally undetectable RCPB leakage was considered in the analysis.

5.2.5.4.2 Containment Sump Level and Flow Monitoring: A one-gallon-per-minute liquid flow rate into the Containment normal sumps result in level increases of 2.67 inches in 1 hour in the normal sump and 10.7 inches in 1 hour in the secondary normal sump.

5.2.5.4.3 Deleted

5.2.5.5 Seismic Capability of Systems. The monitors and instrumentation discussed in Sections 5.2.5.2.1 and 5.2.5.2.2, are designed and qualified to operate following an Operating Basis Earthquake. In addition, the Containment airborne particulate radioactivity monitor is designed and qualified to operate following a Safe Shutdown Earthquake (SSE). Seismic qualification is discussed in Section 3.10.

5.2.5.6 Indicators and Alarms. Indicators and alarms are provided in the main control room for each of the following monitors and instrumentation.

5.2.5.6.1 Containment Airborne Particulate Radioactivity Monitor: The display for this monitor is in units of $\mu Ci/cm^3$. The alarm setpoint is calculated on the basis of monitor sensitivity, which corresponds to an RCPB leakage rate of 1 gal/min, such that the alarm is initiated within one hour from the start of the increased leak.

5.2.5.6.2 Containment Sump Level and Flow: The sump level display is in units of inches. The rate at which the level increases is computed by the plant computer. An alarm occurs whenever the computed rate exceeds the equivalent of 1gal/min in one hour. The operator can also compute the rate manually with reading of sump level taken from a level recorder. In addition to the computed rate alarm, there is a sump high-level alarm.

5.2.5.6.3 Deleted

5.2.5.7 Testing. The radiation monitor is provided with test circuitry which permits on-line testing of channel electronics. A remotely operated radiation check source provides a test signal similar to the monitored radiation. The strength of the check source is sufficient to test the response of the channel, including the alarms. These tests can be initiated by the operator at his discretion. The monitors and the level, flow, and humidity instrumentation are calibrated during plant shutdown and maintenance periods. Testing and calibration comply with Paragraph 4.10 of Institute of Electrical and Electronics Engineers (IEEE) 279-1971.

5.2.5.8 Technical Specifications. The limits for reactor coolant leakage and availability of detection methods are provided in the Technical Specifications.

REFERENCES

Section 5.2:

5.2-1 WCAP-7769, Revision 1 (Topical Report – "Overpressure Protection for Westinghouse Pressurized Water Reactors"); approved by R. Salvatori, dated June 1972. Also, letter NS-CE-622, dated April 16, 1975: C. Eicheldinger to D. B. Vassallo; additional information on WCAP-7769, Revision 1.

5.2-2 WCAP-7907 (Loftran Code Description); approved by J. O. Cermak, dated October 1972.

5.2-3 Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems", WCAP-7735 (August 1971).

5.2-4 Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments", WCAP-8324-A (June 1974).

5.2-5 Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments", WCAP-8693 (January 1976).

5.2-6 Cuplan, J., "The Application of Pre-Heat Temperature after Welding Pressure Vessel Steel", WCAP-8577 (September 1975).

5.2-7 "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat Affected Zone Material and Applicable Weld Metals", WCAP-9292, March 1978.

5.2-8 "Overpressure Protection Report on South Texas Nuclear Power Plant Units 1 & 2", Certified: R. A. Weismann, April 1982.

TABLE 5.2-1

APPLICABLE CODE ADDENDA FOR RCS COMPONENTS

Reactor vessel (excluding replacement head)	ASME III, 1971 Ed. thru Summer 1973
Replacement reactor vessel head	ASME III, 1989 Ed.
Steam generator	ASME III, 1989 Ed.
Pressurizer	ASME III, 1974 Ed.
CRDM housing, full length	ASME III, 1989 Ed.
Reactor coolant pump	ASME III, 1971 Ed. thru Summer 1973
Reactor coolant pipe	ASME III, 1974 Ed. thru Winter 1975
Surge lines	ASME III, 1974 Ed. thru Winter 1975
Valves	
Pressurizer safety	ASME III, 1974 Ed. thru Summer 1975
Motor-operated	ASME III, 1974 Ed. thru Winter 1975
Manual (3 in. and larger)	ASME III, 1974 Ed. thru Winter 1975
Control	ASME III, 1974 Ed. thru Summer 1975
2 in. and under	ASME III, 1974 Ed. thru Winter 1975

TABLE 5.2-2

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS

Reactor Vessel Components	
Head forging	SA-508, Class 3
Shell plates (other than core region)	SA-533 Gr B, Class 1 (vacuum treated)
Shell plates (core region)	SA-533 Gr B, Class 1 (vacuum treated)
Shell, flange, and nozzle forgings, nozzle safe ends	SA-508 Class 2, SA-182 Type F316
CRDM and/or ECCS appurtenances – upper head	SB-167, SA-182 Type F304 and SA-182 Type F316
Instrumentation tube appurtenances – lower head	SB-166
Closure studs, nuts, washers, inserts and adaptors	SA-540 Class 3, Gr B24 (as modified by Code Case 1605)
Core support pads	SB-166 with carbon less than 0.10%
Vent pipe	SB-167 or SA-182 Type F316
Monitor Tubes	SB-166 or SA-167
Vessel supports, seal ledge, and heat lifting lugs	SA-516 Gr 70, quenched and tempered, or SA-533 Gr A, B or C, Class 1 or 2, or SA-508, Class 2 or SA-508 Class 3. (Vessel supports may be of weld metal buildup of equivalent strength.)
Cladding and buttering	Stainless steel weld metal analysis A-7 and Ni-Cr-Fe weld metal F-number 43

TABLE 5.2-2 (Continued)

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS

Steam Generator Components	
Pressure plates (manway covers)	SA-533 Gr. B Class 2
Pressure forgings (including primary nozzles and tubesheet)	SA-508 Class 3a
Nozzle safe end	SA-336 Type 316LN
Divider plates	SB-168 (Alloy 690)
Channel heads	SA-508 Class 3a
Tubes	SB-163 NiCrFe Thermally treated Alloy 690
Tubesheet cladding	SFA 5.14 Class ERNiCrFe-7 (NNS N06052) and SFA 5.11 Class ERNiCrFe-7 (UNS W86152)
Channel head cladding	SFA 5.9 Class ER 309L (1st layer) SFA 5.9 Class ER 308L (Remaining layers)
Closure studs	SA-193 Gr B7
Pressurizer Components	
Pressure plates	SA-533 Gr A Class 2
Pressure forgings	SA-508 Class 2 or 2a
Nozzle safe ends	SA-182 or 376 Type 316 or 316L and Ni-Cr-Fe weld metal F-number 43
Cladding and buttering	Stainless steel weld metal analysis A-7 and Ni-Cr-Fe weld metal F-number 43
Closure bolting	SA-193 Gr B7

TABLE 5.2-2 (Continued)

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS

Reactor Coolant Pump	
Pressure forgings	SA-182 F304, F316, F347, or F348
Pressure casting	SA-351 Gr CF8, CF8A or CF8M
Tube and pipe	SA-213, SA-376, or SA-312 seamless type 304 or 316
Pressure plates	SA-240 Type 304 or 316
Bar material	SA-479 Type 304 or 316
Closure bolting	SA-193, SA-320, SA-540, SA-453, Gr 660, SB-637 Gr. N07718
Flywheel	A-533 Gr B, Class 1
Reactor Coolant Piping	
Reactor coolant pipe	SA-351 Gr CF8A centrifugal casting
Reactor coolant fittings	SA-351 Gr CF8A
Branch nozzles	SA-182 Code Case 1423, Gr 316N
Surge line	SA-376 Gr 304, 316, or F304N
Auxiliary piping ½ in. through 12 in. and wall schedules 40S through 80S (ahead of second isolation valve)	ANSI B36.19
All other auxiliary piping (ahead of second isolation valve)	ANSI B36.10
Socket weld fittings	ANSI B16.11
Piping flanges	ANSI B16.5

TABLE 5.2-2 (Continued)

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS

Full-Length CRDM	
Latch	SA-182 Gr F316
Rod travel housing and cap (one piece)	SA-182 Gr F316
Welding materials	Stainless steel weld metal analysis A-8

TABLE 5.2-3

CLASS 1 AND 2 AUXILIARY COMPONENTS MATERIAL SPECIFICATIONS

Valves		
Bodies	SA-182 Type F316 or SA-351 Gr CF8 or CF8M	
Bonnets	SA-182 Type F316 or SA-351 Gr CF8 or CF8M	
Discs	SA-182 Type F316 or SA-564 Gr 630	
Pressure-retaining bolting	SA-453 Gr 660	
Pressure-retaining nuts	SA-453 Gr 660 or SA-194 Gr 6	
Auxiliary Heat Exchangers		
Tube sheets	SA-240 Type 304, SA-516 Gr 70 with stainless steel cladding (analysis A-8)	
Tubes	SA-213 TP304, SA-249 TP304	
Heads	Tube Side	Shell Side
	SA-240 Type 304 SA-182 Gr F304	SA-285 GrC SA-516 Gr 70
Nozzle necks	SA-240 Type 304	SA-166 Gr B
Shells	SA-240 Type 304	SA-106 Gr B, SA-285 Gr C, SA-516 Gr 70
Flanges	SA-182 Gr F-304	SA105

TABLE 5.2-3 (Continued)

CLASS 1 AND 2 AUXILIARY COMPONENTS MATERIAL SPECIFICATIONS

Auxiliary Pressure Vessels, Tanks, Filters, etc.	
Shells and heads	SA-240 Type 304 or SA-264 consisting of SA-537 Gr B with stainless steel weld metal analysis A-8 cladding
Flanges and nozzles	SA-182 Gr F304 and SA-105 or SA-350 Gr LF2, LF3 with stainless steel weld metal analysis A-8 cladding
Piping	SA-312, SA-240 TP304 or TP316, and SA-182 TP304 seamless
Pipe fittings	SA-403 WP304 seamless, SA-213 TP304
Closure bolting and nuts	SA-193 Gr B7 and SA-194 Gr 2H
Auxiliary Pumps	
Pump casing and heads	SA-351 Gr CF8 or CF8M, SA-182 Gr F304 or F316
Flanges and nozzles	SA-182 Gr F304 or F316, SA-403 Gr WP316L seamless
Piping	SA-312 TP304 or TP316 seamless
Stuffing or packing box cover	SA-351 Gr CF8 or CR8M, SA-240 TP304 or TP316
Piping fittings	SA-403 Gr WP316L seamless
Closure bolting and nuts	SA-193 Gr B6, B7 or B8M and SA-194 Gr 2H or Gr 8M, SA-193 Gr B6, B7 or B8M; SA-453 Gr 660; and nuts, SA-194 Gr 2H, Gr 8M, and Gr 6

TABLE 5.2-4

NSSS SUPPLIER RECOMMENDED REACTOR COOLANT CHEMISTRY SPECIFICATIONS AND GUIDELINES PER POWER OPERATION

Specification Parameters	Value
Boric Acid, ppm B and Lithium, ppm Li-7	Varies with Li/B ratio in coordinated treatment scheme
Hydrogen, $cm^3/kg\ H_20$*	25-50
Dissolved Oxygen, ppb	≤5
Chloride, ppb	≤150
Fluoride, ppb	≤150

Guideline Parameters	Value
Conductivity, µS/cn @ 25°C, pH @ 25°C	Varies with Li/B concentration
Silica, ppb	≤1000
Suspended Solids, ppb	≤200
Aluminum, ppb	≤50
Calcium + Magnesium, ppb	≤50
Magnesium, ppb	≤25
Zinc	Per Chemistry Program (maximum 40 ppb steady-state)

* At standard temperature and pressure

TABLE 5.2.4-1

AVOIDED CONFIGURATIONS IN CODE CLASS 1
PIPING SYSTEMS REQUIRING ULTRASONIC EXAMINATIONS

Avoided Configurations	Size Range
Component-to-component (e.g., valve-to-valve, pump-to-valve, etc.)	All nominal pipe sizes above 1 in. (2.54 cm)
Component-to-fitting (except elbow; see below)	All nominal pipe sizes above 1 in. (2.54 cm)
Fitting-to-fitting (except elbow; see below)	All nominal pipe sizes above 1 in. (2.54 cm)
Short-radius elbow-to-elbow, -to-valve, -to-tee, -to-reducer, -to-flange, -to-nozzle, or to-pipe spool piece less than 6 in. (15.24 cm) in length	Above 1 in. (2.54 cm) nominal pipe size to less than 20 in. (50.8 cm) nominal pipe size
Long-radius elbow-to-elbow, -to-valve, -to-tee, -to-reducer, -to-flange, -to-nozzle, or to-pipe spool piece less than 6 in. (15.24 cm) in length	Above 1 in. (2.54 cm) nominal pipe size to less than 10 in. (25.4 cm) nominal pipe size

TABLE 5.2-5

REACTOR VESSELS INTERNALS MATERIAL SPECIFICATIONS

Forgings	SA-182 Type F304
Plates	SA-240 Type 304
Pipes	SA-312 Type 304 Seamless or SA-376 Type 304
Tubes	SA-213 Type 304
Bars	SA-479 Types 304 & 410
Castings	SA-351 Gr CF8 or CF8A
Bolting	SA-193 GrB8M (65 MYS/90MTS) Code Case 1618 Inconel 750 SA-461 GR688
Nuts	SA-193 Gr B8
Locking devices	SA-479 Type 304

TABLE 5.2-6

TUBULAR PRODUCTS WITHIN WESTINGHOUSE
SCOPE RELATED TO REGULATORY GUIDE 1.66

Westinghouse Application	Tubular Product Form	UT		ET	RT	PT	Complies With RG 1.66
		Circ.	Axial				
Steam generator tubing	Seamless Tube	X		X			Yes
HX tubing	Welded Tube	Part.		X		OD	Yes
Vent pipe monitor tube	Seamless Tube	X(2)				OD	Yes
Instrument tubing	Seamless Tube	X(2)				OD ends	No
Instrument nozzles	Bar	X(2)	X			OD	Yes
CRDM housings	Forging	X(2)	X			OD, ID	Yes
Adaptor flange	Bar	X(2)	X			OD, ID	Yes
PC pipe branch nozzles	Bar Forgings	X(2)	X			OD, ID	Yes
RV nozzles	Forgings	X(2)	X				Yes
SG nozzles	Forgings	X(2)	X				Yes
SG nozzles	Castings				X		Yes
Pressure nozzles	Forgings	X(2)	X				Yes
RC pipe and fittings	Castings				X	OD, ID	Yes

Circ. – Circumferential

(2) Two directions

X – Indicates that the tests are performed

RT – Radiographic

UT – Ultrasonic

PT – Dye penetrant

ET – Eddy current

TABLE 5.2-6 (Continued)

TUBULAR PRODUCTS WITHIN WESTINGHOUSE
SCOPE RELATED TO REGULATORY GUIDE 1.66

Westinghouse Application	Tubular Product From	UT	ET	RT	PT	Complies With RG 1.66
		Circ.	Axial			
RC pipe wrought	Extrusions	X(1)			OD, ID Welds+3T(3)	No
Surge pipe	Extrusions	X(1)			OD, ID Welds+3T(3)	No
Surge pipe castings				X		Yes

Circ. – Circumferential

(1) Single direction

(3) 3T indicated liquid penetrate inspection of base metal at a distance from the weld equal to three times the wall thickness

X – Indicates that the tests are performed

RT – Radiographic

UT – Ultrasonic

PT – Dye penetrant

ET – Eddy current

TABLE 5.2-7

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
XSI0007 A, B, C	HHSI Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0009 A, B, C	HHSI Hot Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XSI0010 A, B, C	LHSI/HHSI Hot Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XRH0020 A, B, C	LHSI Hot Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XRH0032 A, B, C	LHSI/RHR Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0038 A, B, C	LHSI/HHSI/RHR Accumulator Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0046 A, B, C	Accumulator Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XRH0060 A, B, C	RHR Suction Isolation Valves (RCS Loops 1, 2, 3)
XRH0061 A, B, C	RHR Suction Isolation Valves (RCS Loops 1, 2, 3)

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Material Specifications. Material specifications are in accordance with the ASME Code requirements and are given in Section 5.2.3.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication.

1. The vessel is Safety Class 1. Design and fabrication of the reactor vessel is carried out in strict accordance with the American Society of Mechanical Engineers (ASME) Code, Section III, Class 1 requirements. The nozzles are manufactured as forgings. The cylindrical portion of the vessel is made up of several shells, each consisting of formed plates joined by full penetration longitudinal weld seams. The hemispherical replacement heads for Unit 1 and 2 are one-piece forging which eliminates the full penetration butt welds used at penetration locations in the original heads. The reactor vessel parts are joined by welding, using the single or multiple wire submerged arc. CN-2994

2. 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.

3. The surfaces of the guide studs are chrome-plated to prevent possible galling of the mated parts. CN-2994

4. 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.

5. Core region shells fabricated of plate material have longitudinal welds which are angularly located away from the peak neutron exposure experienced in the vessel, where possible.

6. The location of full penetration weld seams in the vessel bottom heads are restricted to areas that permit accessibility during in-service inspection. CN-2994

7. The stainless steel clad surfaces are sampled to assure that composition and delta ferrite requirements are met.

8. The procedure qualification for cladding low alloy steel (SA-508 Class 2) requires a special evaluation to assure freedom from underclad cracking.

5.3.1.3 Special Methods For Nondestructive Examination. The reactor vessel quality assurance program is given in Table 5.3-1.

5.3.1.3.1 Ultrasonic Examination:

1. The only plate material used was for the guide stud bracket; ultrasonic testing was by straight beam only. CN-2994

2. The reactor vessel is examined after hydrotesting for information which may be used by STPEGS to establish its subsequent inservice inspection program. CN-2994

5.3.1.3.2 Penetrant Examinations: The partial penetration welds for the CRDM (Unit 1 only) and CRDM head adaptors (Unit 2 only) and the bottom instrumentation tubes are inspected by dye penetrant after the root pass in addition to code requirements. Core support block attachment welds are inspected by dye penetrant after the first layer of weld metal and after each 1/2 in. of weld metal. All clad surfaces and other vessel and head internal surfaces are inspected by dye penetrant after the hydrostatic test.

5.3.1.3.3 Magnetic Particle Examination: The magnetic particle examination requirements below are in addition to the magnetic particle examination requirements of Section III of the ASME Code.

All magnetic particle examinations of materials and welds shall be performed in accordance with the following:

1. Prior to the final post weld heat treatment – only by prod or coil (reactor head only). CN-2994

   - only by prod, coil or direct contact (reactor vessel only).

2. After the final post weld heat treatment – only by the yoke method

The following surfaces and welds shall be examined by magnetic particle methods. The acceptance standards shall be in accordance with Section III of the ASME Code.

Surface Examinations

1. Magnetic particle examine all exterior vessel and head surfaces after the hydrostatic test.

2. Magnetic particle examine all exterior closure stud surfaces and all nut surfaces after final machining or rolling. Continuous circular and longitudinal magnetization shall be used.

3. Magnetic particle examine all inside diameter surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. This inspection to be performed after forming and machining (if performed) and prior to cladding.

Weld Examination

Magnetic particle examination of the weld metal buildup for vessel support welds attaching the closure head lifting lugs and refueling seal ledge to the reactor vessel after the first layer and each 1/2 in. of weld metal is deposited. All pressure boundary welds shall be examined after back chipping 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 Section 5.2.3. Sections 5.2.3.3 and 5.2.3.4 include discussions which indicate the degree of conformance with Regulatory Guides (RG) 1.44 – "Control of the Use of Sensitized Stainless Steel"; 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 Steels"; 1.71 – "Welder Qualification For Areas of Limited Accessibility". Section 5.2.3.4.6 includes a discussion which indicates the degree of conformance with RG 1.31.

5.3.1.5 Fracture Toughness. Assurance of adequate fracture toughness of ferritic materials in the reactor coolant pressure boundary (RCPB) (ASME Section III Class I Components) is provided by compliance with the requirements for fracture toughness testing included in NB-2300 to Section III of the ASME Boiler and Pressure Vessel Code (B&PV) and Appendix G of 10CFR50.

The initial Charpy V-notch minimum upper shelf fracture energy levels of the reactor vessel beltline (including welds) shall be 75 ft-lb as required per Appendix G of 10CFR50. Materials having a section thickness greater than 10 in. with an upper shelf of less than 75 ft-lb shall be evaluated with regard to effects of chemistry (especially copper content), initial upper shelf energy and fluence to assure that a 50 ft-lb shelf energy as required by Appendix G of 10CFR50 is maintained throughout the life of the vessel. The specimens shall be oriented as required by NB-2300 of Section III of the ASME B&PV Code. Fracture toughness data for the Unit 1 and Unit 2 reactor vessels is presented in Tables 5.3-3 and 5.3-4, respectively.

5.3.1.6 Material Surveillance. In the surveillance program, the evaluation of the radiation damage is based on the pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile and, when desirable, 1/2 T (thickness) 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 will meet the intent of ASTM E185-73, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels", and 10CFR50, Appendix H.

The reactor vessel surveillance program uses six specimen capsules. The capsules are located in guide baskets welded to the outside of the neutron shield pads and are positioned directly opposite the center portion of the core. The capsules can be removed when the vessel head is removed an 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 will be retained.

Dosimeters, including nickel, copper, iron, cobalt-aluminum, cadmium-shielded cobalt-aluminum, cadium-shielded neptuniun-237 and cadmium-shielded uranium-238, 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. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent will be made for surveillance material and as deposited weld metal.

Each of the six capsules contains the following specimens:

Materials	Number of Charpys	Number of Tensiles	Number of CT's
Plate R1606-2 (Unit 1) Plate R2507-1 (Unit 2)*	15	3	4
Plate R1606-2 (Unit 1) Plate R2507-1 (Unit 2)**	15	3	4
Weld Metal***	15	3	4
Heat Affected Zone	15		

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

Dosimeters

Iron

Copper

Nickel

Cobalt-Aluminum (0.15 percent Co)

Cobalt-Aluminum (Cadmium-shielded)

U-238 (Cadmium-shielded)

Np-237 (Cadmium-shielded)

Thermal Monitors

97.5 percent lead, 2.5 percent silver (579°F Melting Point)

97.5 percent lead, 1.75 percent silver, 0.75 percent tin (590°F Melting Point)

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* Specimens oriented in the major rolling or working direction.

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

*** Weld metal identical to heat of wire and lot of flux used to fabricate vessel beltline region intermediate to lower shell girth weld.

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. Appendix H of 10CFR50 and ASTM E185-73 require that surveillance capsules be located such that the neutron flux received by the specimens in the capsules is not more than three times as high s that received by the vessel inner surface; this is an arbitrary limit that was imposed to assure that the capsules are positioned as close as practicable to the inner surface of the vessel. The six capsules are positioned as close as practicable to the inner surface of the vessel at the same radial distance from the center of the core. At these locations, four of the capsules will receive a neutron flux of 4.05 times as high as that received by the vessel inner surface and two will receive a neutron flux of 3.37 times as high as that received by the vessel inner surface. Although this deviates from the Appendix H requirements, the higher (than three) capsule-to-vessel inner wall lead factors are considered to be technically adequate and test results from these capsules are considered to be appropriate for monitoring the effects of radiation on the vessel materials. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition 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 the measurements of the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Section 5.3.1.6.1. They have indicated good agreement. The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution will be considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure will be made by use of data on all capsules withdrawn. The schedule for removal of capsules for post-irradiation testing is based upon the capsule withdrawal schedules identified in Appendix H of 10CFR50 and ASTM E185. The recommended schedule for removal of the STP capsules for post-irradiation testing is presented in Table 16.1-2, “Reactor Vessel Material Surveillance Program – Withdrawal Schedule”.

5.3.1.6.1 Measurement of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples: The use of passive neutron sensors such as those included in the internal surveillance capsule dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average flux level, and hence, time integrated exposure (fluence) experienced by the sensors may be developed from the measurements only if the sensor characteristics and the parameters of the irradiation are well known. In particular, the following variables are of interest:

1) the measured specific activity of each sensor;

2) the physical characteristics of each sensor;

3) the operating history of the reactor;

4) the energy response of each sensor; and

5) the neutron energy spectrum at the sensor location.

In this section the procedures used to determine sensor specific activities, to develop reaction rates for individual sensors from the measured specific activities and the operating history of the reactor, and to derive key fast neutron exposure parameters from the measured reaction rates are described.

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

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

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

$$R = \frac{A}{No\,F\,Y\,\Sigma j\,\frac{Pj}{Pref}\,Cj\left[I - \ell^{\Delta tj}\right]\ell^{td}}$$

where:

A = measured specific activity provided in terms of disintegrations per second per gram of target material (dps/gm).

R = reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} expressed in terms of reactions per second per nucleus of target isotope (rps/nucleus).

No = number of target element atoms per gram of sensor.

F = weight fraction of the target isotope in the sensor material.

Y = number of product atoms produced per reaction.

P_j = average core power level during irradiation period j (MW).

P_{ref} = maximum or reference core power level of the reactor (MW).

C_j = calculated ratio of $\phi(E > 1.0\text{ MeV})$ during irradiation period j to the time weighted average ϕ (E > 1.0 MeV)over the entire irradiation period.

A = decay constant of the product isotope (sec^{-1}).

t_j = length of irradiation period j (sec).

t_d = decay time following irradiation period j (sec).

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

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

5.3.1.6.1.2 Corrections to Reaction Rate Data: Prior to using the measured reaction rates in the least squares adjustment procedure discussed in Section 5.3.1.6.1.3, additional corrections are made to the U-238 measurements to account for the presence of U-235 impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation.

In addition to the corrections made for the presence of U-235 in the U-238 fission sensors, corrections are also made to both the U-238 and Np-237 sensor reaction rates to account for gamma ray induced fission reactions occurring over the course of the irradiation.

5.3.1.6.1.3 Least Squares Adjustment Procedure: Least squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as neutron fluence (E > 1.0 MeV) or iron atom displacements (dpa) along with their uncertainties are then easily obtained from the adjusted spectrum. The use of measurements in combination with the analytical results reduces the uncertainty in the calculated spectrum and acts to remove biases that may be present in the analytical technique.

In general, the least squares methods, as applied to pressure vessel fluence evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{R_1} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross-section, $\sigma_{ig,}$ each with an uncertainty δ.

The use of least squares adjustment methods in LWR dosimetry evaluations is not new. The American Society for Testing and Materials (ASTM) has addressed the use of adjustment codes in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance" and many industry workshops have been held to discuss the various applications. For example, the ASTM-EURATOM Symposia on Reactor Dosimetry holds workshops on neutron spectrum unfolding and adjustment techniques at each of its bi-annual conferences.

The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement. The analytical method alone may be deficient because it inherently contains uncertainty due to the input assumptions to the calculation. Typically these assumptions include parameters such as the temperature of the water in the peripheral fuel assemblies, by-pass region, and downcomer regions, component dimensions, and peripheral core source. Industry consensus indicates that the use of calculation alone results in overall uncertainties in the neutron exposure parameters in the range of 15-20% (1σ).

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For a given application, the calculated neutron spectrum is obtained from the results of plant specific neutron transport calculations applicable to the irradiation period experienced by the dosimetry sensor set. This calculation is performed using the benchmarked transport calculational methodology described in Section 5.3.1.6.2. The sensor reaction rates are derived from the measured specific activities obtained from the counting laboratory using the specific irradiation history of the sensor set to perform the radioactive decay corrections. The dosimetry reaction cross-sections and uncertainties that are utilized in LWR evaluations comply with ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)".

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum are input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties also follows the guidance provided in ASTM Standard E 944.

5.3.1.6.2 Calculation of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples: A generalized set of guidelines for performing fast neutron exposure calculations within the reactor configuration, and procedures for analyzing measured irradiation sample data that can be correlated to these calculations, has been issued by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence [Reference 5.3-17]. Since different calculational models exist and are continuously evolving along with the associated model inputs, e.g., cross-section data, it is worthwhile summarizing the key models, inputs, and procedures that the NRC staff finds acceptable for use in determining fast neutron exposures within the reactor geometry. This material is highlighted in the subsection of material that is provided below.

5.3.1.6.2.1 Calculation and Dosimetry Measurement Procedures: The selection of a particular geometric model, the corresponding input data, and the overall methodology used to determine fast neutron exposures within the reactor geometry are based on the needs for accurately determining a solution to the problem that must be solved and the data/resources that are currently available to accomplish this task. Based on these constraints, engineering judgement is applied to each problem based on an analyst's thorough understanding of the problem, detailed knowledge of the plant, and due consideration to the strengths and weaknesses associated with a given calculational model and/or methodology. Based on these conditions, Regulatory Guide 1.190 does not recommend using a singular calculational technique to determine fast neutron exposures. Instead, Regulatory Guide 1.190 suggests that one of the following neutron transport tools be used to perform this work.

- Discrete Ordinates Transport Calculations

    a. Adjoint calculations benchmarked to a reference-forward calculation, or stand-alone forward calculations.

    b. Various geometrical models utilized with suitable mesh spacing in order to accurately represent the spatial distribution of the material compositions and source.

    c. In performing discrete ordinates calculations, Regulatory Guide 1.190 also suggests that a P_3 angular decomposition of the scattering cross-sections be used, as a minimum.

    d. Regulatory Guide 1.190 also recommends that discrete ordinates calculations utilize S_8 angular quadrature, as a minimum.

    e. Regulatory Guide 1.190 indicates that the latest version of the Evaluated Nuclear Data File, or ENDF/B, should be used for determining the nuclear cross-sections; however, cross-sections based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked are also acceptable.

- Monte Carlo Transport Calculations

A complete description of the Westinghouse pressure vessel neutron fluence methodology, which is based on discrete ordinates transport calculations, is provided in Reference 5.3-18. The Westinghouse methodology adheres to the guidelines set forth in Regulatory Guide 1.190.

5.3.1.6.2.2 Plant-Specific Calculations The most recent dosimetry analyses for both South Texas Project units were carried out using discrete ordinates transport techniques in R-Θ geometry using both a reference forward and unit/cycle-specific adjoint calculations. Similarly, the latest fast

(E > 1.0 MeV) neutron fluence evaluation for each South Texas Project reactor pressure vessel was based on a 2D/1D synthesis of neutron fluxes that were obtained from a series of plant-and cycle-specific forward discrete ordinates transport calculations run in R-Θ, R-Z, and R geometric models. Both sets of calculations, which assessed dosimetry as part of the reactor vessel surveillance program and pressure vessel neutron fluences, were conducted in accordance with the guidelines that are specified in Regulatory Guide 1.190.

5.3.1.7 Reactor Vessel Fasteners. The reactor vessel head closure system used for securing the vessel head to the reactor pressure vessel is defined as Roto-lok. This closure system is made up of a stud, vessel insert, closure nut, and spherical washer. These parts are made of 4340 steel as is the conventional closure system. Specific material properties are SA-540, Class 3, Gr B-24 (as modified by Code Case 1605).

The Roto-lok closure stud uses a modified breech-lock design to secure the reactor vessel to head. The interrupted lugs of the Roto-lok stud cut in the lower and upper ends are generated by cutting separate parallel grooves in the studs rather than a continuous helical groove as is the case with standard breech-lock threads. This modification prevents any contact with the engaging lugs when rotating the stud.

An insert is used in the stud hole of the reactor vessel flange. The Roto-lok lugs are machined on the inside diameter of the insert and the outside diameter is machined with standard threads. The insert is threaded into the vessel flange and locked in place by pins so that the interrupted portions of the lugs assume the same position on all bushings relative to the vessel centerline.

The closure nut threads onto the stud at the reactor vessel head. The closure nut uses a modified four-pitch centralizing acme thread form. CN-2994

The closure nut rests on a spherical washer which sits on top of the reactor closure head. This spherical washer alleviates bending of the Roto-lok stud.

A prototype Roto-lok closure system has been tested to verify this closure design. Static loads were imposed on the studs which simulated loads during operation. Results of tests performed on the Roto-lok closure system are presented in Reference 5.3-2.

The closure stud material meets the fracture toughness requirements of ASME III and 10CFR50, Appendix G. Nondestructive examinations are preformed in accordance with ASME III. Fracture toughness data for the Unit 1 and Unit 2 closure stud bolting materials is presented in Tables 5.3-5 and 5.3-6, respectively.

Westinghouse is in agreement with RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs”, except for material and tensile strength guidelines.

Westinghouse has specified both 45 ft-lb and 25 mils lateral expansion for control of fracture toughness determined by Charpy-V testing, required by the ASME B&PV Code, Section III, Summer 1973 Addenda and 10CFR50, Appendix G (July 17, 1973, Paragraph IV.A.4). These toughness requirements assure optimization of the stud bolt material tempering operation with the accompanying reduction of the tensile strength level when compared with previous ASME B&PV Code requirements.

The specification of both impact and maximum tensile strength as stated in the Guide results in unnecessary hardship in procurement of material without any additional improvement in quality.

The closure stud bolting material is procured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. This strength level is compatible with the fracture toughness requirements of 10CFR50, Appendix G (July 1973, Paragraph 1.C), although higher strength level bolting materials are permitted by the Code. Stress corrosion has not been observed in reactor vessel closure stud bolting manufactured from material of this strength level. Accelerated stress corrosion test data do exist for materials of 170,000 psi minimum yield strength exposed to marine water environments stressed to 75 percent of the yield strength (given in Reference 2 of the Guide). These data are not considered applicable to Westinghouse reactor vessel closure stud bolting because of the specified yield strength differences and a less severe environment; this has been demonstrated by years of satisfactory service experience.

The ASME B&PV Code requirement for toughness for reactor vessel bolting has precluded the Guide's additional recommendation for tensile strength limitation, since to obtain the required toughness levels, the tensile strength levels are reduced. Prior to 1972, the Code required a 35 ft-lb toughness level which provided maximum tensile strength levels ranging from approximately 155 to 178 kpsi (Westinghouse review of limited data – 25 heats). After publication of the Summer 1973 Addenda to the Code and 10CFR50, Appendix G, wherein the toughness requirements were modified to 45 ft-lb with 25 mils laterial expansion, all bolt material data reviewed on Westinghouse plants showed tensile strengths of less than 170 kpsi.

Additional protection against the possibility of incurring corrosion effects is assured by:

1. Decrease in level of tensile strength comparable with the requirement of fracture toughness as described above.

2. Design of the reactor vessel studs, nuts, and washers, allowing them to be completely removed during each refueling permitting visual and/or nondestructive inspection in parallel with refueling operations to assess protection against corrosion, as part of the inservice inspection program (Section 5.2.4).

3. Design of the reactor vessel studs, nuts and washers providing protection against corrosion by allowing them to be completely removed during each refueling and placed in storage racks on the Containment operating deck, as required by Westinghouse refueling procedures. The stud holes in the reactor flange are sealed with bolting materials to prevent exposure to the borated refueling cavity water.

4. Use of a manganese phosphate surface treatment.

   Use of Code Case 1605 does not constitute an issue between the Nuclear Regulatory Commission (NRC) and Westinghouse inasmuch as use of this Code Case has been approved by the NRC via the guideline of RG 1.85 (see Rev. 6, May 1976).

Westinghouse refueling procedures require the studs, nuts, and washers to 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.

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 core region materials of the reactor pressure vessel. Actual material property test data are used. The methods outlined in Appendix G to Section XI of the ASME Code as required by Appendix G to 10CFR50 are employed for the shell regions in the analysis of protection against nonductile failure. The initial operation curves are calculated assuming a period of reactor operation such that the beltline material will be limiting. The heat-up and cool-down curves are given 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}).

The initial predicted ΔRT_{NDT} values were derived using a curve which shows the effect of fluence and copper content on the shift of RT_{NDT} for the reactor vessel steels exposed to 550°F (Figure 5.3-3) and a curve which shows the maximum fluence at 1/4 T (thickness) and 3/4 T location (tips of the code reference flaw when the flaw is assumed at inside diameter and outside diameter locations, respectively).

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 10CFR50, Appendix G and ASME Code Section XI, Appendix G requirements. Changes in fracture toughness of the core region plates or forgings, weldments, and associated heat affected zones due to radiation damage, will be monitored by a surveillance program which conforms to ASTM E185, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" and 10CFR50, Appendix H. The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing of Charpy V-notch and tensile specimens, and post-irradiation testing of Charpy V-notch, tensile, and, when desirable, 1/2 T compact tension specimens. The post-irradiation testing will be carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core midheight and removable from the vessel at specified intervals.

The results of the radiation surveillance program were used to verify that the ΔRT_{NDT} predicted from the effects of the fluence, and copper and nickel content was appropriate. Temperature limits for preservice hydrotests and in-service leak and hydrotests will be calculated in accordance with

10CFR50, Appendix G. Surveillance capsule analyses and dosimetry results will be used to monitor the radiation effect on the reactor vessel material.

Pressure-temperature operating limits (Ref. 5.3-6) for the reactor vessel were determined by using the radiation damage trend curves based on RG 1.99, Rev. 1 as shown in Figure 5.3-2. The operating limits were reviewed with regard to RG 1.99, Rev. 1 and Rev. 2 (Ref. 5.3-12 and 5.3-14). Since the Westinghouse trend curves based on RG 1.99, Rev. 1 (Figure 5.3-2) used to generate the heat-up/cool-down curves were conservative for the RG 1.99, Rev. 2 analysis, revision to the operating limits was not necessary.

The RT_{PTS} values are well below the NRC screening criterion which is 270°F for plates, forgings and axial welds, and 300°F for circumferential welds. Therefore, the reactor vessel belt line material complies with fracture toughness requirements for protection against Pressurized Thermal Shock (PTS) events. The reactor vessel material properties are shown in Tables 5.3-3 and 5.3-4.

5.3.2.2 Operating Procedures. The transient conditions that are considered in the design of the reactor vessel are presented in Section 3.9.1.1. These transients are representative of the operating conditions that should prudently be considered to occur during plant operation. The transients selected from a conservative basis for evaluation of the RCS to ensure the integrity of the RCS equipment.

Those transients listed as upset condition transients are listed in Table 3.9-8. None of these transients will result in pressure-temperature changes which exceed the heatup and cooldown limitations as described in Section 5.3.2.1.

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 (Figure 5.3-1). 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-ring. For Unit 1, a 1/8 in. threaded plug made of stainless steel has been installed and seal welded in the reactor vessel flange drain hole for the inner O-ring leakoff line. The vessel contains the core, core support structures, control rods, and the parts directly associated with the core. Each CRDM assembly is attached directly to a CRDM head penetration via a full- penetration bimetallic weld. Inlet and outlet nozzles are located symmetrically around the vessel. Outlet nozzles are arranged on the vessel to facilitate optimum layout of the RCS equipment. Inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop. The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear in-core instrumentation. Each nozzle consists of a tubular member made of an Inconel tube. Each tube is attached to the inside of the bottom head by a partial penetration weld. CN-2994

Internal surfaces of the vessel which are in contact with primary coolant are weld overlay with 5/32 in. minimum of stainless steel or Inconel. The exterior of the reactor vessel is insulated with Dwight canned stainless steel reflective sheets. The insulation is a minimum of 3 in. thick and contoured to enclose the top, sides, and bottom of the vessel. All insulation modules are removable but the access to vessel side insulation is limited by the surrounding concrete.

The reactor vessel is designed and fabricated in accordance with the requirements of ASME Section III.

Principal design parameters of the reactor vessel are given in Table 5.3-2.

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 period of approximately one week would be applied. Various modes of heating may be used depending on the temperature.

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.

Cyclic loads are introduced by normal power changes, reactor trip, startup, and shutdown operations. These design basis cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life. Fatigue analyses of the vessel are performed in order to assure that the usage factor is less than the ASME Code allowable of 1.0.

The design specifications require analysis to prove that the vessel is in compliance with the fatigue and stress limits of ASME III. The loadings and transients specified for the analysis are based on all conditions expected during service. The heat-up and cool-down rate imposed by plant operating limits are reflected in the vessel design specifications and are communicated in the Technical Specifications.

5.3.3.2 Materials of Construction. The materials used in the fabrication of the reactor vessel are discussed in Section 5.2.3.

5.3.3.3 Fabrication Methods. The fabrication methods used in the construction of the reactor vessel are discussed in Section 5.3.1.2.

5.3.3.4 Inspection Requirements. The nondestructive examinations performed on the reactor vessel are described in Section 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 are painted with a heat-resistant paint before shipment, except for the vessel support surfaces and the slide surface of the external seal ring.

The closure head is also shipped with a shipping cover and skid. An enclosure attached to the replacement head flange protects the control rod mechanism housings. All head openings are sealed

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to prevent the entrance moisture and an adequate quantity of desiccant bags is placed inside the head. A lifting frame is provided for handling the vessel head.

5.3.3.6 Operating Conditions. Operating limitations are presented in Section 5.3.2 and area also presented in the Technical Specifications. The procedures and methods used to ensure the integrity of the reactor vessel under the most severe postulated conditions are described in Section 3.9.1.4. In addition, 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 (LOCA) produces relatively high thermal stresses in regions on 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. CN-3050

For the beltline region, significant developments have recently occurred in order to address PTS events. On the basis of deterministic and probabilistic studies, taking U.S. PWR operating experience into account, the NRC staff has concluded that conservatively calculated screening criterion 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 criteria in the PTS Rule for 10CFR50.34 (new plants) and 10CFR50.61 (operating plants) (Ref. 5.3-9). The conservative methods chosen by the NRC Staff for the calculation of RT_{PTS} for the purpose of comparison with the screening criteria is presented in paragraph (b) (2) of 10CFR50.61. Details of the analysis method and the basis for the PTS Rule can be found in SECY-82-465 (Ref. 5.3-10).

The reactor vessel beltline materials are specified in Section 5.2.3. Based upon the dosimetry results from removed surveillance capsules, the fluence at the vessel inner radius at 32 effective full-power years (EFPY) was determined and used to calculate all of the RT_{PTS} values. RT_{PTS} is RT_{NDT}, the reference nil-ductility transition temperature as calculated by the method chosen by the NRC Staff as presented in paragraph (b) (2) of 10CFR50.61, the "PTS Rule". The PTS Rule states that this method of calculation RT_{PTS} should be used in reporting values used to be compared to the above screening criterion set in the PTS Rule. The screening criteria will not be exceeded using the method of calculation prescribed by the PTS Rule for the vessel design lifetime. The material properties, initial RT_{NDT}, and end-of-life RT_{PTS} values are in Tables 5.3-7 and 5.3-8. The materials identified in Tables 5.3-7 and 5.3-8 are those materials that are exposed to high fluence levels at the beltline region of the reactor vessel and are, therefore, the subject of the PTS Rule.

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

The closure head is examined visually during selected refueling. Optical devices permit a selective inspection of the cladding, CRDM nozzles, and the gasket seating surface. The closure studs can be inspected periodically using visual, magnetic particle or dye penetrant. CN-2994

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

1. Vessel shell – from the inside surface

2. Primary coolant nozzles – from the inside surface

3. Replacement closure head - from the inside and outside surfaces, with the exception of the full penetration welds between the rod travel housings and the CRDM pressure housings, which are not accessible from the inside due to the CRDM internals. CN-2994
   Bottom head – from the outside surface

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

The design considerations which have been incorporated into the system design to permit the above inspection are as follows:

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

2. The closure head can be disassembled from the upper package assembly and can be stored dry on the reactor operating deck to facilitate direct visual inspection.

3. All reactor vessel studs, nut, and washers can be removed to dry storage during refueling.

4. Removable plugs are provided in the primary shield. The insulation covering the nozzle welds may be removed.

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

1. Shop ultrasonic examinations are performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal from inside surface. The size of cladding bonding defect allowed is 1/4 in. by 3/4 in. with the greater direction parallel to the weld in the region bounded by 2T (T = wall thickness) on both sides of each full penetration pressure boundary weld. Unbounded areas exceeding 0.422 in.^{2} (3/4-inch-diameter) in all other regions are rejected.

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

3. The weld deposited clad surface on both sides of the welds to be inspected is specifically prepared to assure meaningful ultrasonic examinations.

4. During fabrication, all full penetration pressure boundary welds are ultrasonically examined in addition to Code examinations.

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

The vessel design and construction enables inspection in accordance with ASME Section XI.

REFERENCES

Section 5.3:

5.3-1 Not used.

5.3-2 "Roto-lok Closure System Development", WCAP-8447, December 1974 (Westinghouse Proprietary) and WCAP-8448, December 1974 (Non-Proprietary).

5.3-3 Not used.

5.3-4 Not used.

5.3-5 Not used.

5.3-6 Westinghouse RESAR-38, Chapter 16, Figure B3/4 4.2, Page B3/4 4-8.

5.3-7 Not used.

5.3-8 Not used.

5.3-9 PTS Rule, Federal Register Volume 50, No. 141, July 23, 1985, 10CFR50.34.

5.3-10 NRC Policy Issue, "Pressurized Thermal Shock", SECY-82-465, November 23, 1982.

5.3-11 Buchalet, C., Bamford, W. H., "Method for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients", WCAP-8510, July 1976.

5.3-12 Regulatory Guide 1.99, Revision 1. "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials", April 1977.

5.3-13 Not used.

5.3-14 Regulatory Guide 1.99, Rev. 2. "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

5.3-15 WCAP 14849, "Evaluation of Pressurized Thermal Shock for South Texas Unit 1."

5.3-16 WCAP 14980, "Evaluation of Pressurized Thermal Shock for South Texas Unit 2."

5.3.17 Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

5.3.18 WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," S. L. Anderson, August 2000.

TABLE 5.3-1

REACTOR VESSEL QUALITY ASSURANCE PROGRAM

Forgings	RT*	UT	PT	MT
1. Studs, Nuts		yes		yes
2. Instrumentation Tube		yes	yes	
3. Main Nozzles		yes		yes
4. Nozzle Safe Ends		yes	yes	
5. Head		yes	yes	yes
Weldments				
1. Main Seam	yes	yes		yes
2. Instrumentation Tube Connection			yes	
3. Main Nozzle	yes	yes		yes
4. Cladding		yes	yes	
5. Nozzle Safe Ends (if forging)	yes	yes	yes	
6. Nozzle Safe Ends (if weld deposit)	yes	yes	yes	
7. All Ferritic Welds Accessible After Hydrotest		yes		yes
8. All Nonferritic Welds Accessible After Hydrotest		yes	yes	
9. Seal Ledge				yes
10. Core Pad Welds			yes	
11. Upper Internal Support Housing	yes	yes	yes	
12. Guide Stud Support Blocks		yes	yes	yes

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*RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle

TABLE 5.3-2

REACTOR VESSEL DESIGN PARAMETERS

Design/Operating Pressure, psig	2,485/2,317
Design Temperature, °F	650
Height of Replacement Head from Mating Surface to Top of CRDM Housing ft-in.	13-0
Thickness of Insulation, Minimum, in.	3
Number of Reactor Closure Head Studs	36
Diameter of Reactor Closure Head/Studs, in. (minimum shank)	8.188
Inside Diameter of Vessel Flange, in.	219.25
Outside Diameter at Head Flange, in.	216.31
Inside Diameter at Shell, in.	173
Inlet Nozzle Inside Diameter, in.	27-1/2
Outlet Nozzle Inside Diameter, in.	29
Clad Thickness, Minimum, in.	1/8
Lower Head Thickness, Minimum, in.	5-3/8
Vessel Belt Line Thickness, Minimum, in.	8-3/8
Closure Head Thickness, in.	7.19 nominal

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TABLE 5.3-3

SOUTH TEXAS UNIT 1 REACTOR VESSEL MATERIAL PROPERTIES

									Average Upper Shelf Energy	
Component	Code No.	Grade	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	50 ft-lb 35 mil Temp (°F)	RT_{NDT} (°F)	Normal to Principal Working Direction (ft-lb)	Principal Working Direction (ft-lb)
Closure Head Forging	07W38-1-1	SA508 CL.3	0.04	0.004	.87	-44	16	-44	221	-
Vessel Flange	R1601-1	"	0.02	0.017	.75	-10	<50	-10	160.5	-
Inlet Nozzle	R1613-1	"	-	0.009	.80	-10	<50	-10	140	-
Inlet Nozzle	R1613-2	"	-	0.013	.82	0	<60	0	130.5	
Inlet Nozzle	R1613-3	"	0.09	0.009	.79	-20	<40	-20	175	-
Inlet Nozzle	R1613-4	"	-	0.006	.85	20	<80	20	128	-
Outlet Nozzle	R1614-1	"	-	0.006	.66	10	<70	10	106	-
Outlet Nozzle	R1614-2	"	-	0.006	.71	0	<60	0	114	-
Outlet Nozzle	R1614-3	"	-	0.009	.69	-30	<30	-30	129	-
Outlet Nozzle	R1614-4	"	-	0.006	.90	10	70	10	118	-
Nozzle Shell	R1607-1	SA533B CL.1	0.08	0.012	.62	0	110	50	89	-
Nozzle Shell	R1607-2	"	0.08	0.012	.66	-20	110	50	85	-
Nozzle Shell	R1607-3	"	0.07	0.010	.60	-50	90	30	82	
Inter. Shell	R1606-1	"	0.04	0.009	.63	-40	70	10	109.5	130
Inter. Shell	R1606-2	"	0.04	0.008	.61	-20	60	0	94	119
Inter. Shell	R1606-3	"	0.05	0.007	.62	-20	70	10	105.5	132
Lower Shell	R1622-1	"	0.05	0.006	.61	-30	30	-30	111	143
Lower Shell	R1622-2	"	0.07	0.006	.64	-30	30	-30	122	149
Lower Shell	R1622-3	"	0.05	0.007	.66	-30	30	-30	127	148
Bottom Head Torus	R1617-1	"	0.14	0.012	.67	-50	<10	-50	143	-
Bottom Head Dome	R1618-1	"	0.08	0.015	.67	-50	<10	-50	128	-
Inter. and Lower Shells Long, Seam Welds	G1.70	Sub-Arc	0.02**	0.004	.07**	-50	<10	-50	158*	-
Inter. to Lower Shell Girth Weld	E3.13	"	0.02**	.007	.07**	-70	<10	-70	100*	

* Normal to Principal welding direction

** Based on CE NPSD-1039 Rev. 1, "Best Estimate Copper And Nickel in CE Fabricated Reactor Vessel Welds."

TABLE 5.3-4

SOUTH TEXAS UNIT 2 REACTOR VESSEL MATERIAL PROPERTIES

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Component	Code No.	Grade	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	50 ft-lb 35 mil Temp . (°F)	RT_{NDT} (°F)	Average Upper Shelf NMWD (ft-lb)	Average Upper Shelf MWD (ft-lb)
Closure Head Forging	07W70-1-1	SA508 CL.3	0.04	0.0004	.87	-44	16	-44	221	-
Vessel Flange	R3001-1	"	0.04	0.008	.67	-10	<50	-10	146	
Inlet Nozzle	R2011-1	"	0.10	0.011	.81	-40	<20	-40	165	-
Inlet Nozzle	R2011-2	"	0.10	0.011	.84	-20	<40	-20	136	-
Inlet Nozzle	R2011-3	"	0.12	0.009	.84	-20	<40	-20	128	-
Inlet Nozzle	R2011-4	"	0.11	0.009	.83	-20	<40	-20	132	-
Outlet Nozzle	R2012-1	"	-	0.006	.72	10	<70	10	132	-
Outlet Nozzle	R2012-2	"	-	0.007	.68	10	<70	10	132	-
Outlet Nozzle	R2012-3	"	-	0.004	.67	0	<60	0	121	-
Outlet Nozzle	R2012-4	"	-	0.007	.68	10	<70	10	126	-
Nozzle Shell	R2505-1	SA533B CL.1	0.05	0.009	.66	-40	60	0	114	-
Nozzle Shell	R2505-2	"	0.07	0.008	.64	-30	60	0	124	-
Nozzle Shell	R2505-3	"	0.05	0.008	.65	-50	50	-10	127	-
Inter. Shell	R2507-1	"	0.04	0.006	.65	-10	<50	-10	109	137
Inter. Shell	R2507-2	"	0.05	0.006	.64	-10	<50	-10	129	145
Inter. Shell	R2507-3	"	0.05	0.005	.61	-40	20	-40	122	149
Lower Shell	R3022-1	"	0.03	0.002	.63	-30	30	-30	124	141
Lower Shell	R3022-2	"	0.04	0.003	.61	-40	20	-40	118	141
Lower Shell	R3022-3	"	0.04	0.004	.60	-40	20	-40	123	126
Bottom Head Torus	R3020-1	"	0.11	0.009	.65	-30	100	40	86	-
Bottom Head Dome	R3021-1	"	0.09	0.008	.64	-60	0	-60	132	-
Inter. Shells Long, Seam Welds	G3.02	Sub Arc	0.04**	0.004	.11**	-70	<10	-70	146	-
Lower Shell Long. Seam Welds and Inter. to Lower Girth Welds	E3.12	Sub Arc	0.04**	.008	.11**	-70	<10	-70	101	-

** Based on CE NPSD-1039 Rev. 2, "Best Estimate Copper And Nickel in CE Fabricated Reactor Vessel Welds." Sample weighted mean.

TABLE 5.3-5

SOUTH TEXAS UNIT 1 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-3928	A540, B24	5F-740	132.5	150.5	19.0	62.5	341	62-66-64	32-33-34
EVD-3928	"	"	135.0	150.5	19.0	63.5	341	71-68-72	44-41-45
EVD-4025	"	5G-887	137.0	153.5	18.0	55.0	341	60-60-55	32-33-31
EVD-4025	"	"	141.0	154.0	23.0	59.0	341	59-60-60	33-32-33
EVD-3927	"	5F-735	135.0	150.0	19.5	63.5	321	73-72-70	42-43-42
EVD-3927	"	"	140.0	156.0	18.5	61.5	321	72-71-74	42-40-43
EVD-3927	"	5F-736	143.5	158.0	18.0	61.5	311	70-70-65	36-38-33
EVD-3927	"	"	144.0	158.5	18.0	60.0	321	70-68-68	34-34-36
EVD-3884	"	5G-535	138.0	152.0	19.0	65.0	311	60-60-53	36-30-31
EVD-3884	"	"	141.0	156.0	18.5	65.0	321	54-60-58	27-26-30
EVD-5038	"	6D-816	156.5	168.5	17.0	59.0	363	57-59-60	32-33-37
EVD-5038	"	"	155.5	166.0	17.0	60.0	363	54-54-55	27-27-28
EVD-4148	"	5H-947	134.0	148.5	18.5	62.5	321	82-77-78	50-47-48
EVD-4148	"	"	137.0	152.5	18.0	62.5	321	75-72-69	45-43-40
EVD-4024	"	5H-626	130.0	147.5	18.0	59.0	321	76-78-75	45-46-42
EVD-4024	"	"	137.0	153.0	20.0	62.5	321	73-74-73	43-45-42
EVD-4025	"	5G-886	131.0	145.0	20.0	60.0	311	73-70-78	46-42-43
EVD-4025	"	"	130.5	146.0	20.0	59.0	311	77-70-72	44-42-42
EVD-3925	"	5G-539	143.0	159.0	18.0	60.0	341	62-64-60	33-36-35
EVD-3925	"	"	136.0	152.0	18.0	59.0	341	62-58-55	32-34-36

TABLE 5.3-5 (Continued)

SOUTH TEXAS UNIT 1 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-4024	A540, B24	5H-627	145.0	160.0	18.0	62.5	341	66-70-66	38-40-38
EVD-4024	"	"	147.0	159.0	18.0	60.0	341	61-65-60	30-33-30
EVD-4025	"	5G-885	141.5	157.0	18.5	59.0	321	59-58-60	34-30-32
EVD-4025	"	"	143.5	157.0	23.0	61.5	321	65-64-62	38-38-36
EVD-4024	"	5H-629	139.5	156.0	18.5	62.5	321	72-71-75	41-41-43
EVD-4024	"	"	138.0	152.0	18.0	61.5	321	76-73-71	44-42-40
EVD-4346	"	5J-993	141.0	155.0	19.0	62.5	352	67-67-65	39-37-36
EVD-4346	"	"	141.0	154.5	19.0	62.5	363	55-56-53	27-31-26
EVD-4646	"	5J-995	145.0	160.5	17.0	56.0	352	55-58-58	30-34-35
EVD-4346	"	"	143.0	161.0	18.0	59.0	363	55-58-58	32-36-37
EVD-4347	"	5J-997	139.0	153.5	19.0	61.5	363	66-62-67	46-43-48
EVD-4347	"	"	147.0	163.0	18.0	57.5	363	62-60-58	46-42-40
EVD-4348	"	5J-1004	145.0	160.0	18.0	60.0	341	63-65-68	33-36-38
EVD-4348	"	"	142.0	157.0	19.0	60.0	341	61-63-63	32-34-33
EVD-4353	"	5J-1006	132.0	148.0	18.0	62.5	331	61-62-60	30-31-30
EVD-4353	"	"	131.0	149.5	19.0	59.0	352	48-47-47	30-30-30
EVD-4148	"	5H-948	137.5	155.5	18.0	60.0	341	63-47-52	38-27-28
EVD-4148	"	"	139.0	154.0	18.0	60.0	321	70-65-62	39-35-37
EVD-4260	"	5J-680	139.0	154.0	19.5	61.5	341	62-61-60	31-30-30
EVD-4260	"	"	134.0	147.0	19.5	65.0	341	72-74-72	42-44-44
EVD-3927	"	5F-738	135.0	153.5	18.0	59.0	321	72-70-70	35-36-36

TABLE 5.3-5 (Continued)

SOUTH TEXAS UNIT 1 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-3927	A540, B24	"	147.0	162.0	18.0	60.0	321	64-63-63	34-31-30
EVD-3884	"	5G-536	133.0	150.0	20.0	63.5	311	78-76-73	50-49-42
EVD-3884	"	"	136.0	151.0	18.0	60.0	321	69-74-68	37-36-37
EVD-4147	"	5H-944	139.0	154.0	18.0	62.5	321	65-62-65	37-34-43
EVD-4147	"	"	134.5	148.5	19.0	65.0	321	72-76-77	42-40-44
EVD-4148	"	5H-945	132.5	149.0	19.5	62.5	321	58-61-70	36-26-33
EVD-4148	"	"	134.0	150.0	20.0	62.5	321	65-67-68	36-38-40
EVD-4148	"	5H-946	134.0	151.0	18.5	61.5	341	57-61-51	29-35-33
EVD-4148	"	"	136.0	151.0	19.0	63.5	341	75-70-73	40-36-39
EVD-4148	"	5H-949	132.5	146.0	18.5	62.5	321	72-74-77	50-55-58
EVD-4148	"	"	140.0	153.5	18.0	62.5	341	47-47-48	29-31-27
EVD-3927	"	5F-734	140.0	153.5	19.0	61.5	341	69-71-68	41-43-38
EVD-3927	"	"	147.0	162.0	18.0	56.0	341	62-63-64	36-33-36
EVD-3927	"	5F-737	134.0	148.0	19.0	62.5	331	80-85-82	41-45-42
EVD-3927	"	"	146.0	161.0	17.5	59.0	341	62-61-58	42-38-32
EVD-3884	"	5G-534	130.0	146.5	19.5	65.0	341	72-75-75	34-36-36
EVD-3884	"	"	131.0	148.0	18.5	60.0	341	57-58-54	30-33-31
EVD-4025	"	5G-888	139.0	156.0	17.0	57.5	331	52-65-60	28-38-34
EVD-4025	"	"	140.0	157.0	18.0	56.0	331	59-62-63	28-32-33
EVD-4024	"	5H-628	133.0	149.0	20.5	60.0	341	65-70-70	36-44-42
EVD-4024	"	"	134.0	150.0	20.0	60.0	341	65-70-68	35-41-40

TABLE 5.3-5 (Continued)

SOUTH TEXAS UNIT 1 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-4353	A540, B24	5J-1007	136.0	152.0	18.0	60.0	352	55-60-59	28-30-26
EVD-4353	"	"	137.0	154.0	18.0	60.0	363	60-58-60	27-26-31
EVD-4353	"	5J-1013	143.0	157.0	18.0	60.0	331	60-60-59	42-41-39
EVD-4353	"	"	140.0	155.0	18.0	60.0	341	62-63-58	44-46-35
EVD-4354	"	5J-1014	134.5	150.5	18.0	60.0	341	68-71-66	38-43-37
EVD-4354	"	"	144.0	158.5	18.0	59.0	363	60-61-60	32-34-32
EVD-4354	"	5J-1012	137.0	152.0	19.0	62.5	352	70-72-68	44-46-38
EVD-4354	"	"	142.0	158.0	17.5	60.0	363	62-65-63	32-34-34
EVD-4356	"	5J-1092	142.5	157.5	17.0	56.0	341	54-56-56	31-34-32
EVD-4356	"	"	132.5	148.0	18.5	56.0	341	66-65-68	36-34-39
EVD-4356	"	5J-1095	139.0	155.0	18.0	55.0	321	55-58-59	32-34-35
EVD-4356	"	"	135.0	151.0	18.0	56.0	331	62-64-62	38-39-36
EVD-4079	"	5J-1089	143.5	156.0	18.0	55.0	341	51-53-51	28-26-26
EVD-4079	"	"	142.0	157.5	18.0	55.0	341	51-52-53	29-34-37
EVD-5389	"	6L-276	137.0	152.0	18.5	62.5	331	50-53-48	30-38-33
EVD-5389	"	"	132.5	148.5	23.5	62.5	331	65-67-61	36-38-35
EVD-3928	"	5F-742	135.0	151.0	19.0	57.5	321	58-50-55	29-27-29
EVD-3928	"	"	139.0	155.0	20.0	62.5	331	56-46-52	32-26-33
EVD-3925	"	5G-540	133.0	149.5	19.0	60.0	311	50-58-63	31-32-42
EVD-3925	"	"	140.0	154.5	19.0	60.0	311	57-51-66	32-30-47

TABLE 5.3-5 (Continued)

SOUTH TEXAS UNIT 1 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
				CLOSURE HEAD INSERTS					
EVD-3926	A540, B24	5F-744	137.5	152.5	18.0	62.5	341	74-72-75	44-42-42
EVD-3926	"	"	134.0	146.0	20.0	60.0	341	71-60-57	43-37-32
EVD-4021	"	6F-665	153.5	165.5	19.0	59.0	352	61-59-62	34-32-35
EVD-4021	"	"	153.0	163.0	18.0	55.0	352	58-60-61	32-34-34
EVD-3945	"	56-531	142.5	156.0	19.0	60.0	352	60-60-61	35-36-38
EVD-3945	"	"	142.0	157.0	18.5	57.5	341	72-71-71	44-44-43
EVD-3945	"	5G-530	137.0	152.0	18.0	61.5	341	68-67-71	39-38-44
EVD-3945	"	"	143.0	157.5	19.0	62.5	341	65-68-65	37-39-38
EVD-3945	"	5G-532	144.0	156.0	19.0	60.0	352	64-66-63	31-30-33
EVD-3945	"	"	138.0	152.0	19.0	63.5	321	70-73-68	37-38-35
EVD-4026	"	5G-882	134.5	148.0	19.0	62.5	341	76-78-76	41-40-40
EVD-4026	"	"	137.0	149.0	19.0	62.5	341	70-75-72	36-38-38
EVD-4026	"	5G-883	146.5	157.0	18.0	62.5	352	68-68-69	36-36-38
EVD-4026	"	"	143.5	155.0	18.5	60.0	321	66-67-69	34-36-38
EVD-4026	"	5G-884	138.0	149.5	19.5	62.5	341	74-75-80	39-40-40
EVD-4026	"	"	142.0	153.5	20.0	62.5	331	72-70-72	40-38-40
EVD-4355	"	5K-774	154.0	166.0	18.0	62.5	352	49-54-55	28-30-31
EVD-4355	"	"	157.0	168.0	18.0	55.0	363	47-49-49	28-30-31
EVD-5038	"	6D-729	151.0	161.0	18.0	57.5	341	62-61-63	40-39-39
EVD-5038	"	"	154.0	163.0	18.0	60.0	341	58-57-55	35-33-30

TABLE 5.3-5 (Continued)

SOUTH TEXAS UNIT 1 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-3926	A540, B24	5F-745	148.5	161.0	19.0	62.5	363	62-65-62	36-38-35
EVD-3926	"	"	147.5	161.0	18.0	61.5	341	68-65-72	38-37-39
EVD-3926	"	5F-746	146.5	160.0	18.0	61.5	341	68-64-67	40-36-38
EVD-3926	"	"	149.0	161.0	19.0	62.5	363	65-66-68	34-35-39
EVD-2142	"	5I-753	131.5	147.5	18.5	55.0	331	55-58-55	34-35-33
EVD-2142	"	"	132.5	174.5	18.0	53.5	311	56-59-57	35-50-35
EVD-4661	"	5L-768	151.0	161.0	18.0	60.0	341	64-62-61	40-40-38
EVD-4661	"	"	153.0	164.0	18.0	56.0	341	60-60-59	37-35-33
EV-4601	"	6D-982	155.0	167.0	16.0	52.0	363	49-50-48	35-29-30
EV-4601	"	"	154.0	164.0	18.0	55.0	341	50-49-50	30-29-30
EV-2222	"	5E-1226	144.0	155.0	18.5	55.0	352	64-62-64	36-36-37
EV-2222	"	"	149.5	163.0	18.0	52.0	341	54-53-55	27-32-34
EV-2229	"	5E-1225	134.0	148.0	18.5	55.0	331	63-64-64	36-35-37
EV-2229	"	"	147.0	160.0	18.0	56.0	321	54-53-55	36-37-38
EV-2232	"	5F-582	135.5	151.0	18.0	52.0	311	56-54-55	31-30-32
EV-2232	"	"	143.5	157.0	17.5	53.5	341	60-58-60	32-34-33
EV-3954	"	5I-1099	141.5	153.0	20.0	62.5	321	70-72-70	40-41-40
EV-3954	"	"	144.5	154.0	18.0	59.0	321	70-68-65	40-37-36

TABLE 5.3-6

SOUTH TEXAS UNIT 2 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-4260	A540, B24	5J-676	133.0	147.0	20.0	63.5	311	63-61-68	39-40-40
"	"	"	135.0	152.0	18.0	60.0	321	59-65-58	35-33-34
EVD-4347	"	5J-999	133.0	150.0	20.5	63.5	341	65-70-71	38-39-41
"	"	"	139.0	154.0	18.0	62.5	341	66-58-65	39-34-38
EVD-4538	"	5K-1029	138.0	152.0	19.0	67.0	321	63-60-62	38-35-38
"	"	"	134.0	147.0	20.0	63.5	341	56-62-53	32-38-31
"	"	5K-1027	137.0	153.0	18.5	62.5	341	61-46-50	28-26-36
"	"	"	135.0	148.0	19.0	63.5	363	74-78-80	46-50-49
EVD-4356	"	5J-1091	134.0	152.0	18.0	57.5	321	60-61-60	40-37-34
"	"	"	134.0	146.0	18.5	56.0	321	68-67-69	40-40-49
EVD-5038	"	6D-814	137.0	151.0	19.0	61.5	341	74-72-74	44-43-45
"	"	"	137.5	152.0	18.0	60.0	341	70-68-70	38-39-40
EVD-4146	"	5I-1168	143.0	157.0	18.0	60.0	341	70-70-70	40-41-40
"	"	"	147.0	161.0	18.0	57.5	321	70-65-65	40-34-35
"	"	5I-1169	144.0	154.0	18.0	56.0	311	61-65-64	39-38-38
"	"	"	140.0	155.0	18.0	57.5	311	65-68-70	40-41-43
EVD-4347	"	5J-996	132.0	150.0	19.0	61.5	311	72-74-75	42-43-46
"	"	"	137.0	153.0	19.5	61.5	311	65-60-65	40-34-36
EVD-4079	"	5J-1090	145.0	160.5	16.5	52.0	352	55-56-55	31-32-31
"	"	"	144.0	162.0	17.5	55.0	341	55-53-52	31-30-30
EVD-4356	"	5J-1093	138.0	153.5	18.5	57.5	311	60-58-62	39-36-40

TABLE 5.3-6 (Continued)

SOUTH TEXAS UNIT 2 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EV-4356	A540, B24	"	136.5	152.0	18.0	56.0	331	62-62-63	40-39-38
"	"	5J-1094	142.0	162.5	17.0	55.0	341	56-52-54	48-43-45
"	"	"	141.0	159.5	18.0	55.0	341	56-55-57	44-44-46
EVD-4146	"	5I-1167	140.0	156.0	19.0	60.0	331	73-74-74	45-44-44
"	"	"	143.0	159.0	18.5	61.5	341	70-71-70	40-42-40
EVD-4260	"	5J-677	138.0	154.0	21.0	62.5	341	58-52-68	32-32-40
"	"	"	146.0	160.0	18.5	62.5	341	64-61-63	37-35-37
EVD-4346	"	5J-994	135.0	151.0	19.0	65.0	321	70-72-70	46-44-44
"	"	"	138.0	153.0	17.0	57.5	331	63-63-65	35-38-41
EVD-4661	"	5L-999	144.0	157.0	18.0	60.0	321	60-63-61	34-36-36
"	"	"	155.0	166.5	18.0	59.0	331	58-56-57	36-34-34
"	"	5L-1000	146.0	161.0	18.0	60.0	321	63-60-55	33-34-32
"	"	"	149.0	161.0	18.0	61.5	341	56-57-56	32-35-30
EVD-4348	"	5J-1002	132.0	149.0	19.0	62.5	311	64-62-65	36-34-37
"	"	"	131.0	147.0	19.5	66.0	321	80-81-85	47-48-50
EVD-4353	"	5J-1010	132.0	149.0	19.0	61.5	311	74-76-75	44-49-45
"	"	"	134.0	149.0	19.0	65.0	311	67-69-66	38-39-37
EVD-4260	"	5J-678	139.0	153.0	20.0	61.5	321	70-65-68	41-37-35
"	"	"	147.0	159.5	18.0	61.5	321	69-67-63	40-39-37
EV-4356	"	5J-1093	138.0	153.5	18.5	57.5	311	60-58-62	39-36-40
"	"	"	136.5	152.0	18.0	56.0	331	62-62-63	40-39-38

TABLE 5.3-6 (Continued)

SOUTH TEXAS UNIT 2 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EV-4356	A540, B24	5J-1094	142.5	162.5	17.0	55.0	341	56-52-54	48-43-45
"	"	"	141.0	159.5	18.0	55.0	341	56-55-57	44-44-46
EVD-4146	"	5I-1167	140.0	156.0	19.0	60.0	331	73-74-74	45-44-44
"	"	"	143.0	159.0	18.5	61.5	341	70-71-70	40-42-40
EVD-4260	"	5J-677	138.0	154.0	21.0	62.5	341	58-52-68	32-32-40
"	"	"	146.0	160.0	18.5	62.5	341	64-61-63	37-35-37
EVD-4346	"	5J-994	135.0	151.0	19.0	65.0	321	70-72-70	46-44-44
"	"	"	138.0	153.0	17.0	57.5	331	63-63-65	35-38-41
EVD-4661	"	5L-999	144.0	157.0	18.0	60.0	321	60-63-61	34-36-36
"	"	"	155.0	166.5	18.0	59.0	331	58-56-57	36-34-34
"	"	5L-1000	146.0	161.0	18.0	60.0	321	63-60-55	33-34-32
"	"	"	149.0	161.0	18.0	61.5	341	56-57-56	32-35-30
EVD-4348	"	5J-1002	132.0	149.0	19.0	62.5	311	64-62-65	36-34-37
"	"	"	131.0	147.0	19.5	66.0	321	80-81-85	47-48-50
EVD-4353	"	5J-1010	132.0	149.0	19.0	61.5	311	74-76-75	44-49-45
"	"	"	134.0	149.0	19.0	65.0	311	67-69-66	38-39-37
EVD-4260	"	5J-678	139.0	153.0	20.0	61.5	321	70-65-68	41-37-35
"	"	"	147.0	159.5	18.0	61.5	321	69-67-63	40-39-37
EVD-4347	"	5J-1000	133.0	150.0	22.0	63.5	321	85-86-83	43-44-48
"	"	"	142.0	156.5	19.0	61.5	321	65-60-63	49-45-47
	"								

TABLE 5.3-6 (Continued)

SOUTH TEXAS UNIT 2 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-4354	A540, B24	5J-1015	138.5	153.5	19.0	61.5	311	50-46-52	26-28-28
"	"	"	136.0	153.5	18.0	62.5	311	49-45-53	33-30-36
EVD-4538	"	5L-998	131.5	149.0	19.5	63.5	321	55-72-50	32-44-34
"	"	"	131.0	148.0	20.0	65.0	321	55-60-57	31-36-32
EVD-4661	"	6E-859	132.5	148.0	19.0	62.5	321	70-72-76	44-43-45
"	"	"	141.0	160.0	17.0	57.5	341	55-56-56	32-32-33
EVD-4348	"	5J-1001	143.0	158.5	18.0	60.0	331	63-65-65	34-36-36
"	"	"	143.0	160.0	18.0	60.0	352	54-53-52	28-27-27
"	"	5J-1003	143.5	158.0	18.0	62.5	341	66-67-62	35-36-32
"	"	"	152.0	165.5	18.0	59.0	341	54-54-55	29-31-32
EVD-4353	"	5J-1008	140.0	155.0	19.0	62.5	341	60-58-60	27-26-31
"	"	"	143.0	158.0	18.5	61.5	352	58-61-62	30-31-33
EVD-4538	"	5K-1028	131.5	146.0	20.5	66.0	311	60-51-55	36-30-33
"	"	"	135.0	150.0	19.0	62.5	321	65-71-63	33-35-36
EVD-3884	"	5G-537	130.5	150.0	20.0	62.5	341	72-70-80	42-44-48
"	"	"	140.0	154.0	18.0	59.0	341	71-54-70	39-36-44
EVD-3925	"	5G-542	136.5	152.5	18.0	57.5	321	46-47-48	34-29-36
"	"	"	137.0	153.0	18.0	60.0	331	76-73-72	40-44-42
EVD-4260	"	5J-679	141.0	156.5	18.0	60.0	341	52-48-50	33-30-32
"	"	"	139.0	154.0	19.0	61.5	341	51-55-52	32-33-38
EVD-3925	"	5G-541	146.5	161.0	17.0	56.0	321	64-60-66	40-38-41

TABLE 5.3-6 (Continued)

SOUTH TEXAS UNIT 2 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-3925	A540,B24	5G-541	133.5	151.0	18.0	59.0	321	46-47-53	31-32-37
EVD-5038	"	6D-817	157.0	169.5	17.0	59.0	341	56-55-56	32-33-35
"	"	"	158.0	169.0	18.0	59.0	341	55-56-56	34-35-35
EVD-4147	"	54-940	131.5	150.0	19.0	62.5	341	73-70-70	38-33-34
"	"	"	135.5	152.0	18.0	60.0	341	76-48-54	44-29-34
EVD-4538	"	5L-756	147.0	163.0	18.0	60.0	341	67-66-68	33-32-33
"	"	"	149.0	164.0	19.0	66.5	341	63-62-61	33-32-31
EVD-5038	"	6D-815	135.0	151.0	19.0	62.5	321	72-75-72	42-43-43
"	"	"	139.0	154.0	19.0	63.5	331	68-72-75	40-42-44

CLOSURE HEAD INSERTS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-4215	A540,B24	5I-578	135.0	152.0	18.0	59.0	331	72-20-68	46-42-40
"	"	"	143.5	155.0	18.0	59.0	321	66-67-70	40-41-43
EVD-4345	"	5I-1008	149.5	161.0	18.0	59.0	321	63-65-63	39-36-35
"	"	"	144.0	156.0	17.0	55.0	321	70-67-70	41-38-41
"	"	5I-1089	145.0	156.0	17.0	55.0	341	66-68-65	40-43-41
"	"	"	149.5	161.0	17.5	59.0	341	65-68-65	36-39-37
EVD-4345	"	5I-1090	150.0	162.0	18.0	59.0	363	65-64-65	38-37-37
"	"	"	152.0	163.0	18.0	59.0	341	65-64-63	39-37-35
EVD-4355	"	5J-1087	131.0	146.5	20.0	60.0	352	60-63-63	34-35-35
"	"	"	135.0	149.0	19.0	59.0	341	60-55-59	34-31-35

TABLE 5.3-6 (Continued)

SOUTH TEXAS UNIT 2 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-4215	A540, B24	5I-577	147.5	162.0	18.0	61.5	331	70-69-65	36-35-38
"	"	"	141.5	156.5	18.5	61.5	341	68-65-70	35-36-38
EVD-4441	"	5J-1085	144.0	156.0	18.0	57.5	363	70-69-70	38-36-38
"	"	"	147.5	159.5	17.0	57.5	352	62-60-61	38-37-37
"	"	5J-1086	147.5	162.0	19.0	56.0	363	60-57-58	34-32-33
"	"	"	152.0	165.0	18.0	59.0	363	62-60-60	34-34-33
EV-4355	"	5J-1088	142.5	157.0	18.0	55.0	363	52-54-52	28-29-27
"	"	"	147.5	162.5	17.0	53.5	363	51-50-49	28-27-25
EVD-4441	"	5J-1084	146.5	159.0	18.0	59.0	321	60-60-58	30-31-30
"	"	"	145.0	157.0	18.0	59.0	363	65-65-64	35-35-35
EVD-4215	"	5I-578	135.0	152.0	18.0	59.0	341	72-70-68	46-42-40
"	"	"	143.5	155.0	18.0	59.0	341	66-67-70	40-41-43

CLOSURE HEAD NUTS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-4021	A540, B24	5J-777	141.0	153.0	18.0	60.0	321	63-61-62	44-42-42
"	"	"	144.0	158.0	18.0	56.0	321	53-50-52	29-26-30
EVD-5038	"	6D-728	156.5	167.5	16.0	56.0	321	54-55-56	30-32-33
"	"	"	156.0	167.0	17.0	57.5	321	55-54-55	32-30-32
EVD-4021	"	5J-776	141.0	153.0	18.0	60.0	341	63-61-62	44-42-42
"	"	"	144.0	158.0	18.0	56.0	311	53-50-52	29-26-30
"	"	5J-777	135.0	147.0	19.0	61.5	311	72-73-70	41-43-40
	"								

TABLE 5.3-6 (Continued)

SOUTH TEXAS UNIT 2 REACTOR VESSEL FASTENER MATERIAL PROPERTIES
CLOSURE HEAD STUDS

Heat No.	Grade	Bar No.	0.2% YS ksi	UTS ksi	Elong %	RA %	BHN	Energy at 50°F ft-lb	Lat. Expansion mils
EVD-4021	A540, B24	5J-777	135.0	149.0	19.0	61.5	331	67-68-67	34-38-33
EV-4040	"	5J-778	135.0	148.5	20.0	59.0	321	55-57-55	33-35-34
"	"	"	146.5	158.0	18.0	56.0	352	50-51-52	25-26-26
"	"	5J-779	141.5	155.0	18.0	56.0	341	53-52-53	32-31-32
"	"	"	137.5	150.0	19.0	62.5	321	65-65-62	35-35-34
CLOSURE HEAD WASHERS									
EVD-3954	A540, B24	5I-1099	141.5	153.0	20.0	62.5	331	70-72-70	40-41-40
"	"	"	144.5	154.0	18.0	59.0	321	70-68-65	40-37-36
EV-2232	"	5F-582	135.5	151.0	18.0	52.0	311	56-54-55	31-30-32
"	"	"	143.5	157.0	17.5	53.5	341	60-58-60	32-34-33
EV-2222	"	5E-1226	144.0	155.0	18.5	55.0	352	64-62-64	36-36-37
"	"	"	149.5	163.0	18.0	52.0	341	54-55-55	27-32-34

TABLE 5.3-7

STPEGS UNIT 1 REACTOR VESSEL VALUES FOR
ANALYSIS OF POTENTIAL PRESSURIZED THERMAL SHOCK EVENTS REF. 5.3-15

Unit 1 Material			Cu	Ni	Initial RT_{NDT}	ID Neutron Fluence End-of-Life (E19)	End-of-Life RT_{PTS}
Intermediate	Shell	R1606-1	.04	.63	10	2.22	73
"	"	R1606-2	.04	.61	0	2.22	63
"	"	R1606-3 (Note 1)	.05	.62	10	2.22	82
Lower Shell		R1622-1	.05	.61	-30	3.00	44
		R1622-2	.07	.64	-30	3.00	61
		R1622-3	.05	.66	-30	3.00	44
Intermediate Shell Longitudinal Welds			.02	.07 (Note 2)	-50	1.55	24
Lower Shell Longitudinal Welds			.02	.07 (Note 2)	-50	2.25	29
Intermediate Shell to Lower Shell Girth Weld			.02	.07 (Note 2)	-70	2.22	9

NOTE:
(1) Limiting material for Unit 1 Reactor Vessel
(2) Based on CE NPSD-1039 Rev. 1, "Best Estimate Copper And Nickel In CE Fabricated Reactor Vessel Welds."

TABLE 5.3-8

STPEGS UNIT 2 REACTOR VESSEL VALUES FOR
ANALYSIS OF POTENTIAL PRESSURIZED THERMAL SHOCK EVENTS REF. 5.3-15

Unit 2

Material		Cu	Ni	Initial RT_{NDT}	ID Neutron Fluence End-of-Life (E19)	End-of-Life RT_{PTS}
Intermediate Shell	R2507-1	.04	.65	-10	2.01	52
" "	R2507-2(Note 1)	.05	.64	-10	2.01	63
" "	R2507-3	.05	.61	-40	2.01	31
Lower Shell	R3022-1	.03	.63	-30	2.71	20
	R3022-2	.04	.61	-40	2.71	26
	R3022-3	.05	.60	-40	2.71	26
Intermediate Shell Longitudinal Welds		.04 (Note 2)	.11	-70	1.29	4
Inter Shell to Lower Shell Girth Welds		.04 (Note 2)	.11	-70	2.01	12
Lower Shell Longitudinal Welds		.04 (Note 2)	.11	-70	1.98	12

NOTE:
(1) Limiting material for Unit 2 Reactor Vessel
(2) Based on CE NPSD-1039 Rev. 2, "Best Estimate Copper And Nickel in CE Fabricated Reactor Vessel Welds."
Sample weighted mean.

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Coolant Pumps

5.4.1.1 Design Bases. The reactor coolant pump (RCP) ensures an adequate core cooling flow rate for sufficient heat transfer to maintain a departure from nucleate boiling ratio (DNBR) greater than 1.28 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 rotation inertia is provided by a flywheel, in conjunction with the impeller and motor assembly, to provide adequate flow during coastdown. This forced flow following an assumed loss of pump power and the subsequent natural circulation effect provides the core with adequate cooling.

The RCP 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 described in Reference 5.4-1.

The RCP is show on Figure 5.4-1. The RCP design parameters are given in Table 5.4-1.

Code and material requirements are provided in Section 5.2.

5.4.1.2 Design Description. The RCP is a vertical, single-stage, centrifugal, shaft seal pump designed to pump large volumes of main coolant at high temperature and pressures.
The pump consists of three areas from bottom to top. They are the hydraulics, the shaft seals, and the motor.

1. The hydraulic section consists of an impeller, diffuser, casing, thermal barrier/heat exchanger assembly, lower radial bearing, bolting ring, and pump rotor assembly.

2. The shaft seal section consists of three devices. They are the number 1 controlled-leakage, film-riding face seal and the no. 2 and no. 3 rubbing face seals. These seals are contained within the seal housings. A Station Blackout Seal is installed on selected reactor coolant pumps between the number 1 and number 2 seal. The seal is designed to activate during conditions consistent with station blackout conditions within the RCP seal housing.

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

Attached to the bottom of the pump shaft is the impeller. The reactor coolant is drawn up through the impeller, discharged through passages in the diffuser, and out through the discharge nozzle in the side of the casing. Above the impeller is a thermal barrier heat exchanger which limits heat transfer between reactor coolant system (RCS) water and pump internals. Component cooling water (CCW) is supplied to the thermal barrier heat exchanger.

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

The RCP motor bearings are of conventional design. The radial bearings are the pivoted pad type, and the thrust bearings are tilting pad Kingsbury bearings. All are oil lubricated. The lower radial bearing and the thrust bearings are submerged in oil and the upper radial bearing is fed oil by the action of the thrust runner. CCW is supplied to the two oil coolers on the pump motor.

The motor is a water/air-cooled, Class F thermalasitc epoxy-insulated, squirrel-cage induction motor. The rotor and stator are of standard construction and are cooled by air. Six resistance temperature detectors (RTDs) located throughout the stator (one of which is a spare) are used to sense the winding temperature. The top of the motor consists of a flywheel and an anti-reverse rotation device.

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 HXs, which are supplied with CCW. Each motor has two such coolers, mounted diametrically opposed to each other. In passing through the coolers, the air is cooled to below 122°F so that minimum heat is rejected to the Containment from the motors.

Each of the RCPs is equipped with two frame vibration pickups mounted at the top of the motor support stand. They are mounted in a horizontal plane to pick up radial vibrations of the pump. One is aligned perpendicular; the other is aligned parallel to the pump discharge; signals from all the RCPs are sent to monitoring equipment located in the Main Control Room. This equipment is provided with alarm and indication features via a human-machine interface host computer, which displays the amplitude and/or frequency of pump frame vibration. Pump shaft vibration is monitored continuously in a manner similar to frame vibration monitoring. The pump shaft vibration levels are checked during performance testing prior to shipment.

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A removable shaft segment, the spool piece, 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 flywheel cover.

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

Code Case 1739, Pump Internals Items, was used in the manufacture of the reactor coolant pumps as described in letter NS-CE-1400 dated April 5, 1977 (to Mr. John F. Stolz, NRC Office of Nuclear Reactor Regulation, from Mr. C. Eicheldinger, Westinghouse PWRSED Nuclear Safety). The Nuclear Regulatory Commission's (NRC) approval of the use of this Code Case in the manufacture of the South Texas Project Electric Generating Station (STPEGS) pumps was received via a May 31, 1977 letter to Mr. C. Eicheldinger from Mr. J. F. Stolz.

The RCP assembly is equipped with an oil collection system that is designed to maintain structural integrity during and after Safe Shutdown Earthquake (SSE). The collection system is capable of collecting oil from all potential leakage sites in the RCP motor lube oil subsystem. In particular, the pump motor is installed with enclosures, splash guards, drip pans, drain lines, and a collection tank.

To protect against an oil leak in the Oil Lift System, and enclosure is provided isolating the high pressure oil components from the environment. The upper oil cooler flanges are fitted with splash guards, directing any oil leakage downward to a drip pan. Drip pans are provided under the upper oil pot level detector, upper oil pot fill and drain valve, and upper resistance temperature detector (RTD) conduit box. A catch basin is provided around the motor shaft, extending out from the motor to include the area below the lower oil pot.

Drain lines large enough to accommodate the largest potential oil leak are provided from the drip pan to a collection tank sized to hold the lube oil inventory of at least one RCP motor.

5.4.1.3 Design Evaluation.

5.4.1.3.1 Pump Performance: The RCPs are sized to deliver flow at rates which equal or exceed the required flow rates. Initial RCS Tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation. The estimated performance characteristic is shown on Figure 5.4-2.

The Reactor Trip System (RTS) ensures that pump operation is within the assumptions used for loss of coolant flow analyses, which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a RCP is lost during operation.

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

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

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 long periods of time (with the pump not rotating); even if the no. 1 seal fails entirely during normal operation, the no. 2 seal would maintain these small leakages rates if proper actions were taken by the operator. The plant operator is warned of no. 1 seal damage by increase in the no. 1 seal leakoff. Following warning of excessive seal leakage conditions, the plant operator should 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 actions are taken subsequent to warning of excessive seal leakage conditions.

The effect of loss of offsite power (LOOP) on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the cooling water for seal and bearing cooling. The emergency diesel generators (DGs) are started automatically upon LOOP enabling component cooling flow to be automatically restored. The centrifugal charging pumps may be manually started after the automatic loads are sequenced onto the DGs.

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 in a LOOP condition, each RCP is provided with a flywheel. Thus, the rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow. The coastdown flow transients are provided on the figures in Section 15.3. The pump/motor system is designed for the SSE at the site. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a LOOP coincident with the SSE. Core flow transients are discussed and figures are provided in Sections 15.3.1 and 15.3.2.

5.4.1.3.3 Bearing Integrity: The design requirements for the RCP 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 a very low value, and even under the most severe seismic transients do not begin to approach loads which cannot be adequately carried for short periods of time.

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

Low oil levels in the lube oil sumps signal alarms in the control room and require shutting down of the pump. Each motor bearing contains embedded temperate detectors, and so initiation of failure, separate from loss of oil, is indicated and alarmed in the control room as a high bearing temperature. This again requires pump shutdown. 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, as 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 will lead to the motor being shutdown by the electrical protection systems.

As discussed in Section 9.2.2.5, low CCW flow from the RCP bearing oil coolers is alarmed on a per-pump basis to provide additional information to the control room operator. Testing has been performed by Westinghouse, which has shown the manufacturer's recommended maximum bearing operating temperature will be reached in approximately 10 minutes upon total loss of CCW flow. Since Westinghouse has demonstrated that the RCPs are capable of operation for 10 minutes upon total loss of CCW flow without incurring damage to the motor bearing, and since the operator has redundant control room alarms to warn of degraded CCW flow conditions, 10 minutes in a conservative operator response time for this event during normal operation. RCP seal cooling is maintained by seal injection from the charging system. In the event that a LOOP has occurred, seal injection from the centrifugal charging pumps (CCPs) will also be lost until the CCPs can be manually re-started.

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 overspeed, and the flywheel maintains its integrity, as it is still supported on a shaft with two bearings. Flow transients are provided on the figures in Section 15.3.3 for the assumed locked rotor.

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 number 1, 2 or 3 seals results in a shearing of the anti-rotation pin in the seal ring. An inadvertent actuation of the Station Blackout Seal will result in a shearing of the Station Blackout Seal material. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions are initially by high temperature signals from the bearing water temperature detector and excessive number 1 seal leakoff indications, respectively. Following these signals, pump vibration levels are checked. Excessive vibration indicates mechanical trouble and the pump is shutdown for investigation.

5.4.1.3.5 Critical Speed: The RCP 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 RCP components assure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing. Further discussion and analysis of missile generation is contained in Reference 5.4-1.

5.4.1.3.7 Pump Cavitation: The minimum NPSH required by the RCP at running speed is 192 ft (approximately 85 psi). In order for the controlled leakage seal to operate correctly, it is necessary to have a minimum differential pressure of 200 psi across the number 1 seal. This corresponds to a primary loop pressure at which the NPSH requirement is exceeded and no limitation on pump operation occurs from this source.

5.4.1.3.8 Pump Overspeed Considerations: For turbine trips actuated by either the RTS or the turbine protection system, the generator circuit breaker opens thereby maintaining offsite power to the RCP motors.

In the case of a generator trip, the generator circuit breaker opens and the RCP motors are powered by offsite power. Therefore, pump overspeed conditions are prevented for these transients. Further discussion of pump overspeed considerations is contained in Reference 5.4-1.

5.4.1.3.9 Anti-Reverse Rotation Device: Each of the RCPs is provided with an anti-reverse rotation device in the motor. This anti-reverse mechanism consists of pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, two spring return mechanisms for the ratchet plate, and three shock absorbers.

After the motor has slowed and come to a stop, the dropped pawls engage the ratchet plate and, as 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, the pawls are bounced into an elevated position and are held in that position by friction resulting from centrifugal forces acting upon the pawls. Considerable plant experience with the design of these pawls has shown high reliability of operation.

5.4.1.3.10 Shaft Seal Leakage: Leakage along the RCP shaft is controlled by three shaft seals arranged in series. Injection flow is directed to each RCP via the seal water injection filter. It enters the pump through a connection on the thermal barrier flange and is directed down to a point between the pump shaft bearing and the pump seals. Here the flow splits. A portion flows down the shaft through and around the lower radial bearing, down past the thermal barrier/HX and into the RCS; the remainder flows up the pump shaft annulus to the no. 1 seal. Above the seal, most of the flow leaves the pump via the no. 1 seal discharge line. Minor flow passes through the no. 2 seal and discharge line. A back flush injection from a standpipe flows into the no. 3 "double dam" seal between its seal faces. At this point, the flow divides with half flushing through one side of the seal and out the no. 2 seal leakoff while the remaining half flushes through the other side and out the no. 3 seal leakoff. This arrangement assures essentially zero leakage of reactor coolant or trapped gases from the pump.

5.4.1.3.11 Seal Discharge Piping: Leakage flow from the no. 1 seal is nominally at the volume control tank (VCT) pressure. Water from each pump no. 1 seal is piped to a common manifold, through the seal water return filter, and through the seal water HX where the temperature is reduced to that of the VCT. The no. 2 seal leakoff line is discharged to the reactor coolant drain tank (RCDT); the no. 3 line is discharged to the Containment sump.

5.4.1.4 Tests and Inspections. The RCPs can be inspected in accordance with ASME Section XI.

The pump casing is cast in two pieces and welded; the full penetration weld in the pressure boundary is prepared with a smooth surface transition between weld metal and parent metal for radiographic inspection. The design enables disassembly and removal of the pump internals for access to the internal surface of the pump casing. Support feet are cast integral with the casing to eliminate a weld region.

The RCP quality assurance program is given in Table 5.4-2.

5.4.1.5 Pump Flywheels. The integrity of the RCP flywheel is assured on the basis of 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 2,000 psi at zero speed, but this stress becomes zero at approximately 600 rpm because of radial expansion of the hub. The RCPs run at approximately 1,190 rpm and may operate briefly at overspeeds up to 109 percent (1,295 rpm) during loss of outside load. For conservatism, however, 125 percent of operating speed was selected as the design speed for the RCPs. The flywheels are given a test of 125 percent of the maximum 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 such as vacuum degassing, vacuum melting, or electroslag remelting that minimizes flaws in the material and improves its fracture toughness properties. Each plate is fabricated from A-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 Regulatory Guide (RG) 1.14.

Flywheel blanks are flame-cut from the A-533, Grade B, Class 1 plates with at least 1/2 in. of stock left on the outer and bore radii for machining to final dimensions. The finished machined bores, keyways, and drilled holes within 4 in. of the bore 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, subsection NB-2530 as supplemented by RG 1.14.

The RCP motors are designed such that, by removing the cover to provide access, the flywheel is available to allow inservice inspection in accordance with requirements of Section XI of the ASME Code and the recommendations of RG 1.14.

5.4.1.5.3 Material Acceptance Criteria: The RCP motor flywheel conforms to the following material acceptance criteria:

1. The Nil-Ductility Transition Temperature (NDTT) of the flywheel material is obtained by two drop weight tests (DWT) which exhibit "no-break" performance at 20°F in accordance with ASTM E208. These DWTs demonstrate that the NDTT of the flywheel material is no higher than 10°F.

2. A minimum of three Charpy V-notch impact specimens from each plate are tested at ambient (70°F) temperature in accordance with ASTM E23. The Charpy V-notch (C_v) energy in both the parallel and normal orientation with respect to the rolling direction of the flywheel material is at least 50 foot pounds at 70°F to demonstrate compliance with RG 1.14.
   A lower bound K_{Id} reference curve (Figure 5.4-3) has been constructed from dynamic fracture toughness data generated in A-533, Grade B, Class 1 steel (Ref. 5.4-2). All data points are plotted on the temperature scale relative to the NDTT. The construction of the lower bound curve below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static, together represent a large degree of conservatism. Reference of this curve to the guaranteed NDTT of +10°F indicates that, at the predicted flywheel operating temperature of 110°F, the minimum fracture toughness is in excess of 100 ksi-in$^{1/2}$. This conforms to the RG 1.14 requirement that the dynamic stress intensity factor must be at least 100 ksi-in$^{1/2}$.

Thus, it is concluded that flywheel plate materials are suitable for use and can meet RG 1.14 acceptance criteria on the basis of suppliers' certification data.

5.4.1.5.4 Reactor Coolant Pump Flywheel Integrity: The Westinghouse design is in agreement with RG 1.14, Rev. 1, except for the following:

1. Post-Spin Inspection

Westinghouse has shown in WCAP-8163, "Topical Report Reactor Coolant Pump Integrity in LOCA", that the flywheel would not fail at 290 percent of normal speed for a flywheel flaw of 1.15 in. or less in length. Results for a double-ended guillotine break at the pump discharge, with full separation of pipe ends assumed, show the maximum overspeed to be less than 110 percent of normal speed. The maximum overspeed was calculated in WCAP-8163 to be about 280 percent of normal speed for the same postulated break, and an assumed instantaneous loss of power to the RCP. In comparison with the overspeed presented above, the flywheel is tested at 125 percent of normal speed. Thus, the flywheel could withstand a speed up to 2.3 times grater than the flywheel spin test speed of 125 percent provided no flaws greater than 1.15 in. are present. If the maximum speed were 125 percent normal speed or less, the critical flaw size for failure would exceed 6 in. in length. Nondestructive tests and critical dimension examinations are all performed before the spin tests. The inspection methods employed (described in WCAP-8163) provide assurance that flaws significantly smaller than the critical flaw size of 1.15 in. for 290 percent of normal speed would be detected. Flaws in the flywheel are recorded in the pre-spin inspection program (see WCAP-8163). Flaw growth attributable to the spin test (i.e., from a single reversal of stress, up to and back), under the most adverse conditions, is about three orders of magnitude smaller than what nondestructive inspection techniques are capable of detecting. For these reasons, Westinghouse performs no post-spin inspection and believes that pre-spin test inspections are adequate.

2. Interference Fit Stresses and Excessive Deformation

Much of RG 1.14, Rev. 1, deals with stresses in the flywheel resulting from the interference fit between the flywheel and the shaft. Because Westinghouse's design specifies a light interference fit between the flywheel and the shaft, at zero speed, the hoop stresses and radial stresses at the flywheel bore are negligible. Centering of the flywheel relative to the shaft is accomplished by means of keys and/or centering devices attached to the shaft, and at normal speed, the flywheel is not in contact with the shaft in the sense intended by Revision 1. Hence, the definition of "Excessive Deformation", as defined in Revision 1 of RG 1.14, is not applicable to the Westinghouse design since the enlargement of the bore and subsequent partial separation of the flywheel from the shaft does not cause imbalance of the flywheel. Extensive Westinghouse experience with RCP flywheels installed in this fashion has verified the adequacy of the design.

Westinghouse's position is that combined primary stress levels, as defined in Revision 0 of Safety Guide 14 C.2 (a) and (c), are both conservative and proven and that no changes of these stress levels are necessary. Westinghouse designs to these stress limits and thus does not have permanent distortion of the flywheel bore at normal or spin test conditions.

3. Section B, Discussion of Cross Rolling Ratio of 1 to 3

Westinghouse's position is that specification of a cross rolling ratio is unnecessary since past evaluations have shown that A-533 Grade B Class 1 materials produced without this requirement have suitable toughness for typical flywheel applications. Proper material selection and specification of minimum material properties in the transverse direction adequately ensure flywheel integrity. An attempt to gain isotrophy in the flywheel material by means of cross rolling in unnecessary since adequate margins of safety are provided by both flywheel material selection (A-533 Grade B Class 1) and by specifying minimum yield and tensile levels and toughness test values taken in the direction perpendicular to the maximum working direction of the material.

4. Section C, Item 1.a Relative to Vacuum-melting and Degassing Process or the Electroslag Process

Vacuum Treatment – The requirements for vacuum melting and degassing process or the electroslag process are not essential in meeting the balance of the Regulatory Position nor do they, in themselves, ensure compliance with the overall Regulatory Position. The initial Safety Guide 14 stated that the "flywheel material should be produced by a process that minimizes flaws in the material and improves its fracture toughness properties." This is accomplished by using A-533 material including acuum treatment.

5. Section C, Item 2.b – Westinghouse suggests that this paragraph be reworded as follows in order to remove the ambiguity of reference to an undefined overspeed transient.

<u>Design Speed Definition</u>

Design speed should be 125 percent of normal speed. Normal speed is defined as the synchronous speed of the AC drive motor at 60 Hz.

5.4.2. Steam Generator

5.4.2.1 Steam Generator Materials.

5.4.2.1.1 Selection and Fabrication of Materials: All pressure boundary materials used in the steam generator (SG) 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 Section 5.2.3, with types of materials listed in Table 5.2-2. Fabrication of Reactor Coolant Pressure Boundary (RCPB) materials is also discussed in Section 5.2.3, particularly in Sections 5.2.3.3 and 5.2.3.4.

Testing has justified the selection of corrosion-resistant Alloy 690, a nickel-chromium-iron alloy, for the SG tubes (ASME SB-163) and divider plate (ASME SB-168). The interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. The primary side of the tube sheet is weld clad with Alloy 690. The tubes are expanded to the tube sheet upper surface after the ends are seal welded to the tube sheet cladding. The flush fusion welds are performed in compliance with Sections III and IX of the ASME Code and are thoroughly inspected before each tube is hydraulically expanded.

The design features of the steam generators include the use of thermally treated Alloy 690 tube material, type 405 stainless steel tube support material, broached flat contact tube supports, hydraulically expanded tubesheet joint, and minimum gap U-bend construction. The use of thermally treated Alloy 690 provides additional corrosion resistance for the tubes. The corrosion resistance has been proven through laboratory testing and through actual plant operation. The use of the full-depth hydraulic expansion of the tubes in the tubesheet closes the crevices with minimal residual stresses. Industrial experience indicates that hydraulic expansion results in one of the lowest residual stresses of any tubesheet joint process. The broached tube support plate reduces the tube-to tube support plate crevice area and permits full steam/water flow in the open areas to the tube while the flat contact geometry provides additional dryout margin over a concave contact geometry.

In the steam generators, a 0.005-inch nominal U-bend gap, in conjuction with tightly controlled U-bend tubing ovality and retainer ring design, has proven through analytical assessment to decrease tube wall degradation due to flow-induced vibration, increasing margin by more that a order of magnitude. Manufacturing of anti-vibration bars (AVBs) using type 405 stainless steel, adoption of rectangular AVB configuration, using four sets of AVBs, and tight control of AVB insertion depth, provides enhanced AVB performance and additional reliability. Staggering the AVBs minimizes the pressure drop in the U-bend region to increase circulation ratio and reduce potential steam blanketing. The increase in the number of sets of AVBs reduces the number of tubes, which are potentially affected by the flow-induced vibration mechanism to which tube degradation has been attributed in earlier steam generator designs.

All major pressure boundary components are forged with the following exception: the upper shell barrels in some steam generators are fabricated from rolled and welded plate that has been machined on the inside diameter and outside diameter. This construction minimizes the in-service inspection requirements on the pressure boundary welds.

Code cases used in material selection are discussed in Section 3.12 as part of the discussion of conformance with RGs 1.84 and 1.85.

During manufacture, cleaning is performed on the primary and secondary sides of the SG in accordance with written procedures which follow the guidance of RG 1.37 and American Nuclear Standards Institute (ANSI) N45.2.1-1973. Westinghouse's recommendations for cleaning are given in their process specifications, as discussed in Section 5.2.3.4.

The fracture toughness of the materials is discussed in Section 5.2.3.3. Adequate fracture toughness of ferritic materials in the RCPB is provided by compliance with Appendix G of l0CFR50 and with Article NB-2300 of Section III of the ASME Code. Per the discussion in Section 5.2.3.3, consideration of fracture toughness is only necessary for materials in Class 1 components.

5.4.2.1.2 Steam Generator Design Effects on Materials: Several features are employed to control the regions where deposits would tend to accumulate and cause corrosion. To avoid extensive crevice areas at the tube sheet, the tubes are expanded to the secondary surface of the tube sheet after their ends are seal welded to the Inconel cladding on the primary side of the tube sheet. To date, aside from damage by foreign object/loose parts, the only form of tube degradation which has been identified in the Westinghouse Model F and later feedring design steam generators using Alloy 690 thermally treated tubing, is fretting wear between the tube and steam generator AVBs. Only a small number of tubes (expected to be less than 1% total over the life of the steam generators) would be subject to this phenomenon. The AVB design for the steam generators maintains smaller, more tightly controlled tube-to-AVB gaps, increased tube-to-AVB contact area, and utilizes more AVBs. These factors should greatly lower the potential for tube fretting at AVB intersections.

5.4.2.1.3 Compatibility of Steam Generator Tubing with Primary and Secondary Coolants: Westinghouse steam generators with Inconel 600 tubing have experienced tube degradation in several forms. These are wastage, intergranular attack, stress corrosion cracking, denting, and mechanical vibrations. Each of these forms of degradation is discussed below and the specific measures to prevent or mitigate their occurrence at South Texas are included.

Wastage is characterized by general loss of metal from the tube wall because of a chemical corrosive reaction. Wastage has occurred in steam generators which used sodium phosphate as a chemical additive. The South Texas steam generators use all volatile treatment.

Intergranular attack is a corrosion phenomenon in which the grain boundaries of the Inconel 600 tube are preferentially attacked without a preferential stress-related orientation. Intergranular attack has occurred in steam generators in which the tubes were not expanded the full depth of the tube sheet.

The resulting crevice can provide a site for concentrating impurities in the bulk environment. The South Texas generator tubes have been expanded the full depth of the tube sheet by hydraulic expansion. Thus, the elimination of the crevice should preclude the occurrence of intergranular attack. In addition, in some plants, intergranular attack has occurred slightly above the tube sheet where sludge tends to accumulate. The steam generators contain thermally treated Alloy 690 tubes, which provide improved resistance to intergranular attack. The South Texas steam generators contain a flow distribution plate located just above the tubesheet, which encourages recirculating flow to sweep the tubesheet before turning upward through the tube bundle.

Stress corrosion cracking (SCC) refers to the type of corrosive intergranular attack in which cracks grow in the direction perpendicular to an applied or residual tensile stress. SCC has occurred in smaller radius inner row U-bend tubes of some Westinghouse steam generators. Early instances were related to applied stresses caused by denting, and the actions taken to mitigate denting are discussed next. Other instances of U-bend cracking were related to residual stresses. However, the design of the steam generators is such that the minimum inner tube U-bend radius is greater than that in other designs which have experienced SCC. This should substantially improve one potential cause of U-bend cracking. In addition, the small radius U-bends (row 1 through 17) are stress relieved after tube bending operation. Primary water stress corrosion cracking has occurred inside the tube expanded into the tubesheet in steam generators using Alloy 600 mill annealed tubing. (However, such cracking is not expected to occur in SGs with thermally treated Alloy 690 tubes).

As previously discussed, the Alloy 690 tubing installed in the steam generators has been shown by test (and operating experience) to be exceptionally resistant to both primary water stress corrosion cracking (PWSCC) and outer diameter stress corrosion cracking (ODSCC). The Alloy 690 tubing specification is in accordance with the requirements of ASME Code Section III with code case N-20-3.

The steam generator tube expansion process employed in the tubesheet region results in reduced residual stress levels. This reduced residual stress will lower the potential for PWSCC in this region. Mitigating measures also include T-hot reduction and secondary water chemistry control.

The potential for ODSCC to develop in the steam generators is also reduced. Axial flow-paths around the steam generator tubes provided by the trifoil support design significantly reduces crevice area and contaminant hideout potential. The stainless steel tube support plate material does not represent a potential for magnetite generation or general corrosion product buildup.

Denting is caused by rapid corrosion of the tube support plates at the holes where the tubes pass through the support plates. Denting is known to be caused by operation of the steam generators outside the allowable range of water chemistry control parameters, specifically during times of major condenser leakage. It appears that the use of copper materials in the feed and condensate systems contributes to the severity of denting. The following actions have been taken to prevent denting:

(a) The plant operates in accordance with the chemistry guidelines described in the Steam Generator Management Program.

(b) The use of copper-containing alloys in the secondary loop has been reviewed. Any wetted parts for which the flow does not go first to the condensate polishers before entering the steam generators have been reviewed for copper content. Items that were identified as containing copper have been replaced with suitable non-copper material. Materials for these flow paths are restricted by design to very low amounts of copper which would yield no more than trace amounts in the secondary flow. For the remainder of the secondary loop which does go first to the condensate polishers, the copper-containing items have been minimized. An exception is the use of aluminum bronze for the condenser tubesheets. Condenser integrity is improved through the use of titanium tubes and full flow condensate polishers are utilized. A full flow dearator has been provided to achieve steady state oxygen concentrations of 2 ppb at full and low load operation.

(c) Additionally, the steam generators employ stainless steel tube support plates, which should preclude the formation of corrosion products within the crevice. The tube support plates employ a limited contact trifoil hole shape, which permits axial flow around the tube, which also minimizes the buildup of corrosion products in this area.

A comprehensive program of SG inspections will ensure detection and correction of any degradation that might occur in the SG tubing, and defective tubes may be plugged or repaired with welded sleeves using an approved welding procedure.

5.4.2.1.4 Cleanup of Secondary Side Materials: Several methods are employed to clean operating SGs of corrosion-causing secondary side deposits. Sludge lancing, a procedure in which a hydraulic jet inserted through an access opening (inspection port) loosens deposits which are removed by means of a suction pump, can be performed when the need is indicated by the results of SG tube inspection. The steam generators have a passive sludge-collecting feature in the steam generator recirculating loop. The water which passes through the tube bundle without turning into steam is separated by the primary and secondary moisture separators and is directed back to the tube bundle. As the water leaves the separators, a portion of it passes through the sludge collector, which provides a laminar flow settling zone for suspended solids. A significant portion of the sludge is expected to settle in the sludge collector, away from the tube bundle. During outages, the sludge collectors can be opened and cleaned. Blowdown procedures are performed as deemed necessary by regular water chemistry testing.

5.4.2.2 Steam Generator Inservice Inspection. The SG is designed to permit inservice inspection of Class 1 and 2 components, including individual tubes. A number of access openings make it possible to inspect and repair or replace a component according to the techniques specified. Two 16-inch ID manways are provided on the secondary side and two 18-inch manways are provided on the primary side. The steam generators have six 6-inch handholes located to maximize their utility for inspection and maintenance of the lower tube bundle (e.g. sludge cleaning). The steam generators also have two 4-inch inspection ports located on the lower shell-transition cone junction at an elevation just above the top tube support plate and two 2-inch inspection ports above each of the other tube support plates. Access provisions (ladders, grab bars, and manways) for inspection of the steam generator upper internals is also provided. Enhanced maintenance and access provisions in conjunction with channel head divider plate electropolishing and low cobalt Alloy material selection will reduce radiation exposure.

The design aspects that provide access for inspection and the proposed inspection program comply with the edition of Section XI, Division 1 of the ASME Code, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," required by 10CFR50.55a, paragraph g and Technical Specifications for Eddy Current Examination.

The inspection program is included in the Steam Generator Program as implemented by Technical Specification Amendments 209 for Unit 1 and 196 for Unit 2. Prior to power operation, a baseline tube inspection was performed which met the technical requirements of RG 1.83 (Table 3.12-1), NEI 97-06, and the Standard Technical Specifications for Westinghouse PWRs. Inservice inspection of Class 1 components includes that of individual SG tubes as specified in the Technical Specifications. Equipment and procedures make it possible to detect and locate tubes with a wall defect penetrating 20 percent or more.

5.4.2.3 Design Bases. Steam generator design data are given in Table 5.4-3. Code classifications of the SG components are given in Section 3.2. Although the ASME classification for the secondary side is specified to be Class 2, the current philosophy is to design all pressure retaining parts of the SG, and thus both the primary and secondary 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 SG are discussed in Section 3.9.1. Estimates of radioactivity levels anticipated in the secondary side of the SG during normal operation and the bases for the estimates are given in Chapter 11. The accident analysis of a SG tube rupture is discussed in Chapter 15.

The SG internal moisture separation equipment is designed to ensure that moisture carryover does not exceed 0.10 percent by weight under the following conditions:

1. Steady-state operation up to 110% of full load steam flow with water at the normal operating level.

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

3. A step load change of 10 percent of full power in the range from 15 percent to 100 percent full load steam flow.

The water chemistry on the reactor side is selected to provide the necessary boron content for reactivity control and to minimize corrosion of RCS surfaces. The water chemistry of the steam side and its effectiveness in corrosion control are discussed in Chapter 10. Compatibility of SG tubing with both primary and secondary coolants is discussed further in Section 5.4.2.1.3.

The SG is designed to prevent unacceptable damage from mechanical or flow-induced vibration. Tube support adequacy is discussed in Section 5.4.2.5.3. Further consideration is given in Section 5.4.2.5.4 to the effect of tube wall thinning on accident condition stresses.

Feedwater is introduced through a nozzle located in the upper shell of the steam generator. This nozzle does not require a flow-limiting device because the feedring itself provides this function. The nozzle contains a welded thermal liner, which minimizes the impact of rapid feedwater temperature transients on the nozzles and prevents feedring drainage during low-level operation. The feedring is located above the elevation of the nozzle to minimize the time required to fill the nozzle during a cold water addition transient. This is an important feature for reducing the thermal fatigue loading in the main feedwater nozzle and helps to eleminate the potential for the occurence of water hammer.

A reactor coolant system T-hot reduction program has been undertaken for the South Texas Project. This program reduces the reactor coolant system operating temperature, extending the life of the steam generators. Evaluations have been performed to assess the impact of this program on various aspects of the steam generators. These evaluations indicate that steam generator critical components will still satisfy the requirements of ASME Section III, Subsection NB for reduced T-hot temperature. Also, the steam generators will retain the same operating characteristics.

5.4.2.4 <u>Design Descriptions</u>. The SG shown on Figure 5.4-4 is a vertical shell and U-tube evaporator with integral moisture separating equipment.

On the primary side, the reactor coolant flows through the inverted U-tubes, entering and leaving through nozzles located in the hemispherical bottom head of the SG. The head is divided into inlet and outlet chambers by a vertical divider plate extending from the head to the tube sheet.

The steam generator is a vertical U-tube feedring-type steam generator. Manways of 18-inch diameter are provided for access to both sides of the divided head. The feedwater inlet is located at approximately 2/3 of the steam generator height and is distributed equally around the circumference of the steam generator shell by the feedring. Feedwater enters the tube bundle by flowing downward between the steam generator external shell and inner wrapper barrel. An open area at the bottom of the wrapper barrel permits the feedwater to enter the tube bundle. Steam is generated and flows upward through the moisture separators and through the flow restrictor outlet nozzle at the top of the steam drum. The high efficiency centrifugal steam separators remove most of the entrained water. Chevron dryers are employed to increase the steam quality to a minimum of 99.90% (0.10% moisture).

The steam generator channel head and tubesheet are protected from the primary water by applying an autogenous weld-desposited Alloy 690 cladding to the primary surfaces of the channel head and tubesheet. The cladding surface is machined to a smooth condition. The channel head and the divider plate is electropolished, thereby reducing the collection of radioactive contamination inside the steam generators during refueling and maintenance periods.

All materials for steam generator construction are selected and fabricated in accordance with the requirements of the ASME Code, Section III. The steam generator tubing material is Alloy 690. Alloy 690 material represents the state of the art technology for heat transfer tubing. The tubing receives a heat treatment process after forming, which results in a grain boundary structure which has been shown by test to be exceptionally resistant to primary water stress corrosion cracking. Under the water conditions expected, Alloy 690 tubing material is also exceptionally resistant to outer diameter stress corrosion cracking, which has affected the secondary side of the tubes in operating plants.

Ferritic material in the primary side of the steam generators includes the channel head casting, tubesheet, primary nozzles, manway covers, and manway studs and nuts. Fracture toughness data for the channel head, tubesheet, and associated weldments is consistent with ASME Code. As-built data generally shows the test data to far exceed the ASME Code minimum values. The materials of construction and general design of the steam generator shell have resulted in a reduction in the number of pressure boundary welds required for assembly of the shell.

5.4.2.5 Design Evaluation.

5.4.2.5.1. Forced Convection: The limiting case for heat transfer capability is the "Nominal 100 Percent Design" case. The SG effective heat transfer coefficient is determined by the physical characteristics of the steam generator and the fluid conditions in the primary and secondary systems for the nominal 100% design case. It includes a conservative allowance for fouling and uncertainty. Sufficient heat transfer area is provided to permit the achievability of the design heat transfer rate with 10% of the tubes plugged.

5.4.2.5.2 Natural Circulation Flow: The driving head created by the change in coolant density as it is heated in the core and rises to the reactor vessel outlet nozzle initiates convection circulation. This circulation is enhanced by the fact that the SGs, which provide a heat sink, are at a higher elevation than the reactor core, which is the heat source. Thus, natural circulation is assured for the removal of decay heat during hot shutdown in the unlikely event of loss of forced circulation.

5.4.2.5.3 Mechanical Flow-Induced Vibration Under Normal Operation: In the design of Westinghouse SGs, the potential for tube wall degradation attributable to mechanical or flow-induced excitation has been thoroughly evaluated. The evaluation included detailed analyses of the tube support systems for various mechanisms of tube vibration.

The primary cause of tube vibration in heat exchangers is hydrodynamic excitation due to secondary fluid flow on the outside of the tubes. In the range of normal SG operating conditions, the effects of primary fluid flow inside the tubes and mechanically-induced tube vibration are considered to be negligible.

The potential for tube wall degradation attributable to mechanical or flow-excitation is exceptionally low in the feedring type steam generator design. Flow-induced vibration at the tube support plate intersections has not been observed in similar steam generator design. The SGs have enhancements, such as the manufacture of anti-vibration bars (AVBs) using type 405 stainless steel, adoption of a rectangular AVB configuration, using four sets of AVBs, and tight control of AVB insertion depth, which provide enhanced AVB performance an additional reliability. Increasing the number of sets of AVBs reduces the number of tubes, which are potentially affected by the vibration mechanism to which tube degradation has been attributed in some conventional steam generators. Adoption of the other features reduces wear potential of the small number of tubes which could be affected by more than a factor of twenty based on classical wear theory.

In summary, tube vibration has been thoroughly evaluated. Mechanical and primary flow excitation are considered negligible. Secondary flow excitation has been evaluated and it is concluded that if tube vibration does occur, the magnitude will be limited. Tube fatigue due to the vibration is judged to be negligible. Any tube wear resulting from tube vibration would be limited and would progress slowly. This allows use of a periodic tube inservice inspection program for detection and monitoring of any tube wear. This inservice inspection program, in conjunction with tube plugging criteria, provides for safe operation of the SGs.

5.4.2.5.4 Allowable Tube Wall Thinning Under Accident Conditions: Details of the analyses carried out in accordance with regulatory guidelines are presented in WCAP-15095 (Ref. 5.4-6), which defines the RG 1.121 analysis and provides the bases for the plugging limit. Tube wall thinning is chemical corrosion of the tubing that results in tube wall loss. The tube material is Alloy 690. The consequences will not be a safety concern if tube thinning should occur in the steam generators. This position is based on the characteristics of tube thinning, which is detectable by eddy current testing and would be detected during normal inservice inspections. When thinning occurs, the rate of degradation is relatively slow, less than 2 percent per year. This allows thinning progess to be monitored over several inspection intervals and tubes to be preventatively plugged prior to exceeding the tube structural limit. In addition, thinning depth tends to be overestimated by eddy current. This is a factor of conservatism that will reduce the possibility of a tube experiencing through-wall degradation.

The SG tubes, existing originally at their minimum wall thickness and reduced by a very conservative general corrosion loss, still provide quite an adequate safety margin. Thus, it can be concluded that the ability of the SG tubes to withstand accident loadings is not affected by a lifetime of general corrosion losses.

5.4.2.6 Quality Assurance. The SG quality assurance program is given in Table 5.4-4. Radiographic inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code.

Magnetic particle inspection and acceptance standards are in accordance with requirements of Section III of the ASME Code.

Hydrostatic tests are performed in accordance with Section III of the ASME Code. In addition, the heat transfer tubes are subjected to a hydrostatic test pressure prior to installation into the vessel which is not less than 1.25 times the primary side design pressure.

5.4.3 Reactor Coolant Piping

5.4.3.1 Design Bases. The 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 of the ASME 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 ASME Section III, Class 1 requirements.

Stainless steel piping conforms to ANSI B36.19 for sizes 1/2 in. through 12 in. and wall thicknesses Schedule 40S through 80S. Stainless steel pipe outside of the scope of ANSI B36.19 conforms to ANSI B36.10.

The minimum wall thicknesses of the loop pipe and fittings are not less than that calculated using the ASME III Class 1 formula of Paragraph NB-3641.1 (3) with an allowable stress value of 17,550 psi.

The pipe wall thickness for the pressurizer surge line is Schedule 160. The minimum pipe bend radius is 5 nominal pipe diameters; ovality does not exceed 6 percent.

All butt welds, and branch connection nozzle welds are of a full penetration design, 2 in. and smaller are socket welded joints.

Processing and minimization of sensitization are discussed in Section 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 Section 5.2.4.

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

Reactor coolant loop (RCL) piping is centrifugally cast pipe. RCL fittings are cast seamless without longitudinal welds and electroslag welds. Pipe and fittings comply with the requirements of the ASME Code, Section II, Parts A and C, Section III, and Section IX.

The RCS piping is specified in the smallest sizes consistent with system requirements. This design philosophy results in the reactor inlet and outlet piping diameters given in Table 5.4-5. The line between the SG and the pump suction is larger to reduce pressure drop and improve flow conditions to the pump suction.

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. All smaller piping which comprises part of the RCS, such as the pressurizer surge line, spray and relief lines, loop drains and connecting lines to other systems, are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank (PRT) is stainless steel. All joints and connections are welded, except for the pressurizer code safety valves, where flanged joints are used. Thermal sleeves are installed on the pressurizer surge and spray nozzles where high thermal stresses could develop due to rapid changes in fluid temperature during normal operational transients.

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

1. Residual heat removal (RHR) pump suction lines, which are 45 degrees 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 Residual Heat Removal System (RHRS), should this be required for maintenance.

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

3. The differential pressure taps for flow measurement, which are downstream of the SGs on the first 90-degree elbow.

4. The pressurizer surge line, which is attached at the horizontal centerline.

5. The safety injection connections to the hot leg, for which inservice inspection requirements and space limitations dictate location on the horizontal centerline.

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

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

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

3. The RCS hot leg, narrow range, fast-response temperature detectors are located in thermowells which extend into the coolant system through modified scoop connections.

4. The wide range hot and cold leg temperature detectors and the cold leg narrow range fast response temperature detectors are located in RTD wells that extend into the reactor coolant pipes.

Narrow range RTDs in the hot and cold leg of each RCL provide temperature signals for the reactor control and protection system.

The average hot leg temperature (T_{hot}) is obtained by averaging the temperature from 3 fast-response, narrow-range RTDs in thermowells. The thermowells are positioned inside the three scoops which were to be used on the RTD Bypass System. The end of the scoop has been removed to allow the thermowell to protrude into the coolant system. Th three RTDs are located in the same plane, 120 degrees apart around the RCS hot leg piping.

The cold leg temperature is measured using one fast-response, narrow-range, RTD in a thermowell at the discharge of the RCP. The thermowell is inserted through the cold leg pipe nozzle.

The average hot leg temperature and the cold leg temperature are used to determine the reactor coolant ΔT between the hot leg and cold leg and the $T_{avg.}$.

The averaging of the three hot leg temperature signals to obtain T_{hot} is accomplished through the Qualified Display Processing System (QDPS) which is described in Section 7.5.6. CN-3240

The RCS piping includes those sections of piping interconnecting the reactor vessel, SG, and RCP. It also includes the following:

1. Charging line and alternate charging line from the system isolation valve up to the branch connections on the RCL.

2. Letdown line and excess letdown line from the branch connections on the RCL to the system isolation valve.

3. Pressurizer spray lines from the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.

4. RHR lines to or from the RCLs up to the designated check valve or isolation valve.

5. Safety injection lines from the designated check valve to the RCLs.

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

7. Loop fill, loop drain, sample*, and instrument* lines to or from the designated isolation valve to or from the RCLs.

8. Pressurizer surge line from one reactor coolant loop hot leg to the pressurizer vessel inlet nozzle.

9. Pressurizer spray scoop, sample connection* with scoop, reactor coolant temperature element installation boss, and the temperature element well itself.

10. All branch connection nozzles attached to RCLs.

11. Pressure relief lines from nozzles on top of the pressurizer vessel up to and through the pressurizer power-operated relief valves (PORVs) and pressurizer safety valves.

12. Seal injection water and labyrinth differential pressure lines to or from the RCP inside Containment.

13. Auxiliary spray line from the isolation valve to the pressurizer spray line header.

14. Sample lines* from the pressurizer to the isolation valve.

15. Reactor vessel head vent piping from the reactor vessel head to the second isolation valve.

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* Lines with a 3/8-in. 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.

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

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.

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.

The design and construction are in compliance with ASME Section XI. Pursuant to this, all pressure-containing welds out to the second valve that delineates the RCS boundary are available for examination with removable insulation.

Components constructed with stainless steel will operate satisfactorily under normal plant chemistry conditions in pressurized water reactor systems because chlorides, fluorides, and particularly oxygen, are controlled to very low levels (Section 5.2.3).

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to the desired reactor coolant water quality listed in Table 5.2-4. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical Volume and Control System (CVCS) and sampling system which are described in Chapter 9.

5.4.3.3.2 Sensitized Stainless Steel: Sensitized stainless steel is discussed in Section 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 $C1/dm^2$ and 0.0015 $F/dm^{2.}$

5.4.3.4 Tests and Inspections. The RCS piping quality assurance program is given 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 the surge line and 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 both the entire outside and inside 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 Restrictor

5.4.4.1 Design Basis. The outlets nozzle of the SG 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 back-pressure which limits further increase in flow. Several protective advantages are thereby provided: rapid rise in Containment pressure is prevented, the rate of heat removal from the reactor coolant is kept within acceptable limits, thrust forces on the main steam line piping are reduced, and most importantly, stresses on internal SG components, particularly the tube sheet and tubes, are limited. The restrictor is also designed 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 inserted into the holes in an integral steam outlet low alloy steel 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 low alloy steel forging holes, the Inconel venturi nozzles are welded to the Inconel cladding on the inner surface of the forging (Figure 5.4-5).

5.4.4.3 Design Evaluation. The flow restrictor design has been sufficiently analyzed to assure its structural adequacy. The equivalent throat diameter of the SG outlet is 16 inches, and the resultant pressure drop through the restrictor at 100 percent steam flow is approximately 3.4 psig. This is based on a design flow rate of 4.24×10^6 lb/hr. Materials of construction and manufacturing of the flow restrictor are in accordance with Section III of the ASME Code.

5.4.4.4 Tests and Inspections. No tests and inspections beyond those during fabrication are anticipated.

5.4.5 Main Steam Line Isolation System

Refer to Section 10.3.2.

5.4.6 Reactor Core Isolation Cooling System

Not applicable to STPEGS.

5.4.7 Residual Heat Removal System

The RHRS transfers heat from the RCS to the Component Cooling Water System (CCWS) to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal plant cooldown and maintains this temperature until the plant is started up again.

Parts of the RHRS also serve as parts of the Safety Injection System (SIS) during the injection and recirculation phases of a LOCA (Section 6.3).

The RHRS also is used to transfer refueling water from the refueling cavity to the refueling water storage tank (RWST) after the refueling operations are completed.

5.4.7.1 Design Bases. RHRS design parameters are listed in Table 5.4-7.
The RHRS is placed in operation following reactor shut-down. Assuming that three HXs and three pumps are in service and that each HX is supplied with CCW at design flow and temperature, the

RHRS is designed to reduce the temperature of the reactor coolant from 350°F to 150°F within 12 hours. The time required under these conditions to reduce reactor coolant temperature from 350°F to 212°F is 6 hours. The heat load handled by the RHRS during the cooldown transient includes residual and decay heat from the core and heat from a single operating RCP. The design heat load is based on the decay heat fraction that exists at 12 hours following reactor shutdown from an run at full power.

Assuming that two HXs and RHR pumps are in service and that the HXs are supplied with CCW at design flow and temperature, the RHRS is capable of reducing the temperature of the reactor coolant from 350°F to 200°F within 17 hours. The time required under these conditions to reduce reactor coolant temperature from 350°F to 212°F is approximately 11 hours.

For the safe shutdown, limiting fire hazard analysis, assuming that only one HX and pump are in service and that the HX is supplied with CCW at design flow and temperature, the RHRS is capable of reducing the temperature of the reactor coolant from 350°F to 200°F (cold shutdown) within 72 hours.

The RHRS is designed to be isolated from the RCS whenever the RCS pressure exceeds the RHRS design pressure. The RHRS is isolated from the RCS on the suction side by two motor-operated valves (MOVs) in series on each suction line. Each MOV is interlocked to prevent its opening if RCS pressure is greater than approximately 350 psig during plant cooldown (refer to the Technical Specifications). Annunciator alarms are provided on the Main Control Boards to alert the Operator of abnormal conditions (increasing RCS pressure and RHR suction isolation valves not fully closed). The RHRS is isolated from the RCS on the discharge side by two check valves in each return line. Also provided on the discharge side is a normally open MOV downstream of each RHRS HX.

Each RHR subsystem is equipped with a pressure relief valve designed to relieve the combined flow of all the charging pumps at the relief valve set pressure of 600 psig. These relief valves also protect the system from inadvertent overpressurization during plant cooldown or startup.

Each STPEGS nuclear unit has an RHRS and there is no sharing of the RHRS between the two units.

The RHRS is designed to be fully operable from the control room for normal operation with the exception of four valves for each train. These valves are the normally closed, MOV closest to the suction of each RHR pump, the normally open MOV downstream of each RHRS HX, and the vent solenoids for the RHR heat exchanger flow and bypass control valves. The motor control center (MCC) breaker for each of the MOVs is power locked out. The fused disconnect switch for each of the vent solenoid valves is closed and the associated train knife switch is open (solenoids deenergized). Manual operations required of the operator from the control room are: opening the suction isolation and miniflow recirculation valves, positioning the flow control valves downstream of the RHRS HXs, and starting the RHR pumps. Manual actions outside the control room are to close the MCC breakers for the motor-operated, pump suction valves that are power locked out and to close the knife switches for the vent solenoid valves. By nature of its redundant three-train design, the RHRS is designed to accept single failures, including the low probability electrical failure of the suction isolation valve interlock circuitry, with the only effect being an extension in the required cooldown time. The arrangement of interlock logic and power train alignment ensure that at least one RHR train can be placed into service assuming the most limiting conditions (i.e., the low probability electrical single failure of one emergency DG in conjunction with LOOP). A detailed description of this arrangement is given in Section 5.4.7.2.6. Although Westinghouse considers it to be of low probability, spurious operation of a single MOV can be accepted without loss of function as a result of the redundant three train design.

Missile protection, protection against dynamic effects associated with the postulated rupture of piping, and seismic design are discussed in Sections 3.5, 3.6, and 3.7, respectively.

5.4.7.2 System Design.

5.4.7.2.1 Schematic Piping and Instrumentation Diagrams: The RHRS, as shown on Figures 5.4-6 (Piping and Instrumentation Diagram) and 5.4-7 (Process Flow Diagram), consists of three RHR HXs, three RHR pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet lines to the RHRS are connected to the hot legs of three RCLs, while the return lines are connected to the cold legs of three RCLs. These return lines are also the SIS cold leg injection lines (Figure 5.4-6).

The RHRS suction lines are isolated from the RCS by two MOVs in series and each discharge line is isolated from the RCS by two check valves and by a normally open MOV. All of these are located inside the Containment.

During RHRS operation, reactor coolant flows from the RCS to the RHR pumps, through the tube side of the RHR HXs, and back to the RCS. The heat is transferred to the CCW circulating through the shell side of the RHR HXs.

Coincident with operation of the RHRS, a portion of the reactor coolant flow may be diverted from downstream of the RHR HXs to the CVCS low pressure letdown line for cleanup and/or pressure control (two of the three RHR trains have this capability). By regulating the diverted flow rate and the charging flow, the RCS pressure may be controlled. Pressure regulation is necessary to maintain the pressure range dictated by nil ductility limits of the reactor vessel and by the no. 1 seal differential pressure and NPSH requirements of the RCPs.

The RCS cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the RHR HXs. A line containing a flow control valve bypasses each RHR HX and is used to maintain a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature, and total flow for each of the trains.

The RHRS is also used to transfer refueling water between the RWST and the refueling cavity. The low head safety injection (LHSI) pumps transfer borated water into the refueling cavity via the RHRS return lines. The RHRS maintains the cold shutdown conditions during refueling. It is then used for emptying the refueling cavity after refueling. The RHR pumps function to transfer water back to the RWST until the water level is brought down to the flange of the reactor vessel. The remainder of the water in the refueling cavity is removed via a drain connection at the bottom of the refueling canal.

When the RHRS is in operation, the water chemistry is the same as that of the reactor coolant. Provision is made for the Process Sampling System (PSS) to extract samples from the flow of reactor coolant in each RHR train, between the pump and HX.

The RHR HXs and the valves and piping which constitute the RHRS return lines also function as part of the SIS. When the RHRS is not in use, each return line is isolated by a check valve from the remainder of its cooling train and is aligned as part of the cold leg safety injection path. The use of the portion of the RHRS as part of the SIS is more completely described in Section 6.3.

The RHR suction isolation valves in each inlet line from the RCS are separately interlocked to prevent their being opened when RCS pressure is greater than approximately 350 psig. This interlock is described in more detail in Sections 5.4.7.2.4 and Section 7.6.2. In addition, annunciator alarms are provided on the Main Control Boards to alert the operator of abnormal conditions (increasing RCS pressure and RHR suction isolation valves not fully closed).

5.4.7.2.2 Equipment and Component Descriptions: The materials used to fabricate RHRS components are in accordance with the applicable Code requirements. All parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or equivalent corrosion resistant material. Component parameters are given in Table 5.4-8.

Residual Heat Removal Pumps

Three pumps are installed in the RHRS. The pumps are sized to deliver reactor coolant flow through the RHR HXs to meet the plant cooldown requirements. The use of three separate RHR trains assures that cooling capacity is only partially lost should one pump become inoperative. A pump performance curve is provided in Figure 5.4-20.

The RHR pumps are protected from overheating and loss of suction flow by miniflow bypass lines that assure flow to the pump suction. A valve located in each miniflow line is under remote manual control by the operator. Prior to starting the RHR pump, the miniflow valve is opened and a pump minimum flow path is established through the miniflow line. Flow instrumentation is provided in the discharge line of each pump to indicate pump flow (Section 5.4.7.2.4).

A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high pressure alarm is also actuated by the pressure sensor. When the RHR flow decreases to a low setpoint, the pump is automatically tripped to prevent pump damage (Section 7.6.6.4). A local pressure gauge in each pump suction line is used for startup and testing purposes. A differential pressure indicator across each RHR pump is also available to facilitate testing activities.

The three pumps are vertical centrifugal units with mechanical seals on the shafts. All pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

RHR Heat Exchangers

Three RHR HXs are installed in the system. The HX design is based on heat load and temperature differences between reactor coolant and CCW existing 12 hours after reactor shutdown when the temperature difference between the two systems is small.

The installation of three HXs in separate and independent RHR trains assures that the heat removal capacity of the system is only partially lost if one train becomes inoperative.

The RHR HXs are of the shell and U-tube type. Reactor coolant circulates through the tubes, while CCW circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

Temperature sensors are located in the system both upstream and downstream of each RHR HX. The temperatures are recorded at the main control board.

The RHR HXs also function as part of the SIS (Section 6.3).

Residual Heat Removal System Valves

Valves that perform a modulating function are equipped with packing.

Manual and MOVs have backseats to facilitate repacking and to limit steam leakage when the valves are open.

5.4.7.2.3 System Operation:

Plant Startup

Generally, while at cold shutdown condition, decay heat from the reactor core is being removed by the RHRS. The number of pumps and HXs in service depends upon the heat load at the time.

At initiation of plant startup, the RHRS is operating and is connected to the CVCS via the low pressure letdown line to control reactor coolant level/pressure. During power operation, the RHR to CVCS letdown isolation valves have no function post accident, and may be power locked out in the closed position to prevent a spurious actuation during a high energy line break event. The RCS is filled and the pressurizer steam bubble is formed as described in section 9.3.4.1.2.6, CVCS System Operations-Reactor Startup. When the pressurizer steam bubble is formed, the RCPs are started for plant heatup. The RHRS pumps are stopped and the RHRS is isolated from the RCS. Pressure is then controlled by normal letdown, pressurizer heaters, and sprays.

Power Generation and Hot Standby Operation

During power generation and hot standby operating, the RHRS is not in service but is aligned for operation as part of the SIS.

Normal Plant Shutdown

A normal plant shutdown is defined as the operation which brings the reactor from no-load temperature and pressure to cold conditions.

The initial phase of normal plant shutdown is accomplished by transferring heat from the RCS to the steam and power conversion system through the use of the SGs.

The second phase of normal plant shutdown starts with the RHRS being placed in operation.

Startup of the RHRS includes a warmup period during which time reactor coolant flow through the HXs is limited to minimize thermal shock. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the tube side of the RHR HXs.

By adjusting the control valves downstream of the RHR HXs, the mixed mean temperature of the return flow is controlled. Using measured total loop return flow, and based upon a return flow setpoint, each HX bypass valve is automatically regulated to give the required total flow.

The reactor cooldown rate is limited by RCS equipment cooling rates based on allowable stress limits, as well as the operating temperature limits of the CCWS. As the reactor coolant temperature decreases, the reactor coolant flow through the RHR HX is increased by adjusting the control valve in each HX tube side outlet line.

As normal plant shutdown continues, the pressurizer is filled with water and the RCS is operated in the water solid condition.

At this stage, pressure control is accomplished by regulating the charging flow rate and the rate of letdown from the RHRS to the CVCS.

After this stage, pressure control is accomplished by regulating the charging flow rate and the rate of letdown from the RHRS to the CVCS.

Refueling

The LHSI pumps are utilized prior to refueling activities to pump borated water from the RWST to fill the refueling cavity. During this operation, the RHR pumps are shut down, the isolation valves in the inlet lines of the RHRS are closed, and the LHSI pumps are utilized to pump borated water from the RWST to the refueling cavity.

The reactor vessel head is lifted slightly. The refueling water is then pumped into the reactor vessel through the normal RHRS return lines and into the refueling cavity through the open reactor vessel. The reactor vessel head is gradually raised as the water level in the refueling cavity increases. After the water level reaches the normal refueling level, the RHR inlet isolation valves are opened, the refueling water storage tank isolation valves are closed, the LHSI pumps are shut down and RHRS operation is resumed.

During refueling, the RHRS is maintained in service with the number of pumps and HXs in operation as required by the heat load.

Following refueling, the RHR pumps are used to drain the refueling cavity to the top of the reactor vessel flange by pumping water from the RCS to the RWST. The remainder of the water is removed via a drain connection at the bottom of the refueling canal.

5.4.7.2.4 Control: Each RHR subsystem is equipped with a pressure relief valve sized to relieve the combined flow of all the charging pumps at the relief valve set pressure. These relief valves also protect the system from inadvertent overpressurization during plant cooldown or startup. Each valve has a relief flow capacity in excess of 1100 gal/min, at a set pressure of 600 psig. An analysis has been conducted to confirm the capability of the RHRS relief valve to prevent overpressurization in the RHRS. All credible events were examined for their potential to overpressurize the RHRS. These events included normal operating conditions, infrequent transients, and abnormal occurrences. The analysis confirmed that one relief valve has the capability to keep the RHRS maximum pressure within Code limits.

The fluid discharged by the relief valves is collected in the pressurizer relief tank.

The design of the RHRS includes two motor-operated gate isolation valves in series on each inlet line between the high pressure RCS and the lower pressure RHRS. They are closed during normal operation and are only opened for RHR during a plant cooldown after the RCS pressure is reduced to approximately 350 psig or lower and RCS temperature is reduced to approximately 350°F. During a plant startup the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above 600 psig. These isolation valves are provided with a "prevent-open" interlock which is designed to prevent possible exposure of the RHRS to normal RCS operating ressure.

The two inlet isolation valves in each RHR subsystem are separately and independently interlocked with pressure signals to prevent their being opened whenever the RCS pressure is greater than approximately 350 psig.

In each RHR subsystem, the interlock signal provided to the isolation valve closest to the RCS is independent and diverse from the interlock signal provided to the isolation valve closest to the RHRS.

The use of two independently powered MOVs in each of the three inlet lines, along with three independent pressure interlock signals for the "prevent-open" function, assures a design which meets applicable single failure criterion. Multiple failures must be postulated to defeat the function of preventing possible exposure of the RHR system to normal RCS operating pressure. These protective interlock designs, in combination with plant operating procedures, provide diverse means of accomplishing the protective function. For further information on the instrumentation and control features, see Section 7.6.2.

The RHR inlet isolation valves are provided with red and green position indication lights on the main control board. These indication lights provide positive position indication for the RHR inlet isolation valves during all modes of operation.

Isolation of the low pressure RHRS from the high pressure RCS is provided on the discharge side by two check valves and one normally open MOV in series. These check valves are described as part of the Emergency Core Cooling System (ECCS) in Section 6.3 and their testing is described in Section 6.3.4.2.

A low-flow interlock is provided for each RHR pump using the pump discharge flow. This interlock trips the pump on low flow and a low flow alarm is provided to the operator. This interlock is described in more detail in Section 7.6.6.4.

5.4.7.2.5 Applicable Codes and Classifications: Each train of the RHRS is designed as Safety Class 2 with the exception of the two isolation valves in each inlet line and the check valves in the outlet lines, which are designed as Safety Class 1. Component codes and classifications are given in Section 3.2.

5.4.7.2.6 System Reliability Considerations: General Design Criterion (GDC) 34 requires that a system to remove residual heat be provided. The safety function of this system is to transfer fission product decay heat and other residual heat from the core at a rate sufficient to prevent fuel or pressure boundary design limits from being exceeded. Safety grade systems are provided in the plant design to perform this safety function. The safety grade systems which perform this function for LOOP and SSE, are: the RCS and SGs, which operate in conjunction with the Auxiliary Feedwater System (AFWS), the SG safety valves, and the SG PORVs; and the RHRS which operates in conjunction with the CCWS and the Essential Cooling Water System (ECWS). For LOCA conditions, the safety grade system which performs the function of removing residual heat from the reactor core is the SIS, which operates in conjunction with the CCWS and the ECWS.

The RHRS is composed of three separate and independent subsystems, each of which contains an RHR pump and HX and associated piping and valves. Each subsystem is connected to a different RCS loop. To assure operability and reliability, the RHR pumps are connected to separate and independent electrical busses so that each pump can receive power from and independent emergency power source if a LOOP should occur. All valves which require electrical power, with the exception of the outermost (from the RCS) inlet isolation valve in each subsystem, are connected to the same bus as their corresponding pump. The outermost inlet isolation valve in each subsystem is connected to a bus which is powered from an emergency power source which is both separate and independent of the source which supplies that subsystem's pump and is also separate and independent of the sources which supply the outermost inlet isolation valves in the other two subsystems. (For further information on electrical power train alignment and instrumentation and control, see Section 7.6.2.)

Both inlet isolation valves in each subsystem are provided with "prevent-open" interlocks (Section 5.4.7.2.4). Three separate and independent pressure interlock signals are generated for each function and are supplied to the valve motor control circuitry in a manner consistent with the valve power supplies described above.

RHRS operation for normal conditions and for major failures is accomplished completely from the control room except for closing the MCC breaker for the MOV closest to the pump suction and the MOV downstream of each RHRS HX, and closing the fused disconnect switch for the vent solenoid valves for the RHR HX flow and bypass control valves. The redundancy in the RHRS design provides the system with the capability to maintain its cooling function even with major single mechanical failures, such as failure of an RHR pump, valve, or HX, since the redundant trains can be used for continued heat removal. The arrangement of interlock logic and power train alignment ensures that at least one RHR train can be put into operation, assuming the most limiting design basis conditions (i.e., LOOP coincident with failure of one emergency power supply). Such an electrical failure is extremely unlikely to occur during the few minutes out of a year when opening suction valves to initiate RHR operations, when it could be of any consequence. However, if such a condition should occur, the one remaining RHR train, in conjunction with the AFWS and SG PORVs, is capable of reducing the RCS temperature from hot shutdown conditions to a cold shutdown value in a reasonable period of time.

A single failure analysis employing Failure Mode and Effects Analysis (FMEA) methodology was conducted for the RHRS. table 5.4.A-2 presents a summary of components included in the analysis. The analysis was limited to the operation of the RHRS during a plant cooldown operation and was bounded by a constraint that only active components performing a fluid system flow function were to be analyzed. Data presented by the table demonstrates that an RHR subsystem can sustain the failure of any single active hydraulic component and the RHR will meet an acceptable level of performance for core cooling in a reasonable time period. The RHRS design configuration analyzed is that shown in Figure 5.4-6. The RHRS shares an emergency core cooling function with the SIS under plant accident conditions. A single failure analysis of RHRS components sharing an SIS function is presented in Section 6.3.

The ability of the RHRS to perform as part of the SIS is not compromised by its normal function, which is normal plant cooldown. The valves associated with the RHRS are normally aligned to allow immediate use of portions of this system as part of the SIS. The system has been designed in such a manner that three separate flow circuits are available, assuring the availability of two trains for safety purposes.

The normal plant cooldown function of the RHRS is accomplished through a suction line arrangement which is independent of any SIS function. The normal cooldown return lines are arranged in separate redundant circuits and utilized also as the cold leg safety injection lines to the RCS. Utilization of the same return circuits for safeguards as well as for normal cooldown lends assurance to the proper functioning of these lines for safeguards purposes.

5.4.7.2.7 Manual Actions: Except for the valves with power disconnected discussed in Section 5.4.7.1, the RHRS is designed to be fully operable from the control room for normal operation. Manual operations required of the operator are: opening the inlet isolation valves, opening and closing of the pump miniflow valves, positioning the flow control valves downstream of the RHRS HXs, opening and closing of the standpipe isolation valves, and starting the RHR pumps. Operation of the RHRS from outside the control room is discussed in Section 7.4.1.9.

As discussed in Section 5.4.7.2.6, at least one RHR train can be put into operation with limited operator action, closing the MCC breaker for the power locked out MOVs and closing the fused disconnect switches for the solenoid valves outside of the control room, assuming the most limiting single failure

5.4.7.3 Performance Evaluation. The performance of the RHRS in reducing reactor coolant temperature is evaluated through the use of heat balance calculations on the RCS and the CCWS at stepwise intervals following in initiation of RHR operation. Heat removal through the RHR and CCW HXs is calculated at each interval by use of standard water-to-water HX performance correlations. The resultant fluid temperatures for the RHR and CCW systems are calculated and used as input to the next interval's heat balance calculation.

Assumptions utilized in the series of heat balance calculations describing normal plant shutdown are as follows:

- RHR operation is initiated four hours after reactor shutdown.
- RHR operation begins at a reactor coolant temperature of 350°F.
- Thermal equilibrium is maintained throughout the RCS during the cooldown.
- CCW temperature during cooldown is limited to a maximum of 120°F for the first four hours of the cooldown and 105°F thereafter.

5.4.7.4 Preoperational Testing. Preoperational testing of the RHRS is addressed in Chapter 14.

5.4.8 Reactor Water Cleanup System

This does not apply to STPEGS.

5.4.9 Main Steam and Feedwater Piping

Refer to Chapter 10.

5.4.10 Pressurizer

5.4.10.1 Design Bases. The general configuration of the pressurizer is shown on Figure 5.4-10. The design data of the pressurizer are given in Table 5.4-9. Codes and material requirements are provided in Section 5.2.

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

5.4.10.1.1 <u>Pressurizer Surge Line</u>: The surge line is sized to limit the pressure drop between the RCS and the safety valves with maximum allowable discharge flow from the safety valves. Overpressure of the RCS does not exceed 110 percent of the design pressure.

The surge line and the thermal sleeve in the pressurizer surge nozzle are designed to withstand the thermal stresses resulting from volume surges of relatively hotter or colder water which may occur during operation.

The 16-in. pressurizer surge line is shown on Figure 5.1-3.

5.4.10.1.2 <u>Pressurizer</u>: The volume of the pressurizer is equal to, or greater than, the minimum volume of steam, water, or total of the two which satisfies all of the following requirements:

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

2. The water volume is sufficient to prevent the heaters from being uncovered during a step load increase of 10 percent of full power.

3. The steam volume is large enough to accommodate the surge resulting from 50 percent reduction of full load with automatic reactor control and 40 percent steam dump without the water level reaching the high level reactor trip point.

4. The steam volume is large enough to prevent water relief through the safety valves following a loss of load, with the high water level initiating a reactor trip without the actuation of accommodating reactor control functions prior to the reactor trip or steam dump.

5. The pressurizer will not empty following reactor trip and turbine trip.

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

5.4.10.2 <u>Design Description</u>.

5.4.10.2.1 <u>Pressurizer Surge Line:</u> The pressurizer surge line connects the pressurizer to one reactor hot leg. The line enables continuous coolant volume and pressure adjustments between the RCS and the pressurizer.

5.4.10.2.2 <u>Pressurizer:</u> The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads, constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant.

The surge line nozzle and removable electrical heaters are installed in the bottom head. The heaters are removable for maintenance or replacement. A thermal sleeve is provided to minimize stresses in the surge line nozzle. A screen at the surge line nozzle prevents foreign matter from entering 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 assist mixing. The spray line nozzle, and relief and safety valve connections are located in the top head of the vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves can also be operated manually by a switch in the control room.

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

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

Material specifications are provided in Table 5.2-2 for the pressurizer, PRT, and the surge line. Design transients for the components of the RCS are discussed in Section 3.9.1. Additional details on the pressurizer design cycle analysis are given in Section 5.4.10.3.5.

Pressurizer Instrumentation

Refer to Chapter 7 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

Spray Line Temperatures

Temperatures in the spray lines from two loops are measured and indicated. Alarms from these signals are actuated by low spray water temperature. Alarm conditions indicate insufficient flow in the spray lines.

5.4.10.3 Design Evaluation.

5.4.10.3.1 System Pressure: The RCS pressure is maintained by the pressurizer whenever a steam bubble is present within the pressurizer. Analyses indicate that proper control of pressure is maintained for the operating conditions.

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

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

During startup and shut down, the rate of temperature change is controlled by the operator. Heatup rate is controlled by pump energy and by the pressurizer electrical heating capacity. This heatup rate takes into account the continuous spray flow provided to the pressurizer. When the reactor core is shut down, the heaters are deenergized.

When the pressurizer is filled with water, i.e., during initial system heatup, and near the end of the second phase of plant cooldown, RCS pressure is maintained by the letdown flow rate via the RHRS.

5.4.10.3.2 Pressurizer Performance: The pressurizer has a minimum free internal volume. The normal operating water volume at full load conditions is a percentage of the free internal vessel volume. Under partial load conditions, the water volume in the vessel is reduced for proportional reductions in plant load. The various plant operating transients are analyzed and the design pressure is not exceeded with the pressurizer design parameters as given in Table 5.4-9.

5.4.10.3.3 Pressure Setpoints: The RCS design and operating pressure, together with the safety, power-operated relief and pressurizer spray valve setpoints, and the protection system setpoint pressures are listed in Table 5.4-16. 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 relief valve characteristics.

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

The pressurizer spray lines and valves are large enough to provide adequate spray using the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg as the driving force. 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. The spray valves and spray line connections are arranged so that the spray will operate when one RCP is not operating. The line may also be used to assist in equalizing the boron concentration between the RCLs and the pressurizer.

A flow path from the CVCS to the pressurizer spray line is also provided. This provides auxiliary spray to the vapor space of the pressurizer during cooldown if the RCPs are not operating. The thermal sleeves on the pressurizer spray nozzle and the spray piping are designed to withstand the thermal stresses resulting from the introduction of cold spray water.

5.4.10.3.5 Pressurizer Design Analysis: The occurrences for pressurizer design cycle analysis are defined as follows:

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

   The only exception occurs when the pressurizer is filled solid during plant startup and cooldown.

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

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

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

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

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

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

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 path is ground smooth for ultrasonic inspection.

- Support skirt to the pressurizer lower head
- Surge nozzle to the lower head
- Nozzles to the safety, relief, and spray lines
- 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.

Peripheral support rings are furnished for the removable insulation modules.

The pressurizer quality assurance program is given in Table 5.4-10.

5.4.11 Pressurizer Relief Discharge System

The pressurizer relief discharge system collects, cools, and directs processing steam and water discharged from the various safety and relief valves in the Containment and from the Reactor Vessel Head Vent System (RVHVS). The system consists of the PRT, the safety and relief valve discharge piping, the PRT spray header and associated piping, the tank nitrogen supply, the gas vent connection, and the drain to the Liquid Waste Processing System (LWPS).

5.4.11.1 Design Basis. Codes and materials of the PRT and associated piping are given in Section 5.2. Design data for the tank are give in Table 5.4-11.

The system design is based on the requirement to absorb a discharge of steam and consequent heat input equivalent to 110 percent of the full power pressurizer steam volume. The steam volume requirement is approximately that which would be experienced if the plant were to suffer a complete loss of load accompanied by a turbine trip but without the resulting direct reactor trip. A delayed reactor trip is assumed in the design of the system for the loss of load condition.

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 95°F. Provision is made to permit cooling the PRT should the water temperature rise above the initial temperature during plant operation. The final design temperature is 200°F, which allows the contents of the PRT, subsequent to a steam discharge, to be drained directly to the LWPS without cooling.

The vessel saddle supports and anchor bolt arrangements are designed to withstand the loadings resulting from a combination of nozzle loadings acting simultaneously with the vessel seismic and static loadings.

5.4.11.2 System Description. The Piping and Instrumentation Diagram (P&ID) for the pressurizer relief discharge system is given on Figure 5.1-4.

The steam and water discharged from the various safety and relief valves inside Containment is routed to the PRT if the discharge fluid is reactor grade quality. Table 5.4-12 provides an itemized list of valves discharging to the tank together with references of the corresponding P&IDs. In addition, the RVHVS discharges to the PRT. The RVHVS is described in Section 5.4.15.

The general configuration of the PRT is shown on Figure 5.4-11. The tank is a horizontal, cylindrical vessel with elliptical dished heads. The vessel is constructed of austenitic stainless steel and is over-pressure protected in accordance with ASME Code Section VIII, Division 1, by means of two safety heads with stainless steel rupture discs.

A flanged nozzle is provided on the tank for the pressurizer relief valve discharge line connection to a sparger pipe. The tank is also equipped with internal spray connected to a cold water inlet and a bottom drain, which are used to cool the tank following a discharge. Cold water is drawn from the reactor makeup water system, or the contents of the tank are circulated through the RCDT HX of the LWPS and back into the spray header.

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 and the relived pressurizer steam is discharged through the sparger pipe located near the bottom, under the water level. The sparger holes are designed to ensure a resultant steam velocity close to sonic.

The 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 calculated using a final tank pressure of 50 psig based on design conditions. Consequently, the design discharge for the worst case initial conditions will raise internal tank pressure to a maximum of 50 psig, a pressure low enough to prevent fatigue of the rupture disks. Provision is made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and oxygen.

The contents of the vessel can be drained to the waste holdup tank in the LWPS or the recycle holdup tank in the Boron Recycle System (BRS) via the RCDT pumps in the LWPS.

5.4.11.3 Safety Evaluation. The pressurizer relief discharge system does not constitute part of the RCPB per 10CFR50, Section 50.2, since all of its components are downstream of the RCS safety and relief valves. Thus, GDCs 14 and 15 are not applicable. Furthermore, failure of the pressurizer relief system will not impair the capability for safe plant shutdown.

No restraints are necessary on the safety and relief valve discharge piping to ensure the integrity and operability of the valves. RG 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. The volume of nitrogen in the PRT is that which is required to limit the maximum tank pressure to 50 psig from a design basis discharge. The volume of water in the pressurizer relief tank is capable of absorbing the heat from the assumed discharge maintaining the water temperature below 200°F. If a discharge exceeding the design basis should occur, the rupture discs on the tank ensure overpressure protection by providing means for passing the discharge through the tank to the Containment.

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 design pressure is twice the calculated pressure resulting from the design basis safety valve discharge described in Section 5.4.11.1. The tank and rupture disc 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 safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20 percent of the setpoint pressure at full flow.

5.4.11.4 Instrumentation Requirements. The PRT pressure transmitter provides both an indication of PRT pressure and a high tank pressure alarm on the main control panel.

The PRT level transmitter supplies a signal for both indication and high and low level alarms on the main control panel.

The temperature of the water in the PRT is indicated on the main control panel. Also provided on the main control panel is a high temperature alarm to inform the operator that cooling of the tank contents is required.

5.4.11.5 Inspection and Testing Requirements. The system components are subject to nondestructive and hydrostatic testing during construction in accordance with Section VIII, Division 1 of the ASME Code.

During plant operation, periodic visual inspections and preventive maintenance are conducted on the system components according to normal industrial practice.

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 normally closed or capable of automatic or remote closure larger than 3/4 in., are American Nuclear Society (ANS) Safety Class 1, Section III of the ASME Code and Code Class 1 valves. All 3/4-inch or smaller valves in lines connected to the RCS are Class 2 since the interface with the Class 1 piping is provided with suitable orificing for such valves. Design data for the RCS valves are given in Table 5.4-13.

For a check valve to qualify as part of the RCS it must be located inside the Containment. When the second of two normally open check valves is considered part of the RCS (as defined in Section 5.1), means are provided to periodically assess backflow leakage of the first valve when closed.

To ensure that the valves will meet the design objectives, the materials of construction minimize corrosion/erosion and ensure compatibility with the environment, leakage is minimized to the extent practicable by Code design, and Class 1 stresses are maintained within the limits of the ASME Code, Section III.

5.4.12.2 Design Description. All manual and MOVs of the RCS which are 3 in. and larger are provided with double-packed stuffing boxes and intermediate lantern ring leakoff connections. All throttling control valves are provided with double-packed stuffing boxes and with stem leakoff connections. In general, RCS leakoff connections are piped to a closed collection system. Leakage to the atmosphere is minimized for these valves.

Gate valves at the engineered safety features (ESF) interface are wedge design and are essentially straight-through. The wedges are flex-wedge or solid. All gate valves have backseats. Check valves are swing type for sizes 3 in. and 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.

The accumulator check valve is designed such that at the required flow, the resulting pressure drop is within the specified limits. All operating parts are contained within the 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/analysis requirements for Class 1 valves, as discussed in Section 5.2, limit stresses to levels which ensure structural integrity. In addition, the testing programs described in Section 3.9 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 assure the compatibility of valve construction materials with the reactor coolant. To ensure that the reactor coolant continues to meet these parameters, the chemical composition of the coolant will be analyzed periodically.

The above requirements and procedures, coupled with the previously described design features for minimizing leakage, ensure that the valves will perform their intended functions as required during plant operation.

5.4.12.4 Tests and Inspections. All RCS valves are tested in accordance with the requirements of the ASME Code, Section III. The tests and inspections discussed in Section 3.9 are performed to ensure the operability of active valves. In-place operational testing, as defined in the ASME Code, Section XI, is performed on valves.

There are no full-penetration welds within valve body walls. Valves are accessible for disassembly and internal visual inspection to the extent practicable. Plant layout configurations determine the degree of inspectability. Valve nondestructive examinations are given in Table 5.4-14. In-service inspection is discussed in Section 5.2.4.

5.4.13 Safety and Relief Valves

5.4.13.1 Design Bases. The combined capacity of the pressurizer safety valves is that required to accommodate the maximum surge resulting from complete loss of load. This objective is met assuming conditions as noted in Section 5.2.2.1.

The RCS utilizes various pressure control equipment in conjunction with the ASME Code safety valves. Although this pressure control equipment is not required by the ASME Code, it is used to assist in maintaining the RCS within the normal operating pressure.

The pressurizer PORVs are designed to limit pressurizer pressure to a value below the fixed high pressure reactor trip setpoint. They are designed to fail to the closed position on loss of power supply.

The pressurizer PORVs are not required to open in order to prevent the overpressurization of the RCS. The pressurizer safety valves, by themselves, are sized to relieve enough steam to prevent an overpressurization of the primary system. Therefore, a loss of pressurizer PORV automatic control, and the subsequent failure of the PORVs to open, will result in higher reactor coolant pressures, but will not cause any overpressurization problems. In fact, the opening of the PORV is a conservative assumption for the departure from nucleate boiling (DNB) limited transients by tending to keep the primary system pressure down.

The pressurizer spray valves are also utilized to control pressurizer pressure variations. During an insurge, the spray system, which is fed from two cold legs, condenses steam in the pressurizer to prevent the pressure from reaching the setpoint of the PORVs.

5.4.13.2 Design Description. The pressurizer safety valves are of the pop type. The valves are spring-loaded, opened by direct fluid pressure action, and have backpressure compensation features. Position indication and alarm for each safety valve is provided in the control room.

The 6-in. pipes connecting the pressurizer nozzles to their respective code safety valves are shaped in the form of a loop seal. Condensate resulting from normal heat losses accumulates in the loop. The water prevents any leakage of hydrogen gas or steam through the safety valves seats. If the pressurizer pressure exceeds the set pressure of the safety valves, they start lifting, and the water from the seal discharges during the accumulation period.

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

The pressurizer PORVs provide the safety-related means for RCS depressurization to achieve cold shutdown.

Safety-related position indication for the pressurizer PORVs is provided by stem-actuated magnetic reed switches. Valve position is indicated and alarmed in the control room and indicated on the auxiliary shutdown panel.

Temperatures in the pressurizer safety and relief valve discharge lines are measured, indicated, and alarmed in the control room. An increase in a discharge line temperature is an indication of leakage through the associated valve.

The spray valves on the pressurizer are modulating air-operated valves which also respond to a signal from pressure sensing instrumentation. These valves can also be controlled manually from the control room.

Design parameters for the pressurizer spray safety and PORVs are given in Table 5.4-15.

5.4.13.3 Design Evaluation. The pressurizer safety valves prevent RCS pressure from exceeding 110 percent of system design pressure, in compliance with the ASME Code, Section III.

The pressurizer PORVs prevent actuation of the fixed reactor high pressure trip for all design transients up to and including the design step load decrease with steam dump. The relief valves also limit undesirable opening of the spring-loaded safety valves. Note that setpoint studies to date indicate that the pressure rise in a four-loop plant for the design step load decrease of 10 percent from full power is limited to 60 psi. The pressure rise is not sufficient to actuate the PORVs, and thus this design is conservative.

The pressurizer spray valves help to prevent actuation of the PORVs. The spray rate is selected to prevent the pressurizer pressure from reaching the operating setpoint of the PORVs following a step load reduction in power of ten percent of full load with reactor control.

5.4.13.4 Tests and Inspections. All safety and relief valves are subjected to hydrostatic tests, seal leakage tests, operational tests, and inspections as required. For safety and relief valves that are required to function during a faulted condition, additional tests are performed. These tests are described in Section 3.9.

5.4.14 Component Supports

5.4.14.1 Design Bases. Component supports allow virtually unrestrained lateral thermal movement of the loop during plant operation and provide restraint to the loops and components during accident conditions. As discussed in Reference 3.6-14 and Section 3.6.2.1.1.1a, RCL ruptures and the associated dynamic effects are not included in the design bases. The loading combinations and design stress limits are discussed in Sections 3.8.3.3.2.3 and 3.9.1.4.1. Support design is in accordance with the ASME Code, Section III, Subsection NF. The design maintains the integrity of the RCS boundary for normal and accident conditions. Results of support stress evaluations are presented in Section 3.9.1.4.4.

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 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 embedded anchor bolts or fabricated assemblies.

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

Because of manufacturing and construction tolerances, ample adjustment in the support structures must be 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.

Reactor Pressure Vessel

Supports for the reactor vessel (Figure 5.4-12) are individual air-cooled rectangular box structures beneath the vessel nozzles bolted to the primary shield wall concrete. Each box structure consists of a horizontal top plate that receives loads from the reactor vessel shoe, a horizontal bottom plate supported by and transferring loads to the primary shield wall concrete, and connecting vertical plates. The supports are air-cooled to maintain the supporting concrete temperature within acceptable levels.

Steam Generator

As shown on Figure 5.4-13, the SG supports consist of the following elements:

1. Vertical Support

Four individual columns provide vertical support for each SG. These are bolted at the top to the SG 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 SG during heatup and cooldown. The column base design permits both horizontal and vertical adjustment of the SG for erection and adjustment of the system.

2. 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 SG but permit unrestrained movement of the SG during changes in system temperature. Stresses in the beam caused by wall displacements during compartment pressurization and the building seismic evaluation are considered in the design.

3. Upper Lateral Support

The upper lateral support of the SG is provided by a built-up ring plate girder at the operating deck. Two-way acting snubbers restrain sudden seismic or blowdown induced motion, but permit the normal thermal movement of the SG. Movement perpendicular to the thermal growth direction of the SG is prevented by struts.

Rector Coolant Pump

Three individual columns, similar to those used for the SG, provide the vertical support for each pump. Lateral support for seismic and blowdown loading is provided by tension tie bars and compression struts. The pump supports are shown on Figure 5.4-14.

Pressurizer

The supports for the pressurizer, as shown in Figure 5.4-15, consist of:

1. A steel ring plate between the pressurizer skirt and the supporting concrete slab. The ring serves as a leveling and adjusting member for the pressurizer and may also be used as a template for positioning the concrete anchor bolts.

2. The upper lateral support consist of struts cantilevered off the compartment walls that bear against the "seismic lugs" provided on the pressurizer.

5.4.14.3 Evaluation. A detailed evaluation ensures the design adequacy and structural integrity of the RCL 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 a large or small seismic disturbance and/or LOCA conditions. Loads which the system is expected to encounter often during its lifetime (thermal, weight, pressure) are applied and stresses are compared to allowable values as described in Section 3.9.1.4.7.

The SSE and design basis LOCA resulting in a rapid depressurization of the system are required design conditions for public health and safety. The methods used for the analyses of the SSE and LOCA conditions are given in Section 3.9.1.4.

The reactor vessel supports are not designed to provide restraint for vertical uplift movement resulting from seismic and pipe break loadings. However, reactor pressure vessel (RPV) motion resulting from seismic and pipe break events is conservatively included in the RCS analyses described in Section 3.9.1.4.

Thermal analyses are performed for the RPV supports. Thermal growth of the supports are included in the RCS analyses as thermal anchor movement.

5.4.14.4 Tests and Inspections. Weld inspection and standards are specified in accordance with Section V of the ASME Code. Welder qualifications and welding procedures are specified in accordance with Section IX, ASME Code.

5.4.15 Reactor Vessel Head Vent System

The RVHVS (Figure 5.1-1) removes noncondensible gases and permits letdown from the reactor vessel head. This system is designed to mitigate a possible condition of inadequate core cooling or impaired natural circulation resulting from the accumulation of noncondensible gases in the RCS.

5.4.15.1 Design Bases. The RVHVS is designed to remove noncondensible gases or steam from the RCS via remote manual operations from the control room. The system discharges to the PRT.

The system provides for venting the reactor vessel head by using only safety-related equipment in accordance with the requirements of Three Mile Island (TMI) action plan item II.B.1 of NUREG-0737. The system also provides a safety grade letdown path for safety grade cold shutdown (Appendix 5.4.A). The RVHVS satisfies applicable requirements and industry standards, including ASME Code classification, safety classification, single-failure criteria, and environmental qualification.

All piping and equipment from the vessel head vent up to and including the second isolation valve in each flow path are designed and fabricated in accordance with ASME Section III, Class 1 requirements. The piping and equipment in the flow paths from the second isolation valve to the throttling valves are designed and fabricated in accordance with ASME Section III, Class 2 requirements. The remainder of the piping and equipment is seismically designed, non-nuclear safety.

All supports and support structures conform with the requirements of the ASME Code.

5.4.15.2 System Description. The RVHVS consists of single-active-failure-proof flow paths with redundant isolation valves. The equipment design parameters are listed in Table 5.4-17.

The active portion of the system consists of four open/close solenoid-actuated isolation valves connected in two parallel paths to the 1-in head vent pipe, which is located near the center of the reactor vessel head. The system design with two valves in series in each flow path minimizes the possibility of RCPB 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. Downstream of these isolation valves are two solenoid-operated throttling valves in parallel, used to control the letdown flow rate. These throttling valves are controlled through the QDPS, described in Section 7.5.6. All of these valves are fail-closed, normally closed valves. The valves are included in the valve operability program and are qualified to Institute of Electrical and Electronics Engineers (IEEE) 323-1974, 344-1975, and 382-1972.

The vent system piping is supported to ensure that the resulting loads and stresses on the piping and on the vent connection to the vessel head are acceptable.

5.4.15.3 Safety Evaluation. If a single active failure prevents a venting operation through one flow path, the redundant path is available for venting. With two fail-closed valves in series, the failure of any one valve or power supply will not inadvertently open a vent path or prevent closing a flow path. Thus, the combination of safety –related train assignments and valve failure modes will not prevent vessel head venting or venting isolation with any single active failure.

The RVHVS has two normally deenergized valves in series in each flow path. This arrangement eliminates the possibility of an opened flow path due to the spurious movement of one valve. As such, power lockout to any valve is not considered necessary.

A break of the RVHVS line would result in a small LOCA of not greater than 1-inch-diameter. Such a break is similar to those analyzed in WCAP-9600. Since a break in the head vent line would behave similarly to the hot leg break case presented in WCAP-9600, the results presented therein are applicable to a RVHVS line break. Therefore, this postulated vent line break results in no calculated core uncovery. It should be noted that the reactor vessel head vent system isolation valves are normally closed, fail closed valves. These valves are pilot-operated solenoid valves with flow "over" the disk. When one of these valves is deenergized, the pilot closes, creating a differential pressure across the main disc. This pressure differential closes the main disc with the flow over the disc increasing the closing speed. Thus, the greater the flow through the valve, the more rapidly the valve closes. Therefore, there will be no adverse impact on RHVS isolation valve closure.

5.4.15.4 Tests and Inspections. inservice inspection is conducted in accordance with Section 5.2.4.

5.4.15.5 Instrumentation. The system is operated from the control room or the auxiliary shutdown panel. The isolation valves have stem position switches. The position indication from each valve is monitored at the control room or the auxiliary shutdown panel by position indication lights. The throttling valves position indication is provided in the control room and at the auxiliary shutdown panel through the QDPS.

REFERENCES

Section 5.4:

5.4-1 "Reactor Coolant Pump Integrity in LOCA", WCAP-8163, September 1973.

5.4-2 "Dynamic Fracture Toughness Properties of Heavy Section A533 Grade B Class 1 Steel Plate", WCAP-7623, December 1970.

5.4-3 "Evaluation of Steam Generator Tube, Tube Sheet and Divider Plate under Combined LOCA plus SSE Conditions", WCAP-7832, October 1974.

5.4-4 "Report on small break accidents for Westinghouse NSSS", WCAP 9600, June 1979.

5.4-5 "Independent Evaluation of Proposed Modification to Westinghouse D4, D5 and E Steam Generators", CSGORG-001 July 15, 1983.

5.4-6 WCAP-15095, "Steam Generator tube Plugging Limit Analysis for South Texas Unit 1 Delta 94 Steam Generators."

TABLE 5.4-1

REACTOR COOLANT PUMP DESIGN PARAMETERS

Design pressure, psig	2,485
Design temperature, °F	650*
Overall height, ft	27.2
Seal water injection, gal/min	8
Seal water return, gal/min	3
Cooling water flow, gal/min	496
Maximum continuous cooling water inlet temperature, °F	105
Total (motor and pump) moment of inertia, lb-ft^2	110,000
Pump	
Capacity, gal/min	102,500
Developed head, ft	341
NPSH required, ft	Figure 5.4-2
Suction temperature, °F	559.7
Pump discharge nozzle, inside diameter, in.	27-1/2
Pump suction nozzle, inside diameter, in.	31
Speed, rpm, nominal	1,200
Water volume, ft^3	128**
Weight (dry), lbs	219,956
Motor	
Type	Drip-proof, squirrel-cage induction, water/air-cooled
Power, hp	8000
Voltage, volts	13,200
Phase	3
Frequency, Hz	60
Insulation class	Class F, thermalastic epoxy insulation
Current	
Starting	1800 amp @ 13,200 volts
Running, hot reactor coolant	264 amp
Running, cold reactor coolant	346 amp

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* 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 primary loop temperature of 650°F.

** Composed of reactor coolant in the casing and of injection and cooling water in thermal barrier.

TABLE 5.4-2

REACTOR COOLANT PUMP QUALITY ASSURANCE PROGRAM

	RT*	UT	PT	MT
Castings (Pressure-Retaining)	yes		yes	
Forgings				
1. main shaft – pump		yes	yes	
2. main studs		yes		
3. flywheel (rolled plate)		yes		
flywheel (finished)		yes	**	**
Weldments				
1. circumferential	yes		yes	
2. instrument connections			yes	

* RT – Radiographic
UT – Ultrasonic
PT – Dye Penetrant
MT – Magnetic Particle

** Finished machined bores, keyways, and drilled holes within 4 inches of the bore are subjected to magnetic particle or dye penetrant examinations.

TABLE 5.4-3

STEAM GENERATOR DESIGN DATA

Shell Design Pressure, reactor coolant side, psig	2,485
Shell Design Pressure, steam side, psig	1,285
Design Temperature, reactor coolant side, °F	650
Design Temperature, steam side, °F	600
Total Heat Transfer Surface Area, ft^2	94,500
Maximum Moisture Carryover, wt percent	0.10
Overall Height, ft-in	72'-6"
Number of U-tubes	7,585
U-Tube Nominal Diameter, in.	0.688
Tube Wall Nominal Thickness, in.	0.040
Number of Manways (Primary Side)	2
(Secondary Side)	2
Inside Diameter of Manways, in. (Primary Side)	18
(Secondary Side)	16
Number of Inspection Ports	18
Design Fouling Factor	0.00011
Handholes	6
Inside Diameter of Handholes, in.	6.00

TABLE 5.4-4

STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	RT*	UT	PT	MT	ET
Tubesheet					
Forging		yes		yes	
Cladding		yes**	yes		
Channel Head					
Cladding			yes		
Forging		yes		yes	
Secondary Shell & Head					
Plates		yes			
Tubes		yes			yes
Forging		yes		yes	
Nozzles (Forgings)		yes		yes	
Weldments					
Shell, longitudinal		yes			yes
Shell, circumferential	yes			yes	
Cladding (channel head-tube sheet joint cladding restoration)			yes		
Steam and feedwater nozzle to shell	yes			yes	
Support brackets				yes	
Tube to tubesheet			yes		
Instrument connections (primary and secondary)				yes	
Temporary attachments (after removal)				yes	
After hydrostatic test (all welds and complete cast channel head – where accessible)				yes	
Nozzle safe ends (if forgings)	yes		yes		
Nozzle safe ends (if weld deposit)		yes			

* RT – Radiographic
UT – Ultrasonic
PT – Dye Penetrant
MT – Magnetic Particle
ET – Eddy Current

** Flat Surfaces only

TABLE 5.4-5

REACTOR COOLANT PIPING DESIGN PARAMETERS

Reactor Vessel Inlet Piping, inside diameter, in.	27-1/2
Reactor Vessel Inlet Piping, nominal wall thickness, in.	2.32
Reactor Vessel Outlet Piping, inside diameter, in.	29
Reactor Vessel Outlet Piping, nominal wall thickness, in.	2.45
Coolant Pump Suction Piping, inside diameter, in.	31
Coolant Pump Suction Piping, nominal wall thickness, in.	2.60
Pressurizer Surge Line Piping, nominal pipe size, in.	16
Pressurizer Surge Line Piping, minimum design thickness, in.	1.171
Reactor Coolant Loop Piping	
Design pressure, psig	2,485
Design temperature, °F	650
Pressurizer Surge Line	
Design pressure, psig	2,485
Design temperature, °F	680
Pressurizer Safety Valve Inlet Line	
Design pressure, psig	2,485
Design temperature, °F	680
Pressurizer Power-Operated Relief Valve Inlet Line	
Design pressure, psig	2,485
Design temperature, °F	680
Pressurizer Relief Tank Inlet Line	
Design pressure, psig	700
Design temperature, °F	600
Pressurizer Safety and Relief Valve Discharge Lines	
Design pressure, psig	500
Design temperature, °F	470

TABLE 5.4-6

REACTOR COOLANT PIPING QUALITY ASSURANCE PROGRAM

	RT*	UT	PT
Fittings and Pipe (Castings)	yes		yes
Fittings and Pipe (Forgings)		yes	yes
Weldments			
1. Circumferential	yes		Yes
2. Nozzle to runpipe (except no RT for nozzles less than 6 inches)	yes		Yes
3. Instrument connections			Yes
Castings	yes		yes (after finishing)
Forgings		yes	yes (after finishing)

* RT – Radiographic
UT – Ultrasonic
PT – Dye Penetrant

TABLE 5.4-7

DESIGN BASES FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION

Residual Heat Removal System startup	~4 hours after Reactor Shutdown
Reactor Coolant System initial pressure, psig	~350
Reactor Coolant System initial temperature, °F	~350
Component cooling water design temperature, °F	105
Cooldown time, hours after initiation of Residual Heat Removal System operation (3 trains operating)	~12
Reactor Coolant System Temperature at end of cooldown, °F	150
Decay heat generation at 12 hours after reactor shutdown, Btu/hr	~98 x 10^6

TABLE 5.4-8

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pump

Number	3
Design pressure, psig	600
Design temperature, °F	400
Design flow, gal/min	3,400
Design head, ft	205
NPSH required at 3400 gal/min, ft	11.5
Motor rating, hp	300

RHR Heat Exchanger

Number	3
Design heat removal capacity, Btu/hr (each)	39.4×10^6
Estimated UA, Btu/hr - °F	2.0×10^6

	Tube Side	Shell Side
Design pressure, psig	600	150
Design temperature, °F	400	200
Design flow, lb/hr	1.5×10^6	2.45×10^6
Inlet temperature, °F	150	105
Outlet temperature, °F	123.7	121.1
Material	Austenitic Stainless Steel	Carbon Steel
Fluid	Reactor Coolant	Component Cooling Water

TABLE 5.4-9

PRESSURIZER DESIGN DATA

Design pressure, psig	2,485
Design temperature, °F	680
Surge line nozzle diameter, in.	16
Heatup rate of pressurizer using heaters only, °F/hr	55
Internal volume, ft^3	2,100

TABLE 5.4-10

PRESSURIZER QUALITY ASSURANCE PROGRAM

	RT*	UT	PT	MT
Heads				
1. Plates		yes		
2. Cladding			yes	
Shell				
1. Plates		yes		
2. Cladding			yes	
Heaters				
1. Tubing***		yes	yes	
2. Centering of element	yes			
Nozzle (Forgings)		yes	yes**	yes**
Weldments				
1. Shell, longitudinal	yes			yes
2. Shell, circumferential	yes			yes
3. Cladding			yes	
4. Nozzle safe end (if forging)	yes		yes	
5. Instrument connections			yes	
6. Support skirt, longitudinal seam	yes			yes
7. Support skirt to lower dead		yes		yes
8. Temporary attachments (after removal)				yes
9. All external pressure boundary welds after shop hydrostatic test				yes

* RT – Radiographic
UT – Ultrasonic
PT – Dye Penetrant
MT – Magnetic Particle

** – MT or PT

*** – Or a UT and ET

TABLE 5.4-11

PRESSURIZER RELIEF TANK DESIGN DATA

Design Pressure, psig		
Internal		100
External		15
Rupture Disc Release Pressure, psig	Nominal:	91
	Range:	86-100
Design Temperature, °F		340
Total Rupture Disc Relief Capacity, lb/hr at 100 psig		1.6×10^6

TABLE 5.4-12

VALVE DISCHARGE TO THE PRESSURIZER RELIEF TANK

Reactor Coolant System		
3	Pressurizer safety valves	Figure 5.1-3
2	Pressurizer power-operated relief valves	Figure 5.1-3
1	Reactor vessel head vent system	Figure 5.1-1
Residual Heat Removal System		
3	Residual heat removal pump discharge lines to the reactor coolant system cold legs relief valves	Figure 5.4-6
Chemical and Volume Control System		
1	Seal water return line	Figure 9.3.4-1
1	Letdown line	Figure 9.3.4-1
Safety Injection System		
3	Low Head Safety Injection discharge line to RCS	Figures 6.3-1 through 6.3-3
3	High Head Safety Injection discharge line to RCS	Figures 6.3-1 through 6.3-3

TABLE 5.4-13

REACTOR COOLANT SYSTEM VALVE DESIGN PARAMETERS

Design Pressure, psig	2,485
Pre-Operational Plant Hydrotest, psig	3,107
Design Temperature, °F	650

TABLE 5.4-14

REACTOR COOLANT SYSTEM VALVES NONDESTRUCTIVE EXAMINATION

Valve Body Type		RT*	UT	PT
Castings	(larger than 4 inches)	yes		yes
	(2 inches to 4 inches)	yes[(1)]		yes
Forgings	(larger than 4 inches)	(2)	(2)	yes
	(2 inches to 4 inches)			yes

RT – Radiographic

UT – Ultrasonic

PT – Dye Penetrant

(1) – Weld ends only

(2) – Either RT or UT

TABLE 5.4-15

PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Spray Valves	
Number	2
Design pressure, psig	2,485
Design temperature, °F	650
Design flow for valves full open, each gal/min	525
Pressurizer Safety Valves	
Number	3
Relieving capacity (at 3% accumulation), ASME rated flow, lb/hr	504,953
Set pressure, psig	2,485
Design Temperature, °F	650
Fluid	Saturated steam
Transient Condition, °F	(Superheated steam) 680
Backpressure:	
Normal, psig	3 to 5
Expected during discharge, psig	500
Pressurizer Power-Operated Relief Valves	
Number	2
Design pressure, psig	2,485
Design temperature, °F	650
Relieving capacity at 660°F and 2,385 psig, lb/hr (per valve)	210,000
Fluid	Saturated steam
Transient Condition, °F	(Superheated steam) 680

TABLE 5.4-16

REACTOR COOLANT SYSTEM NOMINAL PRESSURE SETTINGS

	psig
Hydrostatic Test Pressure	3,107
Design Pressure	2,485
Safety Valves (Begin to Open)	2,485
High Pressure Reactor Trip	2,380
High Pressure Alarm	2,310
Power-Operated Relief Valves	2,335*
Pressurizer Spray Valves (Full Open)	2,310
Pressurizer Spray Valves (Begin to Open)	2,260
Proportional Heaters (Begin to Operate)	2,250
Proportional Heaters (Full Operation)	2,220
Backup Heaters On	2,210
Low Pressure Alarm	2,210
Pressurizer Power-Operated Relief Valve Interlock	2,185
Low Pressure Reactor Trip	1,870

* At 2,335 psig, a pressure signal initiates actuation (opening) of these valves. Remote manual control is also provided.

TABLE 5.4-17

REACTOR VESSEL HEAD VENT SYSTEM EQUIPMENT DESIGN PARAMETERS

Valves	
Number (includes one manual valve)	7
Design pressure, psig	2,485
Design temperatures, °F	650
Piping	
Vent line, nominal diameter, in.	1
Design pressure, psig	2,485
Design temperature, °F	650
Maximum operating temperature, °F	620

NOTES TO FIGURE 5.4-7

Modes of Operation

Mode A: Initiation of RHR Operation

Following reactor shutdown, the second phase of plant cooldown starts with the RHRS being placed in operation. Before starting the pumps, the inlet isolation valves are opened, the miniflow valves are opened, the heat exchanger flow control valves are set at minimum flow, and the outlet valves are verified open. The miniflow valves are manually closed when flow is established to the RCS cold legs. Should the pump flow drop below the low flow set point the pump will automatically be shut off and an alarm will sound in the control room.

Startup of the RHRS includes a warm-up period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock on the RCS. The rate of heat removal from the reactor coolant is controlled manually by regulating the reactor coolant flow through the RHR heat exchangers. The total flow is regulated automatically by a control valve in each heat exchanger bypass line to maintain a constant total flow. The cooldown rate is limited to 100 F/hr based on equipment stress limits and a 120 F maximum component cooling water temperature.

Mode B: End Conditions of a Normal Cooldown

This situation characterizes most of the RHRS operation. As the reactor coolant temperature decreases, the flow through the RHR heat exchanger is increased until all of the flow is directed through the heat exchanger to obtain maximum cooling.

Note: The Emergency Core Cooling System components shown on the process flow diagram are not used for normal residual heat removal operations. The process conditions associated with the Emergency Core Cooling System operation (location numbers 1 through 37) are presented in the notes to Figure 6.3-5.

APPENDIX 5.4.A

COLD SHUTDOWN CAPABILITY

Response to Branch Technical

Position RSB 5-1

5.4.A.1 INTRODUCTION

While the safe shutdown design basis for the South Texas Project Electric Generating Station (STPEGS) is hot standby, the cold shutdown capability of the plant has been evaluated in order to demonstrate how the plant can achieve cold shutdown conditions following a Safe Shutdown Earthquake (SSE), assuming Loss-of-Offsite Power (LOOP) and the most limiting single failure. Under such conditions, the plant is capable of achieving residual heat removal system (RHRS) initiation conditions within 36 hours.

This Appendix documents the degree of compliance of the STPEGS design with the requirements as outlined in Branch Technical Position (BTP) RSB 5-1, Revision 2. Table 5.4.A-1 provides a brief line-by-line comparison of the STPEGS design with the requirements listed in Table 1 of the BTP.

The cold shutdown functions described in this Appendix are controlled and monitored from the control room. For a discussion of safe shutdown using controls and indications entirely outside of the control room, see Section 7.4.1.9.

5.4.A.2 SYSTEMS REQUIRED TO GO FROM HOT STANDBY TO COLD SHUTDOWN

In order to safely shut down the plant, the following functions must be performed:

1. Circulation of the reactor coolant
2. Heat removal (short-term and long-term)
3. Boration and inventory control
4. Depressurization

The systems that are required to achieve and maintain a cold shutdown have redundancy/diversity, and no single failure will compromise safety functions. All power supplies and control functions for required portions of these systems are Class 1E, as described in Chapters 7 and 8. As discussed in Section 3.2, all components meet the requirements of Regulatory Guides (RGs) 1.26 and 1.29. The following are the major systems that are employed to achieve and maintain a safe shutdown:

1. Reactor Coolant System (RCS) (Chapter 5)
2. Main Steam (MS) System (Section 10.3)
3. Auxiliary Feedwater System (AFWS) (Section 10.4.9)
4. Chemical and Volume Control System (CVCS) (Section 9.3.4)

5. Safety Injection System (SIS) (Section 6.3)

6. Residual Heat Removal System (RHRS) (Section 5.4.7)

7. Component Cooling Water Systems (CCWS) (Section 9.2.2)

8. Essential Cooling Water System (ECWS) (Section 9.2.1.2)

9. Supportive Heating, Ventilating and Air Conditioning (HVAC) (Section 9.4)

10. Standby Diesel Generators (DGs) (Sections 8.3 and 9.5.4 through 9.5.8) and associated electrical distribution systems

11. Reactor Makeup Water System (RMWS) (Section 9.2.7)

Each of the system descriptions in the respective sections identify the integral role that the system plays in achieving and maintaining a safe shutdown. Instrumentation applications for safe shutdown are described in this section and in Sections 7.4 and 7.5.

5.4.A.3 METHODOLOGY TO ACHIEVE AND MAINTAIN COLD SHUTDOWN

It is expected that the systems normally used for cold shutdown will be available should the operator choose to perform a reactor cooldown. However, to ensure that the plant can be taken to cold shutdown following an SSE, the STPEGS cold shutdown design enables the RCS to be taken from no-load temperature and pressure to cold conditions using only safety-related systems, with only onsite or offsite power available, and assuming the most limiting single failure.

Should portions of the normal shutdown systems be unavailable, the operator should maintain the plant in a hot standby condition while making the normal systems functional. Local manual actions could be performed where such are permitted by the prevailing environmental conditions. Appropriate procedures are provided for the use of safety-related backups contingent upon the unavailability of the normal systems. The operator should use any of the normal systems that are available in combination with the safety-related backups for the systems that cannot be made operable.

The STPEGS cold shutdown design enables the operator to maintain the plant in hot standby until manual actions or repairs are completed. Since it is assumed that the reactor coolant pumps (RCPs) are not available, circulation of the reactor coolant is provided by natural circulation with the reactor core as the heat source and the steam generators (SGs) as the heat sink. Heat removal is accomplished via the SG power-operated relief valves (PORVs) and auxiliary feedwater system.

The charging pumps are used to deliver 4-wt % boric acid from the boric acid tanks to the RCS at a rate of approximately 50 gal/min. The borated water is delivered to the RCS via the safety-related charging line or the RCP seal injection lines. The safety-related charging line is provided with flow control capability. Charging flow via the RCP seal injection lines is predetermined by the manual valves in each line. Both flow paths have provisions for flow monitoring. To accommodate this additional RCS inventory, letdown may be discharged from the reactor vessel head vent line with letdown routed to the pressurizer relief tank (PRT). The head vent throttling valves are controlled through the Qualified Display Processing System (QDPS) described in Section 7.5.6.

The safety-related cooldown is accomplished by increasing the steam release from the SG PORVs to attain a rate of primary side cooling of 15 - 20ºF/hr. The SG PORVs are also controlled through the QDPS. In conjunction with this portion of the cooldown, the charging pumps are used to deliver water to makeup for primary system contraction due to cooling. Makeup is also required for inventory control in the event the reactor vessel head vent path is periodically opened to provide head cooling prior to RCS depressurization. The safety grade auxiliary feedwater storage tank (AFST) has adequate capacity to accommodate the identified RCS cooldown and depressurization periods. Upon approaching the end of this phase of cooldown, (RCS temperature of approximately 350°F), the RCS is depressurized to approximately 350 psia by venting the pressurizer through the safety-related pressurizer PORVs.

To ensure that the accumulators do not repressurize the RCS, the accumulator discharge valves are closed prior to the RCS pressure dropping below the accumulator discharge pressure. Each accumulator is provided with a Class 1E solenoid-actuated valve to ensure that the accumulator may be vented through the nitrogen supply header should the accumulator discharge isolation valve fail. A branch line inside the Containment with a parallel set of Class 1E valves allows venting the nitrogen header to Containment atmosphere.

Actuation of the SIS is precluded by use of the pressurizer low pressure and low steam line pressure signal blocks.

When the reactor coolant temperature and pressure are reduced to approximately 350° and 350 psig, respectively, the second phase of cooldown starts with the RHRS being placed in operation. Since loss of the nonsafety-grade instrument air system results in a loss of the air supply to the flow control valves that are normally used to limit the initial RHRS cooldown rate, the operator may choose to use only one of the Residual Heat Removal (RHR) subsystems as a means to control cooldown rate. Should a single failure occur, such as that of an RHRS component, precluding operation of one of the RHR subsystems, the operator could elect to use a fully operational RHR train. Cooldown would continue using the fully operational RHR train(s), until the failed equipment or component could be made available. A failure mode and effects analysis for cold shutdown operations is provided in Table 5.4.A-2.

Cooldown of the RCS is continued using available RHR trains and following cooldown rate limits. The time required to reach the cold shutdown conditions (see definition in Technical Specifications) depends upon the number of RHR trains available, and the Component Cooling Water (CCW) and Essential Cooling Water (ECW) temperatures.

TABLE 5.4.A-1

COMPLIANCE COMPARISON WITH BRANCH TECHNICAL POSITION RSB 5-1

Design Requirements of BTP RSB 5-1	Process and [System or Component]	Possible Solution for Full Compliance	Recommended Implementation for Class 2 plants*	Degree of STPEGS Compliance**
I. Functional requirements for taking to cold shutdown a. Capability using only safety grade systems b. Capability with either only onsite or only offsite power and with single failure (limited action outside control room to meet single failure) c. Reasonable time for cooldown assuming most limiting single failure and only offsite or only onsite power	Long-term cooling [RHR drop line]	Provide double drop line (or valves in parallel) to prevent valve failure from stopping RHR cooling function. (Note: This requirement, in conjunction with meeting effects of single failure for long-term cooling and isolation requirements, in valves increased number of independent power supplies and possibly more than four valves)	Compliance will not be required if it can be shown that correction for single failure by manual actions inside or outside of containment or return to hot standby until manual actions (or repairs) are found to be acceptable for the individual plant	Complies – Three RHR lines, each containing two normally closed Class 1E valves in series, are provided. Loss of a power supply or a valve failure will not stop RHR cooling function (Section 5.4.7.2.6)
	Heat removal and RCS circulation during cooldown to cold shutdown. (Note: Need SG cooling to maintain RCS circulation even after RHRS in operation when under natural circulation (steam dump valves)	Provide safety-grade dump valves, operators, and power supply, etc. so that manual action should not be required after SSE except to meet single failure	Compliance required	Complies – Steam generator PORVs are Class 1E (Section 10.3)

* The implementation for Class 2 plants does not result in a major impact while providing additional capability to go to cold shutdown. The major impact results from the requirement for safety-related steam dump valves.

** STP falls with the category of Class 2 plant as defined by Section H, "Implementation", of Branch Technical Position RSB 5-1, Revision 2.

TABLE 5.4.A-1 (Continued)

COMPLIANCE COMPARISON WITH BRANCH TECHNICAL POSITION RSB 5-1

Design Requirements of BTP RSB 5-1	Process and [System or Component]	Possible Solution for Full Compliance	Recommended Implementation for Class 2 plants*	Degree of STPEGS Compliance**
	Depressurization [pressurizer auxiliary spray or power-operated relief valves]	Provide upgrading and additional valves to ensure operation of auxiliary pressurizer spray using only safety grade subsystem meeting single failure. Possible alternative may involve using pressurizer PORVs which have been upgraded. Meet SSE and single failure without manual operation inside containment	Compliance will not be required if (a) dependence on manual actions inside containment after SSE or single failure or (b) remaining at hot standby until manual actions or repairs are complete and are found to be acceptable for the individual plant	Complies – Normal depressurization is via pressurizer auxiliary spray or pressurizer PORVs as safety-related backup (Sections 5.4.13 and 9.3.4)
	Boration for cold shutdown [CVCS and boron sampling]	Provide procedure and upgrading where necessary such that boration to cold shutdown concentration meets the requirements of 1. Solution could range from (1) upgrading and adding valves to have both letdown and charging paths safety grade and meet single failure to (2) use of backup procedures involving less cost. For example, boration without letdown may be acceptable and eliminate need for upgrading letdown	Same as above	Complies – Boration is accomplished by use of the boric acid tanks, the centrifugal charging pumps and the normal charging and/or reactor coolant pump seal injection lines Safety-related letdown capability is provided by the reactor vessel head vent system. The safety-related excore detectors will alert the operator of any criticality potential

* The implementation for Class 2 plants does not result in a major impact while providing additional capability to go to cold shutdown. The major impact results from the requirement for safety-related steam dump valves.

** STP falls with the category of Class 2 plant as defined by Section H, "Implementation", of Branch Technical Position RSB 5-1, Revision 2.

TABLE 5.4.A-1 (Continued)

COMPLIANCE COMPARISON WITH BRANCH TECHNICAL POSITION RSB 5-1

Design Requirements of BTP RSB 5-1	Process and [System or Component]	Possible Solution for Full Compliance	Recommended Implementation for Class 2 plants*	Degree of STPEGS Compliance**
		path. Use of ECCS for injection of borated water may also be acceptable. Need surveillance of boron concentration (boronometer and/or sampling). Limited operator action inside or outside of containment if justified..		Charging to and letdown from the RCS are known quantities. No boron sampling is required (Section 9.3.4, 5.4.7, 7.5)
II. Residual Heat Removal System isolation	Residual Heat Removal System	Comply with one of allowable arrangements given	Compliance required. (Plants normally meet the requirement under existing SRP, Section 5.4.7)	Complies – (Section 5.4.7.2)
III. Residual Heat Removal System pressure relief	Residual Heat Removal System	Determine piping, etc, needed to meet requirement and provide in design	Compliance will not be required if it is shown that adequate alternate methods of disposing of discharge are available	Complies – Residual heat removal relief valve discharge is piped to the PRT (Section 5.4.7.2.4)
V. Test requirement Meet RG 1.68 for PWRs, test plus analysis for cooldown under natural circulation to confirm adequate mixing and cooldown within limits specified in Emergency Operating Procedures		Run tests and confirming analysis to meet requirement	Compliance required	Meets the intent of RG 1.68. Test data and analysis for a plant similar in design to STPEGS will verify adequate mixing and cooldown under natural circulation conditions (Section 14.2)

* The implementation for Class 2 plants does not result in a major impact while providing additional capability to go to cold shutdown. The major impact results from the requirement for safety-related steam dump valves.

** STP falls with the category of Class 2 plant as defined by Section H, "Implementation", of Branch Technical Position RSB 5-1, Revision 2.

TABLE 5.4.A-1 (Continued)

COMPLIANCE COMPARISON WITH BRANCH TECHNICAL POSITION RSB 5-1

VI.	Operational procedure Meet RG 1.33. For PWRs, including specific procedures and information for cooldown under natural circulation		Develop procedures and information from tests and analysis	Compliance required	Generic Procedures as developed by the Westinghouse Owners Group will be used as the basis for plant specific procedures
VII.	Auxiliary Feedwater Supply Seismic Category I supply for auxiliary feedwater for at least 4 hours at hot standby plus cooldown to residual heat removal cut-in temperature based on longest time for only onsite or only offsite power and assumed single failure	Emergency feedwater supply	From tests and analysis obtain conservative estimate of auxiliary feedwater supply to meet requirements and provide seismic Category I supply	Compliance will not be required if it is shown that an adequate alternate seismic Category I source is available	The AFST Technical Specification required volume of 485,000 gals is adequate to support 4 hours at hot standby and a cooldown followed by depressurization to RHR cut-in conditions with a margin for contingencies. The most limiting failure regarding cooldown capability is a main feedwater line break with a failure of an AFW flow controller. The AFST meets seismic Category I requirements (Section 10.4.9)

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* The implementation for Class 2 plants does not result in a major impact while providing additional capability to go to cold shutdown. The major impact results from the requirement for safety-related steam dump valves.

** STP falls with the category of Class 2 plant as defined by Section H, "Implementation", of Branch Technical Position RSB 5-1, Revision 2.

TABLE 5.4.A-2

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Chemical and Volume Control System						
Centrifugal charging pump 1A (pump 1B analogous)	Operate to provide high pressure boration and makeup capability	1-5	Fails to operate indicator lights at MCB	Discharge header pressure, charging flow and seal injection flow indications and capable of providing adequate boration and makeup flow	Failure of the single pump as no effect on system operation The redundant centrifugal charging pump is	For CVCS P&ID see Figures 9.3.4-1, 9.3.4-3, and 9.3.4-5
Boric acid transfer pump 1A (pump 1B analogous)	Operate to provide boration flow from boric acid tank to suction of centrifugal charging pumps	1-5	Fails to operate	Pump flow and boric acid tank level indications and indicator lights at MCB	Failure of the single pump has no effect on system operation. The redundant boric acid transfer pump is capable of providing adequate boration flow	Gravity drain to centrifugal charging pumps also provided
FCV-0110B (FCV-0111B analogous) (normally closed fail closed)	Close to isolate boration path boundary	1-5	Fails to close	Makeup flow indication, boration flow indication, volume control tank level indication at MCB	None – RWST, via MOV-0112c or MOV-0113B can provide borated water to charging pump suction	Open valve may degrade preferred borated water source from boric acid tanks

* Plant Modes

1. Power Operation
2. Startup
3. Hot standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Check valve XCV0217	Open to allow boric acid flow to charging pump suction	1-5	Fails to open	Boric acid flow indication, charging and seal injection flow indications at MCB	None – RWST, via MOV-0112c or MOV-0113B can provide borated water to the charging pump suction	Manual action to open FCV-011OA and CV-0221 will restore boric acid flow to charging pumps
Check valve XCV0338 (valve XCV0349 analogous)	Open to allow boric acid flow to charging pump	1-5	Fails closed	Boric acid flow indication, charging flow indication at MCB	None – Redundant boric acid pump available to deliver boric acid to the charging pumps	
	Close to isolate pump in standby	1-5	Fails open	Boric acid flow indication, charging flow indication at MCB	None – Redundant boric acid pump available to provide boric acid flow to the charging pump suction header.	Boric acid flow would be back to boric acid tanks via standby pump util standby pump is started
Motor-operated gate valve MOV-0218 (normally closed)	Open to provide boric acid flow	1-5	Fails to open on demand	Valve position indication and boric acid transfer pump flow at MCB	None-RWST, via MOV-112C or MOV 113B can provide borated water to charging pump suction	Boric acid flow is available via valves FCV-011A and FCV-011OB (or FCV-0111B) or with manual action via FCV-011OA and CV-O221 or FCV-011OB. Gravity flow from boric acid tanks is available via manual valves CV-0226 and CV-0333

* Plant Modes
1. Power Operation
2. Startup
3. Hot standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Solenoid-operated globe valve HCV-0206 (normally closed)	Open to control boration and makeup flow via centrifugal charging pump 1B into safety-related charging line	1-5	Fails to open (or modulate) on demand	Valve position indication and charging flow indication at MCB	None – Control of boration and makeup via RCP seal injection lines is available via centrifugal charging pump 1A and valve MOV-8348	
Air-operated valve FCV-0205 (normally open, fail open)	Remain open to permit charging flow	1-5	Fails to open	Charging flow indication at MCB	None – Alternate safety-related charging path via charging pump 1A and MOV-8348 available	
Motor-operated gate valve MOV-8377B (normally open)	Close to isolate charging pump 1B from discharge header to establish safety-related charging path	1-5	Fails to close on demand	Valve position indication at MCB	None – Alternate path via centrifugal charging pump 1A and RCP seal injection available	
Motor-operated gate valve MOV-8377A	Close to isolate charging pump 1A from discharge header to establish safety-related charging path via RCP seal injection	1-5	Fails to close on demand	Valve position indication and seal injection flow at MCB	None – Alternate path via centrifugal charging pump 1B and HCV-0206 available	

* Plant Modes

1. Power Operation	4. Hot Shutdown
2. Startup	5. Cold Shutdown
3. Hot Standby	6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Motor-operated valve MOV-8348 (normally closed)	Open to provide boration and makeup flow via centrifugal charging pump 1A into RCP seal injection lines	1-5	Fails to open on demand	Valve position indication and RCP seal injection flow indication at MCB	None – Boration and makeup via centrifugal charging pump 1B and HCV-0206 is available	
Motor-operated gate valve MOV-0025 (normally open)	Open to provide charging flow	1-5	Fails Closed	Valve position indication and charging flow indication MCB	None – Boration and makeup can be accomplished via RCP seal injection lines	
Air-operated path valve LV-3119 (normally closed, fail closed)	Close to establish safety-related charging path		1.5	Fails open indication, RCS pressure indication at MCB	Valve position Can be isolated and charging provided via seal injection. RCS pressure can be maintained by pressurizer heaters	None – Normal charging
Motor-operated gate valve MOV-0003 (normally open)	Remain open to allow charging flow	1-5	Fails closed	Valve position indication and charging flow indication at MCB	None – An alternate charging path via valve MOV-0006 available	
Motor-operated gate valve MOV-0006 (normally closed)	Open to provide alternate charging	1-5	Fails to open on demand	Valve position indication and charging flow indication at MCB	None – Normal charging path via valve MOV-0003 available	

* Plant Modes

1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Motor-operated valve MOV-0033A (valves MOV-0033B, MOV-0033C, MOV-0033D (analogous) (normally open)	Remain open to allow charging flow	1-5	Fails closed	Valve position indication and seal injection flow indication at MCB	None – redundant charging path available	
Motor-operated gate valve MOV-0112B (valve MOV-0113A analogous) (normally open)	Close to isolate VCT from suction of centrifugal charging pumps	1-5	Fails to close on demand	Valve position indication at MCB	None – Two valves in series provide isolation	
Motor-operated gate valve MOV-0112C (valve MOV-0113B analogous) (normally closed)	Open to provide borated water to centrifugal charging pump suction from RWST	1-5	Fails to open on demand	Valve position indication at MCB	None – Two valves in parallel ensure flow path from the RWST	Boric acid flow from boric acid tanks via valve XCV0218 is preferred borated water source
Air-operated valve FV8400A (valve FV8400B analogous) (normally closed)	Close to isolate reactor coolant purity control system and maintain integrity of preferred boration source	1-5	Fails to close on demand	Valve position indication at MCB	None – Valve FV8400B closes to isolate reactor coolant purity control system	

* Plant Modes

1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Check valve XCV0235A (valve XCV0235B analogous)	Open to allow charging flow to RCS	1-5	Fails to open	Charging flow, seal injection flow indication at MCB	None – Charging available via redundant charging pump	
Check valve XCV0670 (valve XCV0671 analogous)	Close to establish safety-related charging path via charging pump 1B	1-5	Fails to close	Charging flow, seal injection flow indication at MCB	None – Controlled charging via pump 1A available	
Check valve CV0034A (valves CV0034B C, D, and CV0036 A, B, C, D and CV0037 A, B, C, D analogous)	Open to allow seal injection flow	1-5	Fails to open	Seal injection flow indication at MCB	None – Controlled charging via normal charging available	
	Close to isolate seal injection flow	1-5	Fails to open	None	Non – 3 check valves in series provide isolation	
Check valve XCV0026	Open to allow charging flow	1-5	Fails to open	Charging flow indication at MCB	None – Charging via RCP seal injection available	
Check valve XCV001 (valve XCV002 analogous)	Open to allow charging flow	1-5	Fails to open	Charging flow indication at MCB	None – Charging via RCP seal injection available	

* Plant Modes

1. Power Operation	4. Hot Shutdown
2. Startup	5. Cold Shutdown
3. Hot Standby	6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Motor-operated valve LCV-0465 (valve LCV-0468 analogous) (normally open)	Close to isolate normal letdown	1-5	Fails to close	Valve position indicating lights letdown, flow, temperature and pressure indication at MCB	None – Valve LCV-0468 provides isolation	
Motor-operated valve MOV- 0082 (valve MOV-0083 analogous) (normally open)	Close to isolate excess letdown	1-5	Fails to close	Valve position indication lights, excess letdown pressure and temperature indication at MCB	None – Valve MOV-0083 provides isolation	
Reactor Coolant System						
Solenoid actuated valves FV-3657A and FV-3658A (valves FV-3657B and FV-3658B analogous) (normally closed)	Close to isolate reactor vessel head vent line	1-5	Once open, valve fails open	Valve position indication, PRT temperature and level indications at MCB	None – Two valves in series provide isolation	Valves are normally closed, fail closed valves, therefore loss of power to one train of valves will not prevent isolation (see Figure 5.1-1)
	Open to provide RCS venting and letdown path	1-5	Fails to open on demand	Valve position indication, PRT temperature and level indications at MCB	None – Two parallel paths ensure the availability of the head vent system	

* Plant Modes

1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Solenoid-actuated valve HCV-601 (valve HCV-602 analogous) (normally closed)	Open to control letdown rate via reactor vessel head vent system	1-5	Fails to open (or modulate) on demand	Valve position indication, PRT temperature, pressure and level indications at MCB	None – Two parallel paths are provided, each having the capability of controlling the letdown rate	
Solenoid-operated globe valve PCV-0655A (valve PCV-0656A analogous) (normally closed)	Open to provide RCS depressurization capability	1-5	Fails to open on demand	Valve position indication, discharge line temperature, PRT temperature, pressure and level indications at MCB	None – Two valves in parallel ensure depressurization capability	(See Figure 5.1-3)
Solenoid-operated globe valve PCV-0655A (valve PCV-0656A analogous) (normally closed)	Close to provide isolation of RCS	1-5	Fails to close once open	Valve position indication, discharge line temperature, PRT temperature, pressure and level indications at MCB	None – Failed valve can be isolated by valve MOV-0001A (valve MOV-0001B)	
Motor-operated gate valve MOV-0001A (valve MOV-0001B analogous) (normally open)	Close to provide RCS isolation	1-5	Fails to close on demand	Valve position indication, discharge line temperature, PRT temperature, pressure and level indications at MCB	None – Valve PCV-0655A (valve PCV-0656A) will isolate RCS	

* Plant Modes
1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Safety Injection System						
Motor-operated gate valve MOV-0039A (valves MOV-0039B and MOV-0039C analogous) (normally open)	Close to isolate accumulator A (B and C) from RCS	1-5	Fails to close on demand	Vale position indication at MCB	None – Accumulator N_2 overpressure can be vented via valves PV-3930 (valves PV-3929 and PV-3928) and HCV-0900 or HV-0899	Valve is normally open with power locked out (Figure 6.3-4)
Solenoid-operated globe valve PV-3930 (valves PV-3928 and PV-3929 analogous) (normally closed)	Open to provide vent capability for accumulator A (B and C)	1-5	Fails to open on demand	Valve position indication and accumulator pressure indicate at MCB	None – Accumulator A can be isolated from RCS by MOV-0039A (valves MOV-0039B and MOV-0039C)	
Solenoid-operated globe valve HCV-0900, (valve HV-0899 analogous) (normally closed)	Open to provide vent capability for common accumulator vent line	1-5	Fails to open on demand	Valve position indication for HCV-0900, accumulator pressure indication at MCB	No effect – Two valves in parallel ensure vent capability	

* Plant Modes
1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Residual Heat Removal System						
RHR Pump 1A (pumps 1B and 1C analogous)	Operate and provide coolant flow for RHR operation	4-5	Fails to operate	Pump flow and discharge pressure indication and indicator lights at MCB	None – Two redundant RHR pumps available	(See Figure 5.4-6) Loss of pump may result in longer cooldown time
Motor-operated gate valves MOV-0060A and MOV-0061A (valves MOV-0060B and MOV-0061B, MOV-0060C and MOV-0061C analogous) (normally closed)	Open to allow RHR flow	4-5	Either valve fails to open on demand or once open, fails closed	Valve position indication, pump flow indication and alarm and discharge pressure indication at MCB	None – Redundant RHR trains provide cooling	
	Close to isolate the RHR system from the RCS	4-5	Fails to close	Valve position indication, pressure and temperature indications at MCB	None – Redundant valves in series provide isolation	

* Plant Modes
1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Residual Heat Removal System						
RHR Pump 1A (pumps 1B and 1C analogous)	Operate and provide coolant flow for RHR operation	4-5	Fails to operate	Pump flow and discharge pressure indication and indicator lights at MCB	None – Two redundant RHR pumps available	(See Figure 5.4-6) Loss of pump may result in longer cooldown time
Motor-operated gate valves MOV-0060A and MOV-0061A (valves MOV-0060B, MOV-0060C and MOV-0061C analogous) (normally closed)	Close to isolate the RHR system from the RCS	4-5	Fails to close	Valve position indication, pressure and temperature indications at MCB	None – Redundant valves in series provide isolation	

* Plant Modes

1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Air-operated butterfly valve HCV-864 (valves HCV-865 and HCV-866 analogous) (normally open, fail open)	Remain open to permit RHR flow	4-5	Fails closed	Valve position indication, RHR discharge temperature indication at MCB	None – Redundant RHR trains available	Solenoid valve FY3860A (FY3861A and FY3862A) is available to vent air to the valve and open the valve if the valve is functional Inability to access a RHR train may result in longer cooldown times
Air-operated butterfly valve FCV-851 (valves FCV-852 and FCV-853 analogous) (normally closed, fail closed)	Remain closed to direct RHR flow through RHR heat exchanger	4-5	Fails open	Valve position indication, RHR discharge flow and temperature indication at MCB	None – Redundant RHR trains available	Solenoid valve FY3857A (FY3858A and FY3859A) may be used to vent the air to the valve and close the valve, if the valve is functional
Motor-operated valve MOV-0031A (valves MOV-0031B and MOV-0031C analogous (normally open)	Remain open to allow RHR flow to RCS	4-5	Fails closed	Valve position indication RHR flow indication at MCB	None – Redundant RHR trains available	

* Plant Modes

1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Component	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Check valve XRH0032A (valves XRH0032B, XRH0032C, XRH0065A, XRH0065B, and XRH0065C analogous)	Open to allow RHR flow to the RCS	4-5	Fails closed	RHR flow indication at MCB	None – Redundant RHR trains available	
Motor-operated valve MOV-0067A (valves MOV-0067B and MOV-0067C analogous) (normally closed)	Remain closed to direct RHR flow to RCS	4-5	Fails open	Flow indication down-stream of RHR heat exchanger, RHR temperature indication at MCB. Non-1E indicator lights	Non – Redundant RHR trains available	
Motor-operated valve MOV-0066A (valve MOV-0066B analogous) (normally closed)	Remain closed to direct RHR flow to RCS	4-5	Fails open	RHR flow downstream of RHR heat exchanger, CVCS letdown flow indications at MCB	None – Redundant RHR trains available	
Support Systems						
Class 1E AC Power Train A (Trains B & C analogous)	Provide power to Train A AC components	1-6	Loss of power on bus	Bus undervoltage alarms ESF status monitoring for ESF Diesel Generator System and components, ESF monitoring for system and AC components	None – Components from trains B and C available to provide system safety capability	

* Plant Modes

1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

TABLE 5.4.A-2 (Continued)

COLD SHUTDOWN

FAILURE MODES AND EFFECTS ANALYSIS

Description of Components	Safety Function	Plant Operating Mode*	Failure Mode(s)	Method Of Failure Detection	Failure Effect On System Safety Function Capability	General Remarks
Channel I DC Power (Train A) (Channels II, III, & IV (Trains D, C & B) (analogous)	Provide DC power to Channel I components	1-6	Loss of DC power	ESF monitoring on UPS failure, DC trouble alarm ESF monitoring for pump (not running, not control power)	None – Components from channels II, III, & IV available to provide system safety capability	Pump status lights off
Instrument Air (nonsafety)	None	1-6	Instrument air lost	Header pressure indication and alarms	None – Loss of instrument air causes air-operated components to go to their safety position	

FMEAs for other systems required to achieve and maintain cold shutdown are presented as identified below:

Main Steam System – Tables 10.3-1 and 10.3-1A

Auxiliary Feedwater System – Tables 10.4-3 and 10.4-3A

Component Cooling Water System – Table 9.2.2-3

Essential Cooling Water System – Table 9.2.1-2

Reactor Makeup Water System – Table 9.2.7-2

Emergency Diesel Generators – Table 9.5.5-2

HVAC Systems – Tables 9.4.5-1 through 9.4.5-7

Qualified Display Processing System – Table 7.5-4

* Plant Modes

1. Power Operation
2. Startup
3. Hot Standby
4. Hot Shutdown
5. Cold Shutdown
6. Refueling

UFSAR

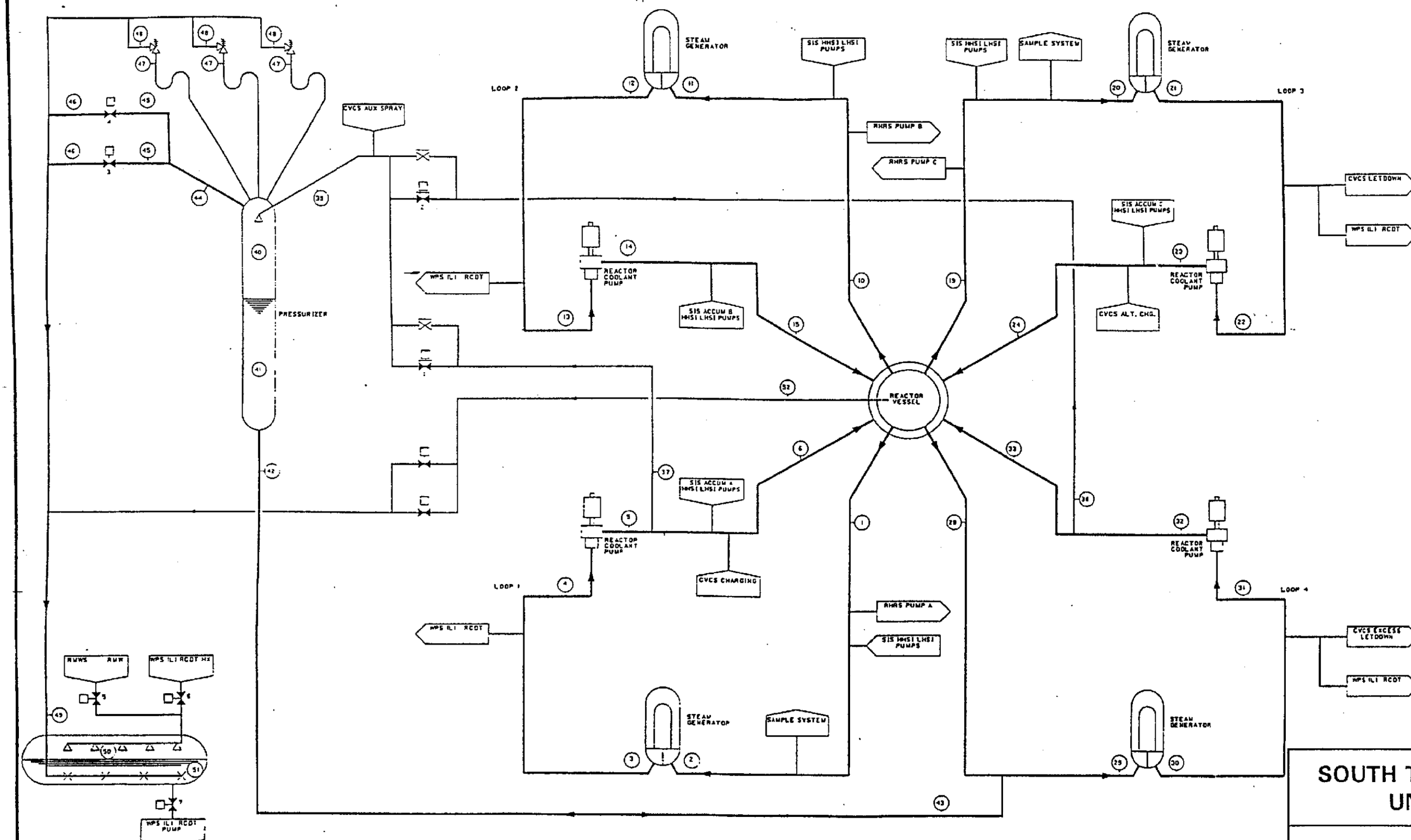
Section 5.1

UFSAR FIGURE 5.1-1
Reference Drawing(s) 5R149F05001#1
5R149F05001#2

UFSAR FIGURE 5.1-2
Reference Drawing(s) 4R149F05002#1
4R149F05002#2

UFSAR FIGURE 5.1-3
Reference Drawing(s) 5R149F05003#1
5R149F05003#2

UFSAR FIGURE 5.1-4
Reference Drawing(s) 5R149F05004#1
5R149F05004#2



NOTE:

THIS DIAGRAM IS A SIMPLIFICATION OF THE SYSTEM INTENDED TO FACILITATE THE UNDERSTANDING OF THE PROCESS. FOR DETAILS OF PIPING, VALVES, INSTRUMENTATION, ETC. REFER TO THE ENGINEERING FLOW DIAGRAM. REFER TO PROCESS FLOW DIAGRAM TABLES FOR CONDITIONS AT EACH NUMBERED POINT.

SOUTH TEXAS PROJECT
UNITS 1 & 2

REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM

Figure 5.1-5 Revision 0

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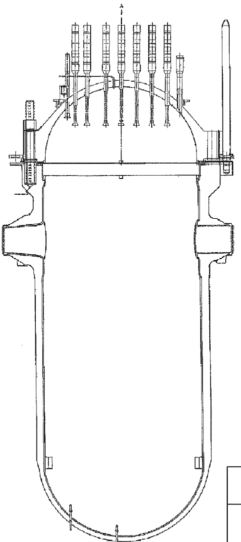
Section 5.3



SOUTH TEXAS PROJECT
UNIT 1

REACTOR VESSEL

FIGURE 5.3-1A

REVISION 15

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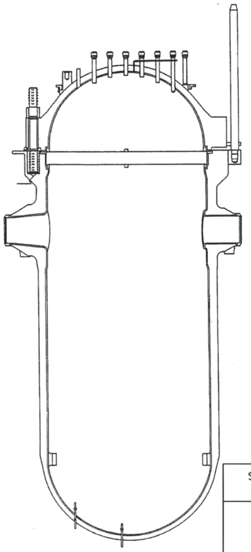

SOUTH TEXAS PROJECT UNIT 2	
REACTOR VESSEL	
FIGURE 5.3-1B	REVISION 15

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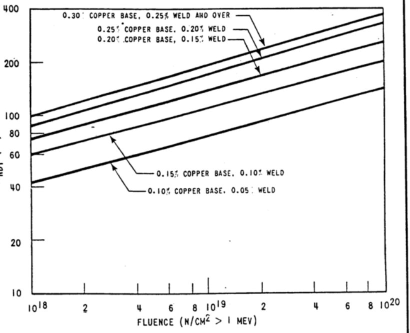
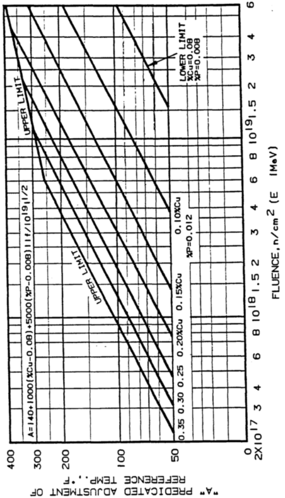


SOUTH TEXAS PROJECT
UNITS 1 & 2

WESTINGHOUSE CURVES - EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSEL STEELS EXPOSED TO 550°F TEMPERATURE

Figure 5.3-2 Revision 0



Note: For copper and phosphorus contents other than those plotted, use the expression for "A" given on the figure..

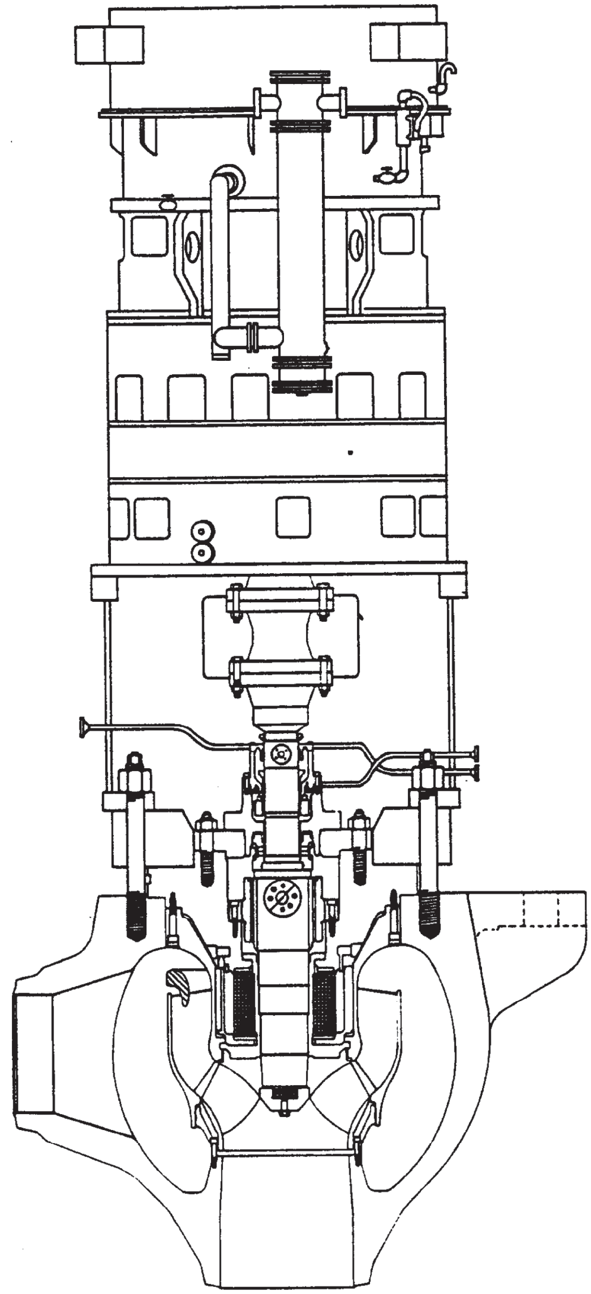
SOUTH TEXAS PROJECT
UNITS 1 & 2

PREDICTED ADJUSTMENT OF REFERENCE TEMPERATURE "A" AS A FUNCTION OF FLUENCE AND COPPER CONTENT

Figure 5.3-3 | **Revision 0**

UFSAR

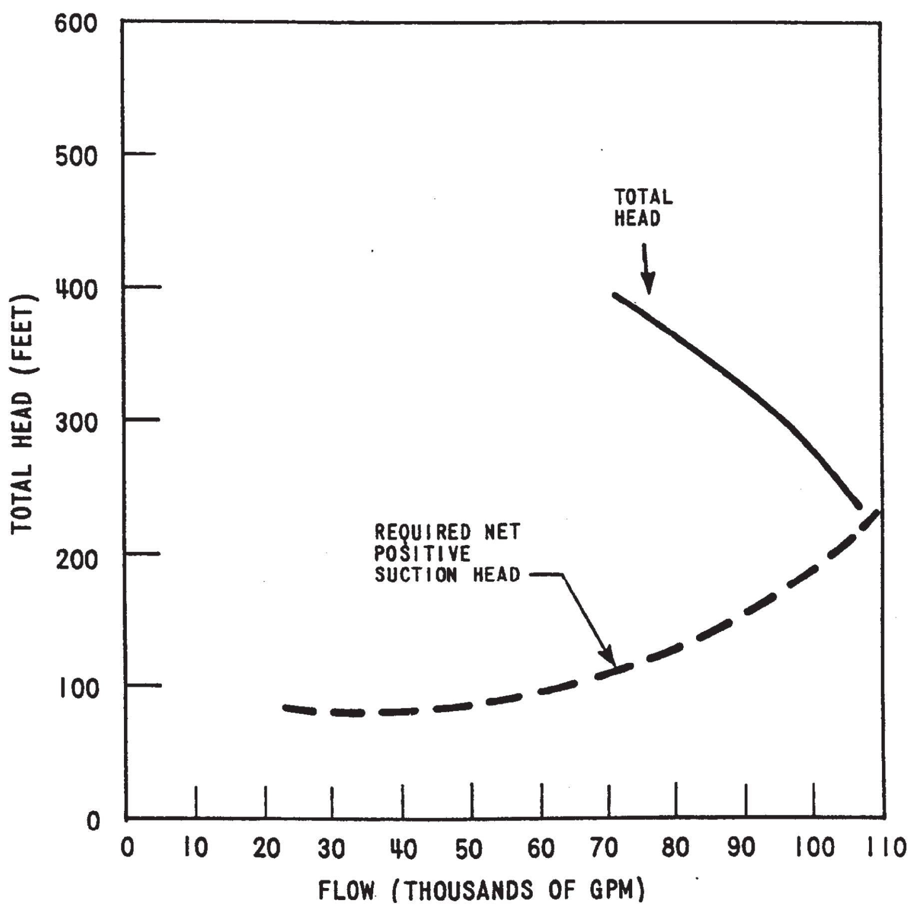
Section 5.4



SOUTH TEXAS PROJECT
UNITS 1 & 2

REACTOR COOLANT PUMP

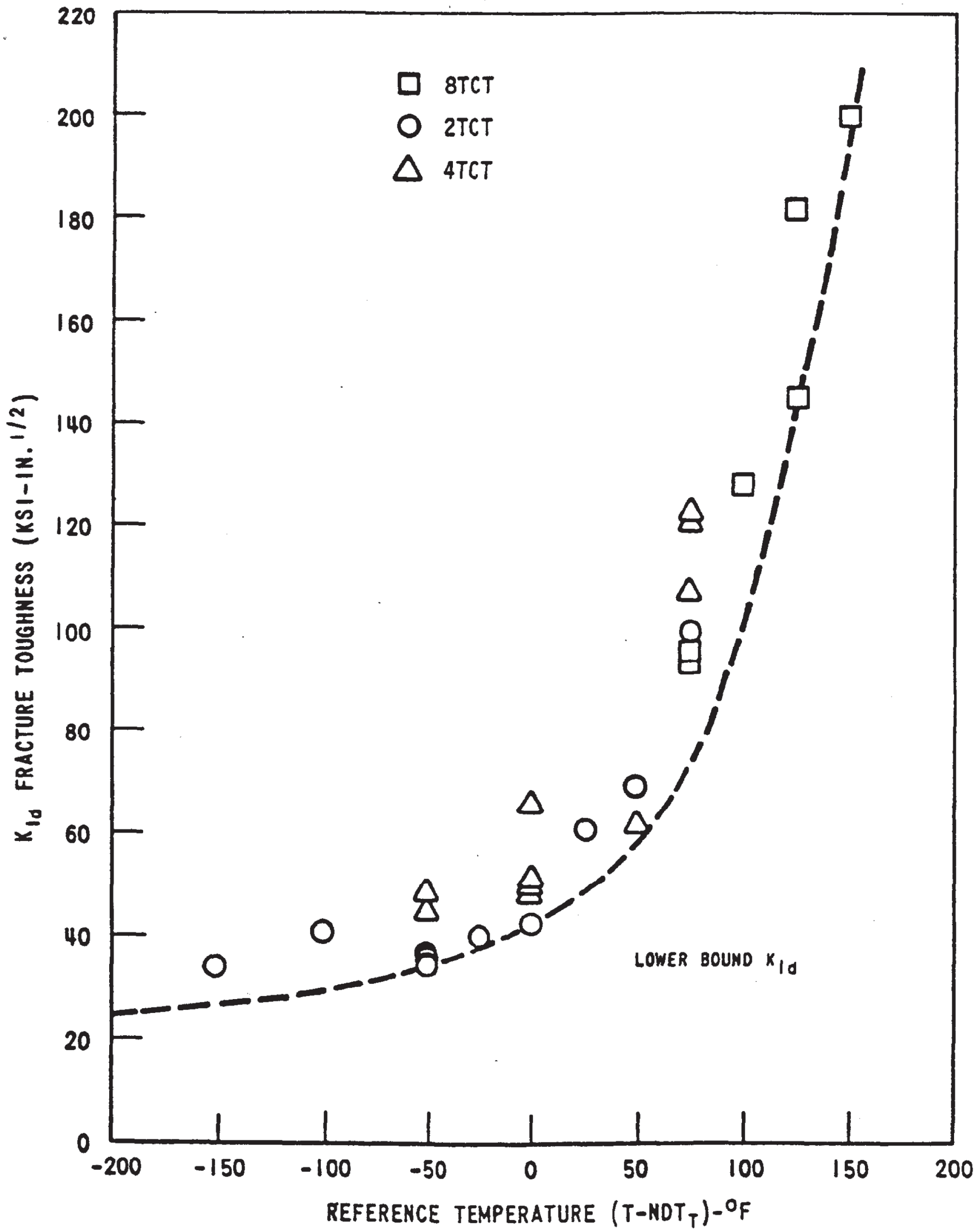
Figure 5.4-1 | Revision 0



SOUTH TEXAS PROJECT UNITS 1 & 2

REACTOR COOLANT PUMP ESTIMATED PERFORMANCE CHARACTERISTICS

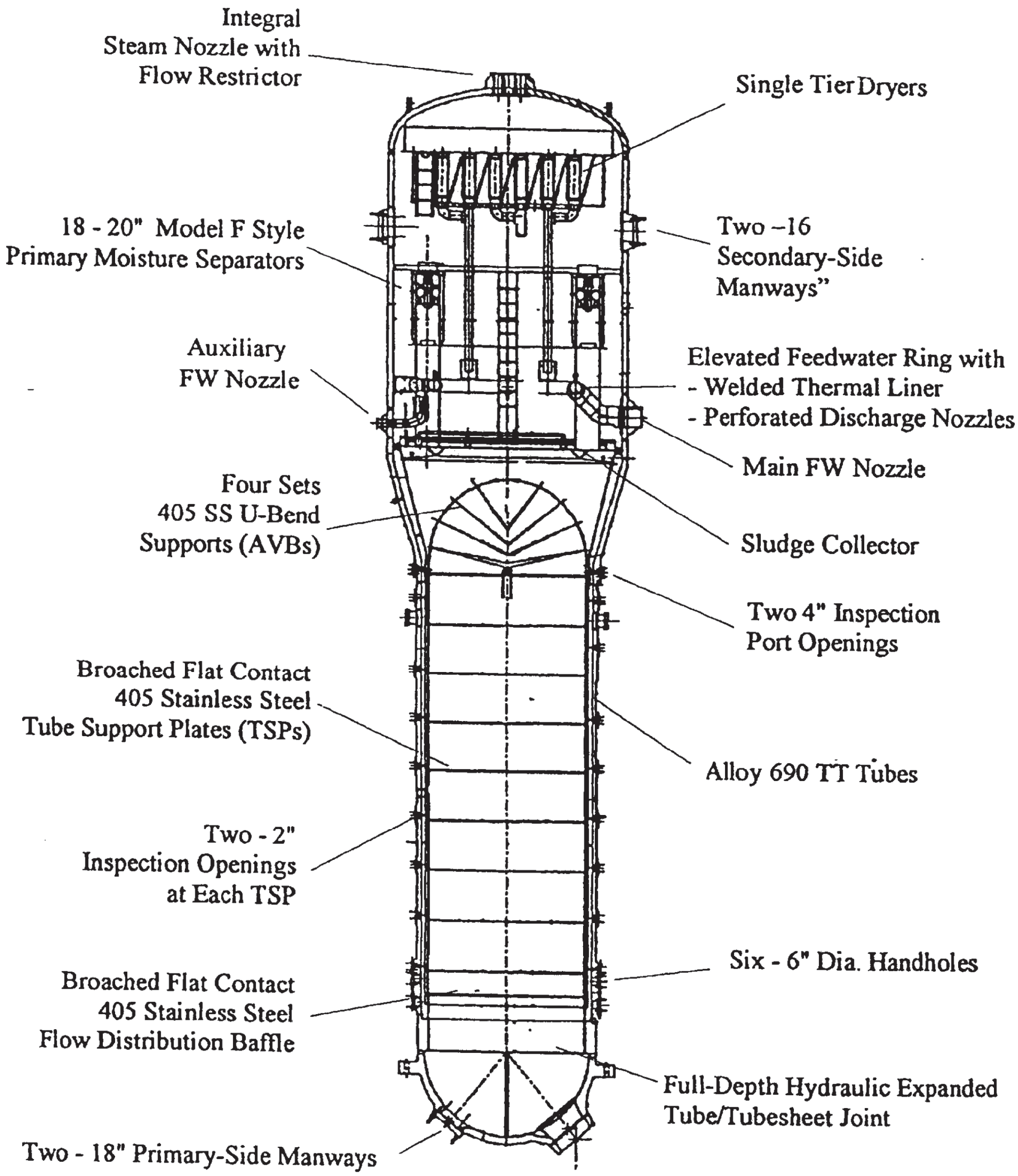
Figure 5.4-2 Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

K_{Id} LOWER BOUND FRACTURE TOUGHNESS A533V (REFERENCE WCAP 7623) GRADE B CLASS 1

Figure 5.4-3 Revision 0

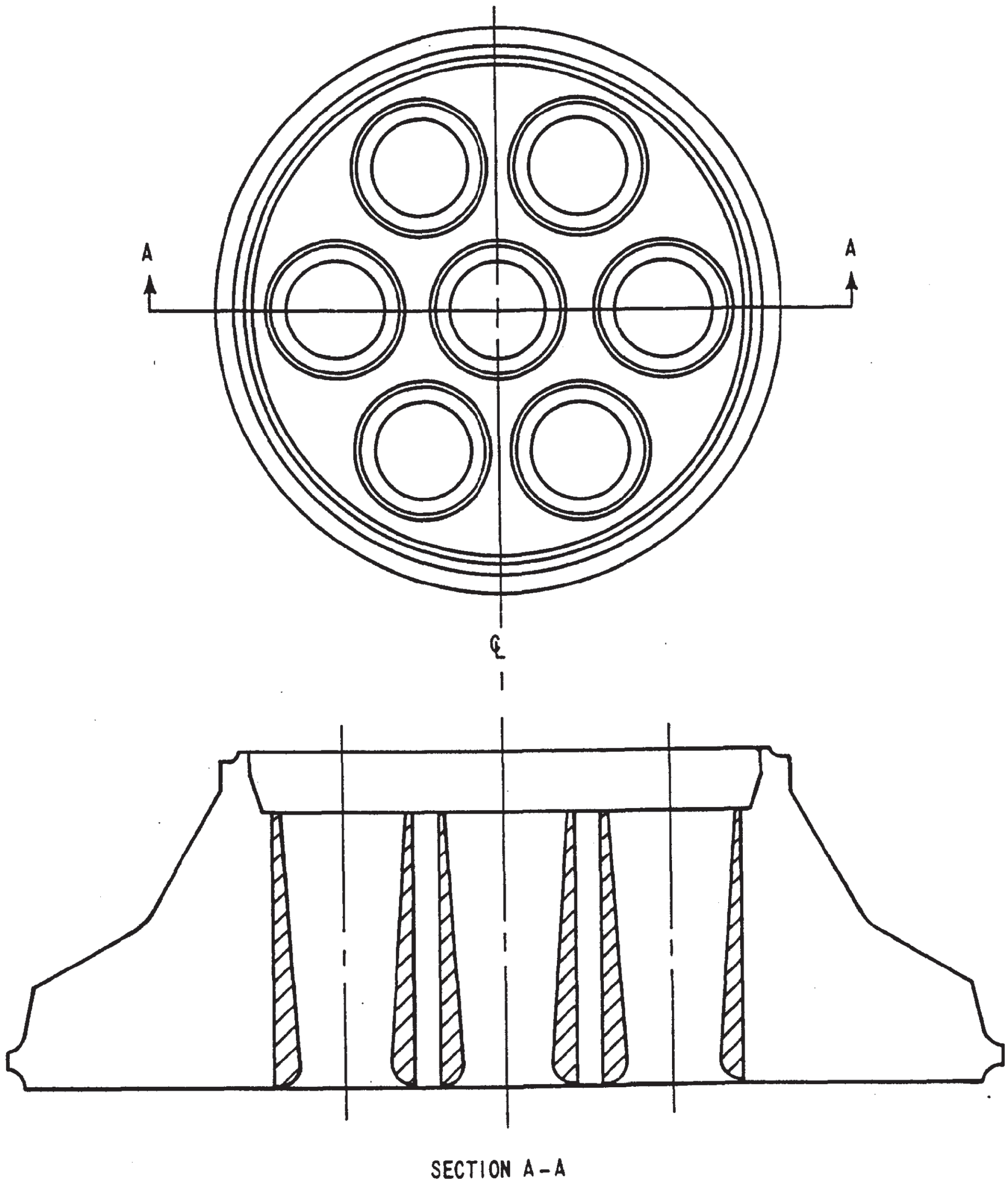


SOUTH TEXAS PROJECT
UNITS 1 & 2

STEAM GENERATOR

FIGURE 5.4-4 REVISION 11

L0999.CIT
L0999.111

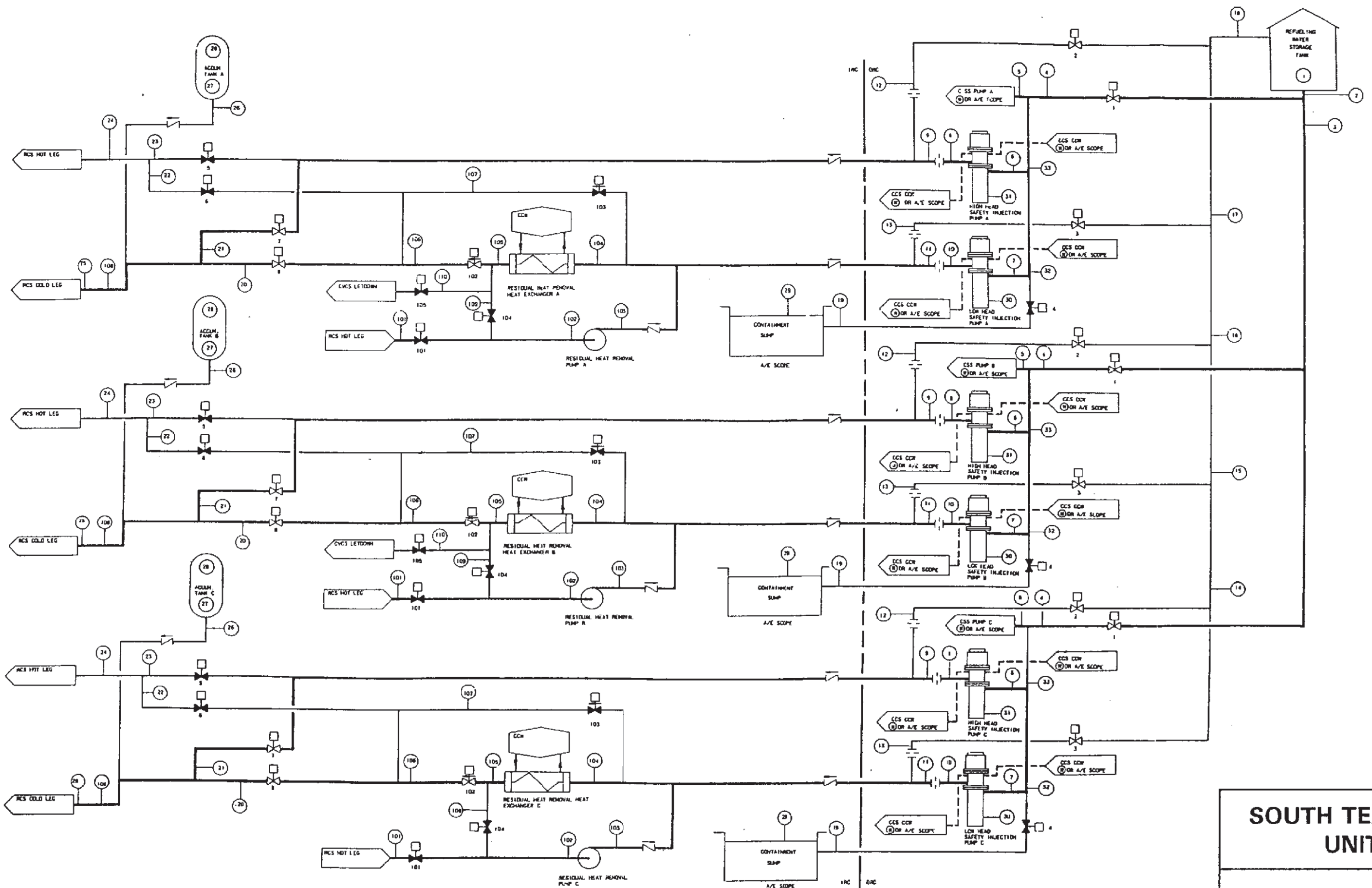


SOUTH TEXAS PROJECT
UNITS 1 & 2

STEAM LINE FLOW RESTRICTOR

Figure 5.4-5 Revision 0

UFSAR FIGURE 5.4-6
Reference Drawing(s) 5R169F20000#1
5R169F20000#2



NOTE

1. THIS DIAGRAM IS A SIMPLICATION OF THE SYSTEMS INTENDED TO FACILATE THE UNDERSTANDING OF THE PROCESS FOR DETAILS OF THE PIPING VALVES INSTRUMENTATION ETC. REFER TO THE ENGINEERING FLOW DIAGRAM. REFER TO PROCESS FLOW DIAGRAM TABLES FOR THE CONDITIONS AT EACH NUMBERED POINT.

2. THE CCS CCW FLOW TO THE SAFETY INJECTION PUMPS IS NOT APPLICABLE TO STP.

SOUTH TEXAS PROJECT UNITS 1 & 2	
RESIDUAL HEAT REMOVAL SYSTEM PROCESS FLOW DIAGRAM	
Figure 5.4-7	Revision 0

NOTES TO FIGURE 5.4-7

Modes of Operation

Mode A: Initiation of RHR Operation

Following reactor shutdown, the second phase of plant cooldown starts with the RHRS being placed in operation. Before starting the pumps, the inlet isolation valves are opened, the miniflow valves are opened, the heat exchanger flow control valves are set at minimum flow, and the outlet valves are verified open. The miniflow valves are manually closed when flow is established to the RCS cold legs. Should the pump flow drop below the low flow set point the pump will automatically be shut off and an alarm will sound in the control room. CN-2145

Startup of the RHRS includes a warm-up period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock on the RCS. The rate of heat removal from the reactor coolant is controlled manually by regulating the reactor coolant flow through the RHR heat exchangers. The total flow is regulated automatically by a control valve in each heat exchanger bypass line to maintain a constant total flow. The cooldown rate is limited to 100 F/hr based on equipment stress limits and a 120 F maximum component cooling water temperature.

Mode B: End Conditions of a Normal Cooldown

This situation characterizes most of the RHRS operation. As the reactor coolant temperature decreases, the flow through the RHR heat exchanger is increased until all of the flow is directed through the heat exchanger to obtain maximum cooling.

Note: The Emergency Core Cooling System components shown on the process flow diagram are not used for normal residual heat removal operations. The process conditions associated with the Emergency Core Cooling System operation (location numbers 1 through 37) are presented in the notes to Figure 6.3-5.

NOTES TO FIGURE 5.4-7 (Continued)

VALVE ALIGNMENT CHART

Valve No.	Operational Mode		
	A	A'	B
101	O	O	O
102	P	P	O
103	P	P	C
104	C*	C*	C
105	O	C	C
7	C	C	C
8	O	O	O

O - Open
C - Closed
P - Partially Open

* During initiation, valve 104 is open until flow is established in the RCS.

NOTES TO FIGURE 5.4-7 (Continued)

MODE A INITIATION OF RHR OPERATION (SUB-SYSTEM WITH RC PUMP & CVCS LETDOWN)[1]

Location[2]	Fluid	Pressure psig	Temperature °F	Flow gal/min[3]	Flow 10^6 lb/hr
101	RC	400	345	3400	1.544
102	RC	409	345	3400	1.544
103	RC	488	345	3400	1.544
104	RC	464	345	1144	0.51
105	RC	461	144.4	1144	0.51
106	RC	461	144.4	744	0.33
107	RC	440	345	2256	1.034
108	RC	417	296	3000	1.364
109	RC	461	144.4	400	0.18
110	RC	<461	144.4	400	0.18

1. During mode A operation one of the three RHR subsystems may be operating in the loop with the active reactor coolant pump. In this situation the RHR pump must overcome the reactor coolant pump head in the cold leg piping. Also, one of the subsystems (train A or B) may be supplying purification flow to the CVCS via the low pressure letdown path. For convenience this operation is included in this mode.

2. All location numbers not listed identify lines containing fluid at static conditions.

3. At reference conditions 350°F and 400 psig.

NOTES TO FIGURE 5.4-7 (Continued)

MODE B END CONDITIONS OF A NORMAL COOLDOWN

Location[1]	Fluid	Pressure psig	Temperature °F	Flow gal/min[2]	Flow 10^6 lb/hr
101	RC	3.8[3]	150	3400	1.69
102	RC	14	150	3400	1.69
103	RC	102.2	150	3400	1.69
104	RC	78	150	3400	1.69
105	RC	52.3	123.5	3400	1.69
106	RC	<52.3	123.5	3000	1.49
107	RC	27.2	150	0	0
108	RC	3.8	123.5	3000	1.49
109	RC	52.3	123.5	400	0.2
110	RC	52.3	123.5	400	0.2

1. All location numbers not listed identify lines containing fluid at static conditions.

2. At reference conditions 350°F and 400 psig.

3. It is assumed here that the RCS is at a vacuum condition equal to the vapor pressure of water at 150°F.

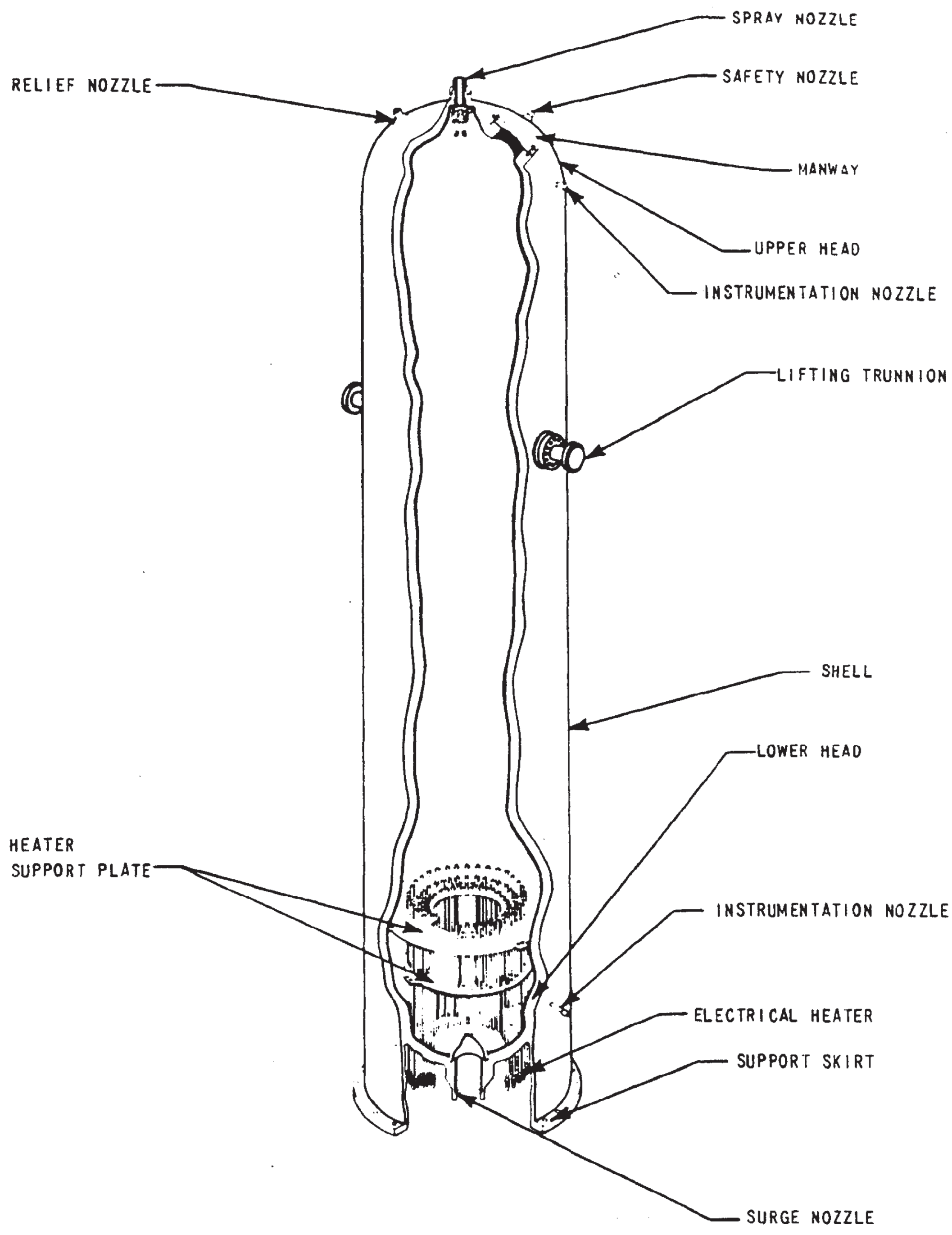
NOTES TO FIGURE 5.4-7 (Continued)

MODE A INITIATION OF RHR OPERATION
(SUB-SYSTEM NOT CONNECTED TO LOOP WITH RC PUMP RUNNING)

Location[1]	Fluid	Pressure psig	Temperature °F	Flow gal/min[2]	Flow 10^6 lb/hr
101	RC	393	345	3400	1.544
102	RC	403	345	3400	1.544
103	RC	488	345	3400	1.544
104	RC	469	345	1144	0.51
105	RC	466	144.4	1144	0.51
106	RC	466	144.4	744	0.33
107	RC	420	345	2256	1.034
108	RC	396	268.5	3400	1.544
109	RC	396	268.5	0	0
110	RC	396	268.5	0	0

1. All location numbers not listed identify lines containing fluid at static conditions.

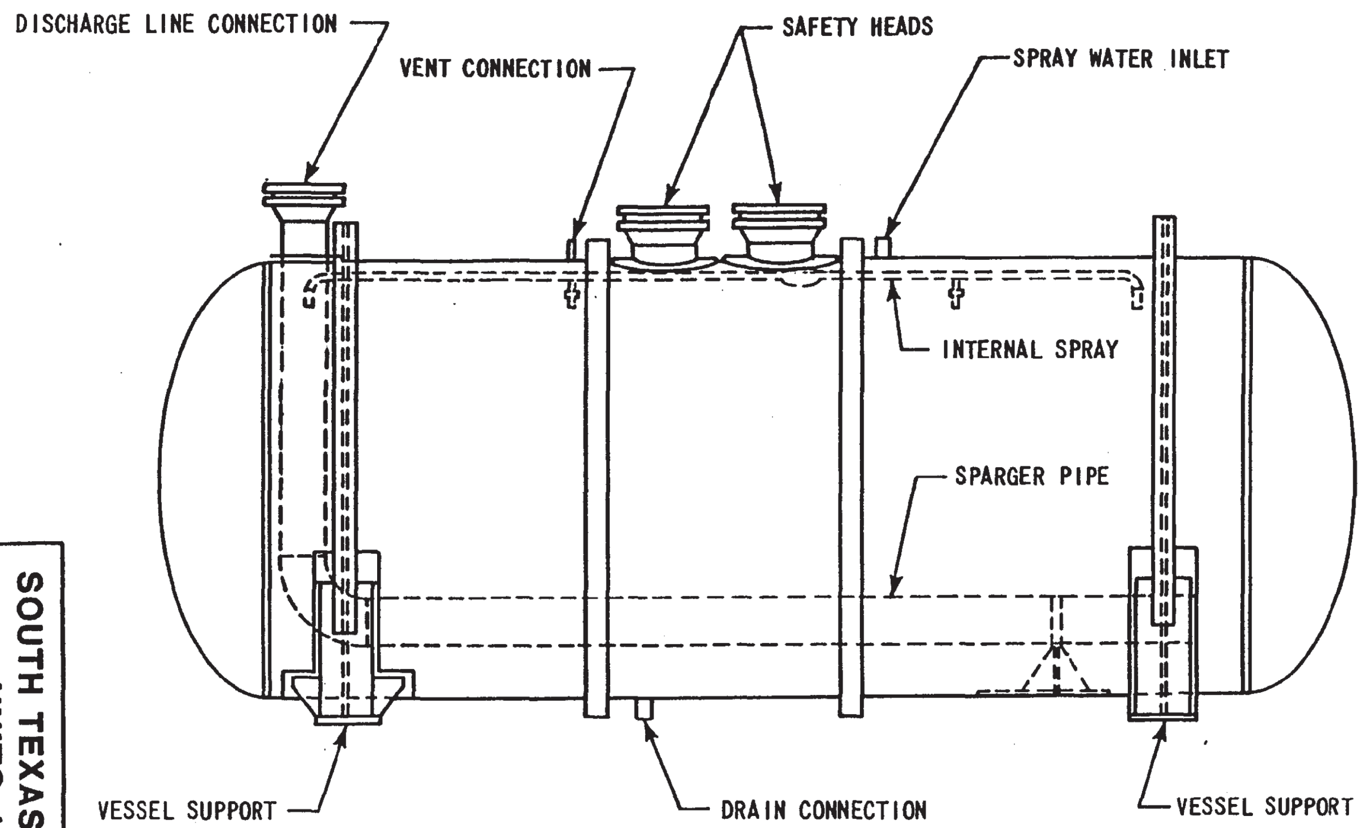
2. At reference conditions 350°F and 400 psig.



SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURIZER

Figure 5.4-10 **Revision 0**

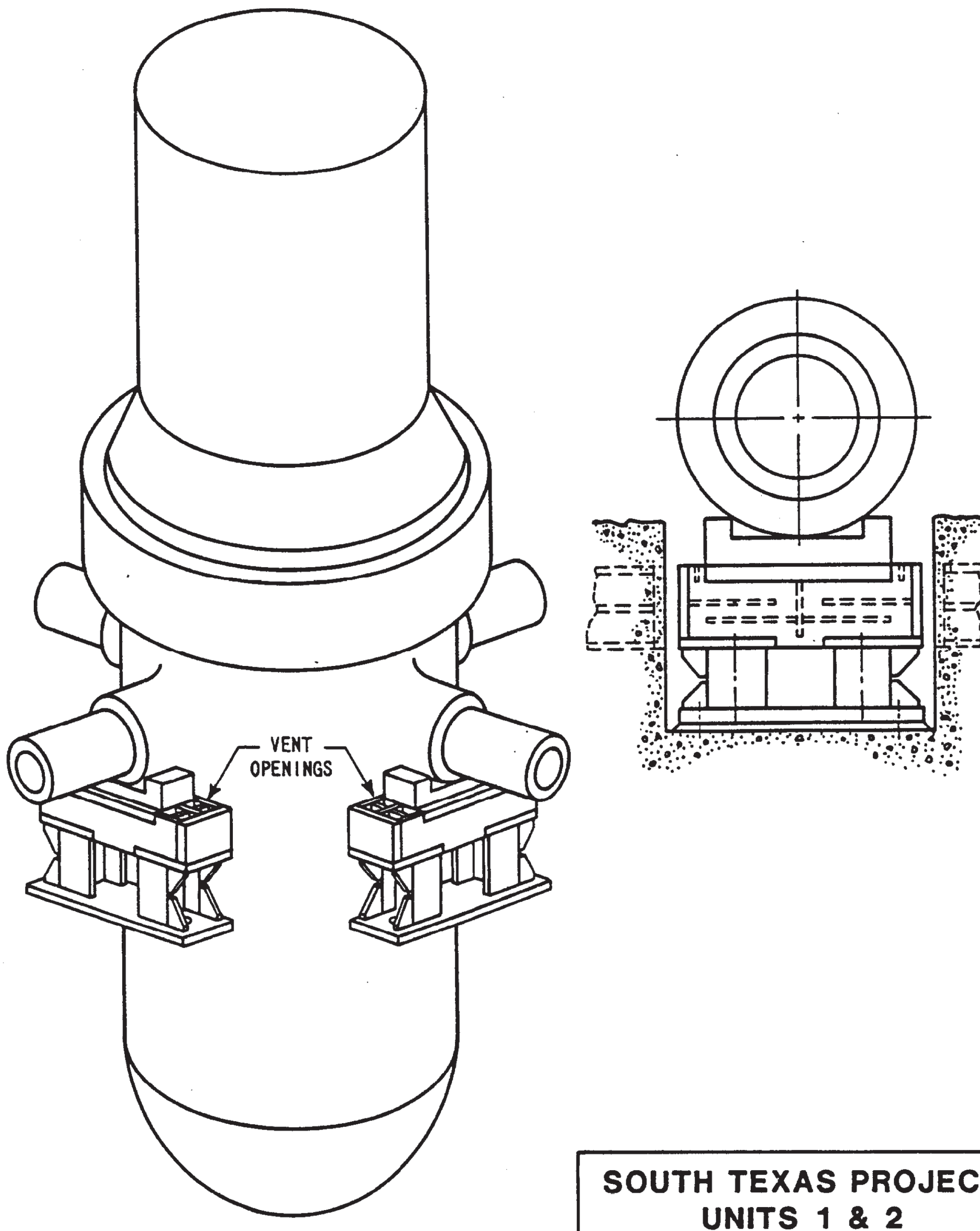


SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURIZER RELIEF TANK

Figure 5.4-11

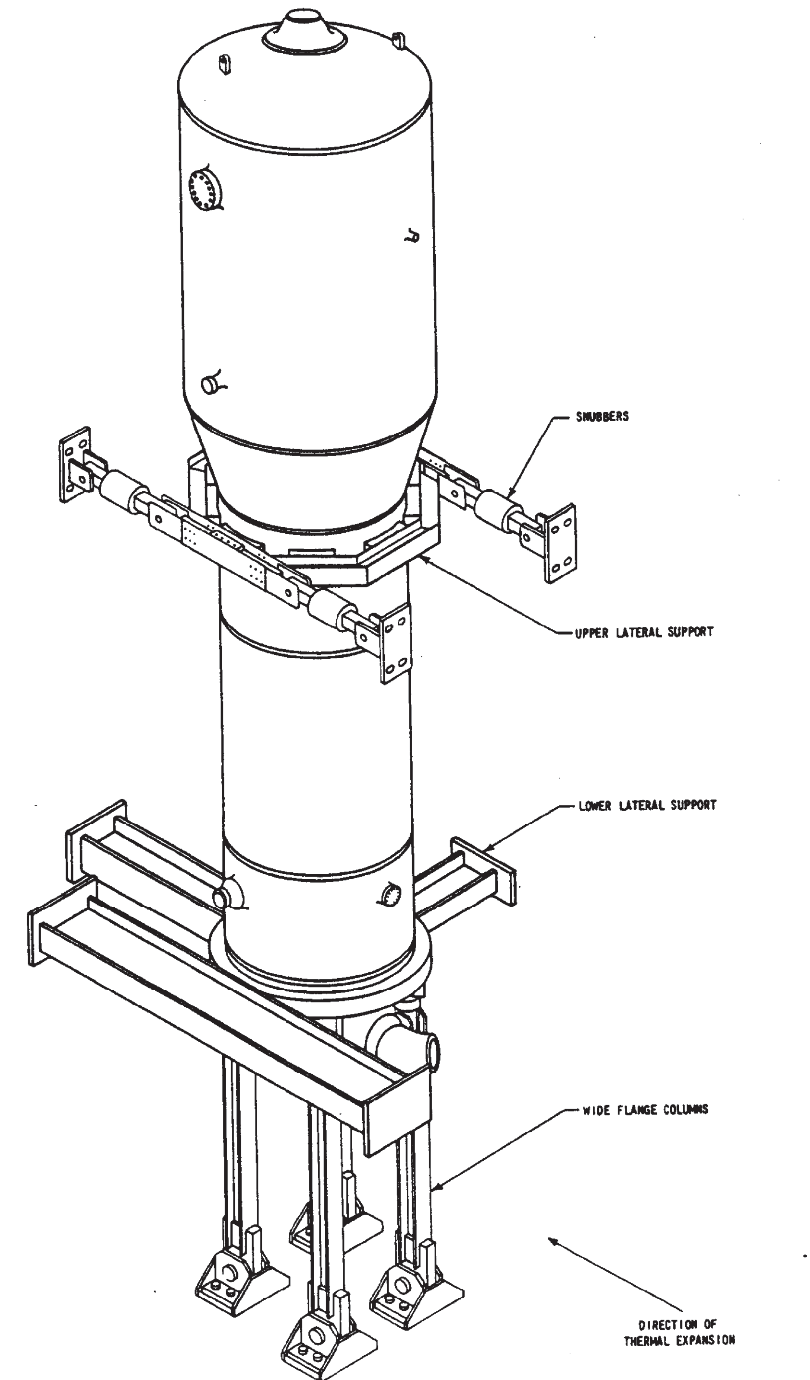
Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

REACTOR VESSEL SUPPORTS

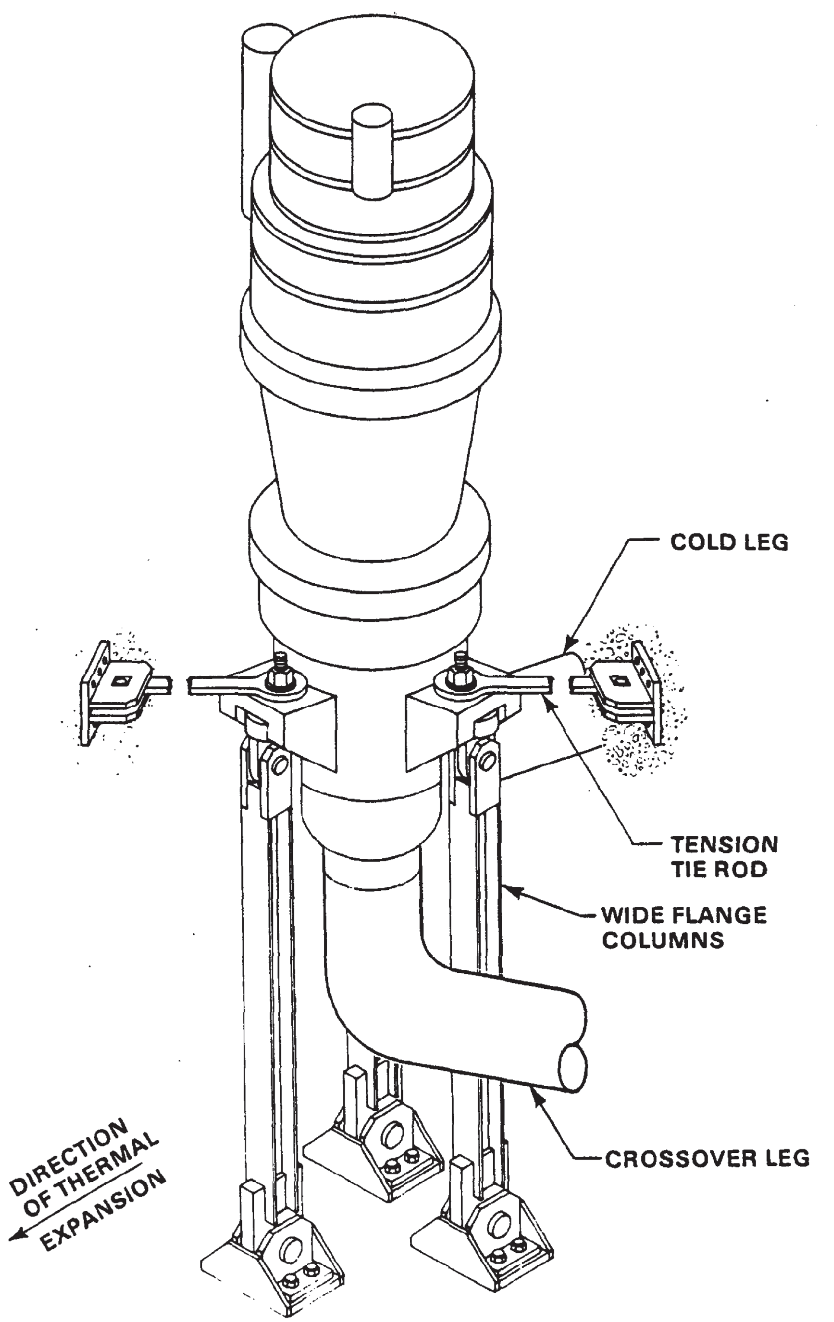
Figure 5.4-12 | Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

STEAM GENERATOR SUPPORTS

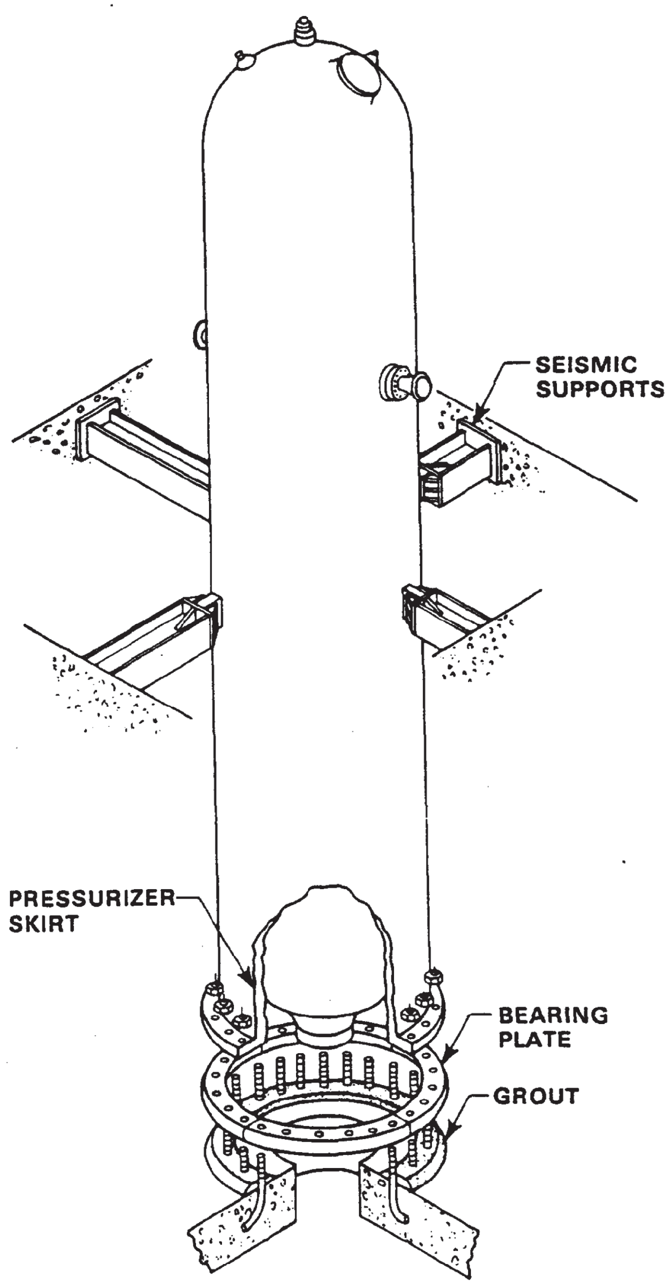
Figure 5.4-13 Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

REACTOR COOLANT PUMP SUPPORT

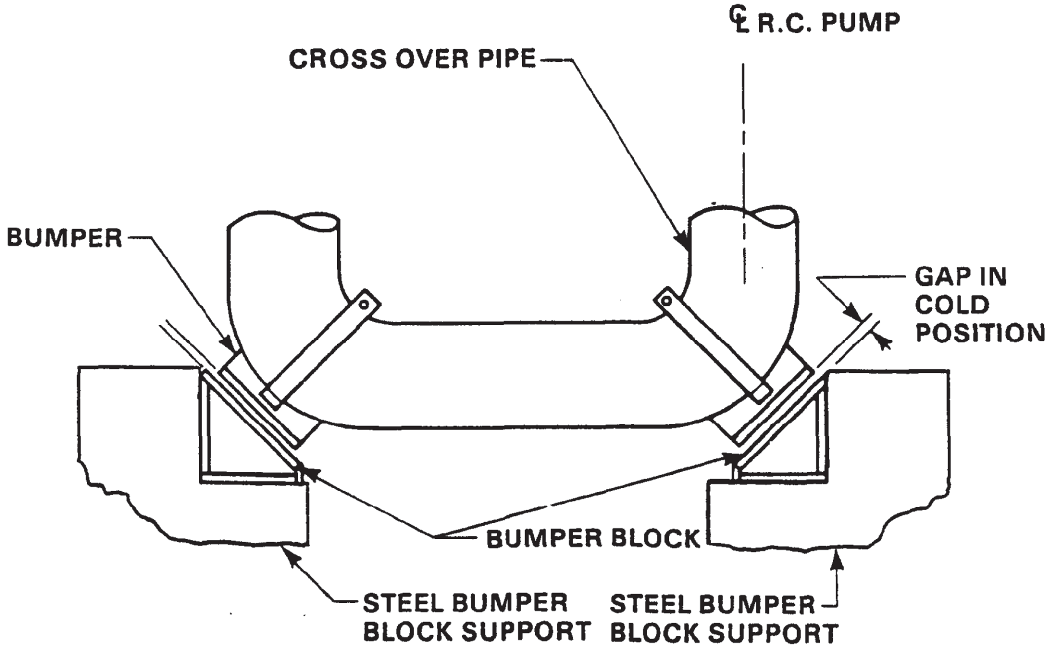
Figure 5.4-14 Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURIZER SUPPORTS

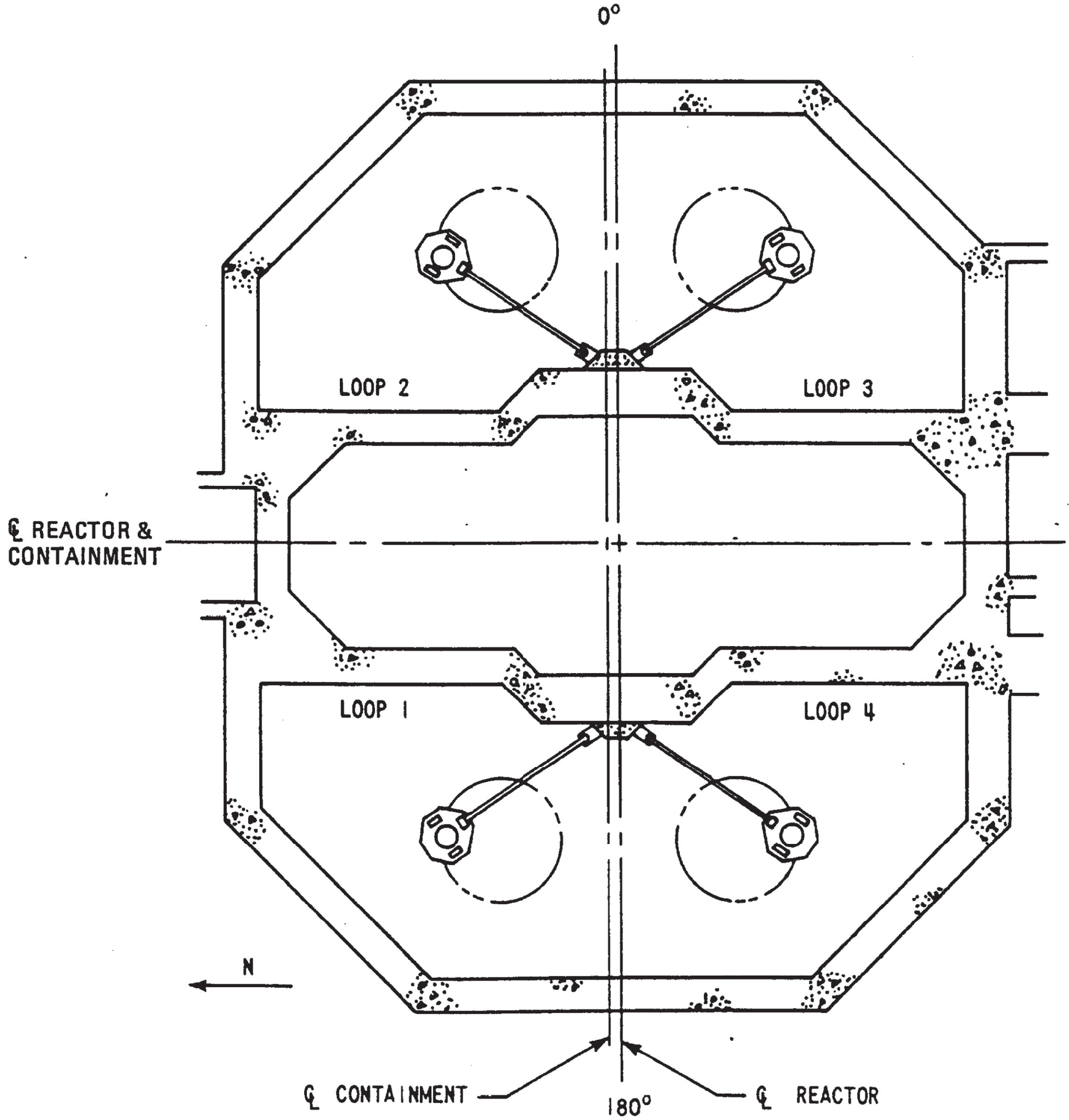
Figure 5.4-15 Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

CROSSOVER LEG SUPPORT

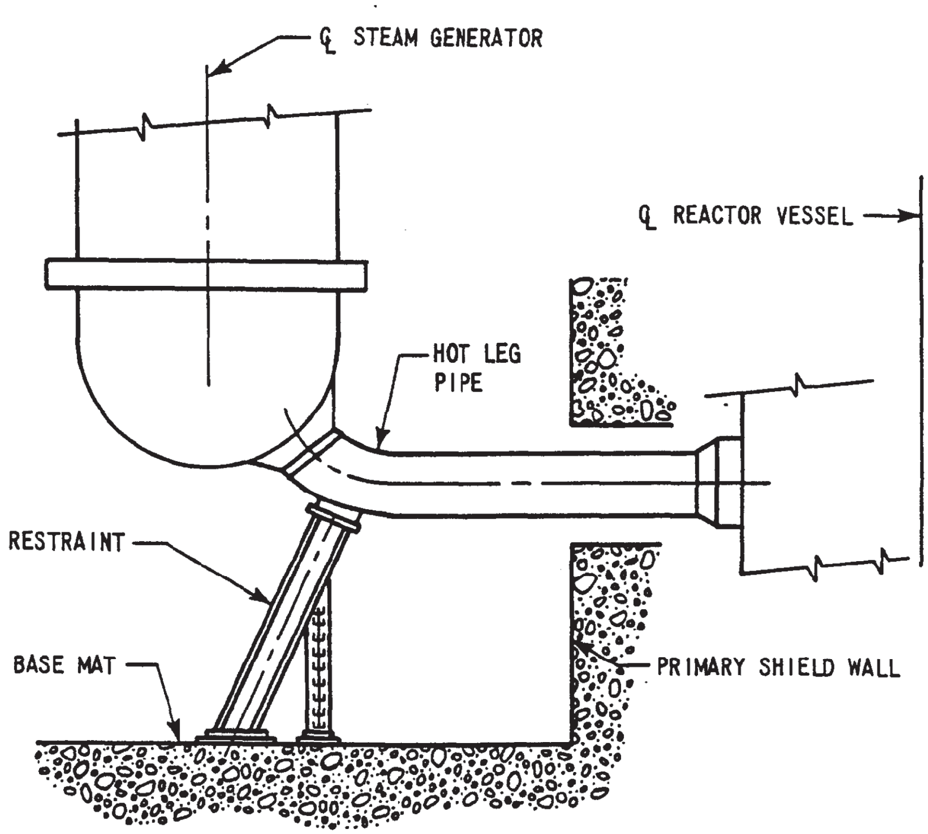
Figure 5.4-16 Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

CROSSOVER LEG VERTICAL RUN RESTRAINTS

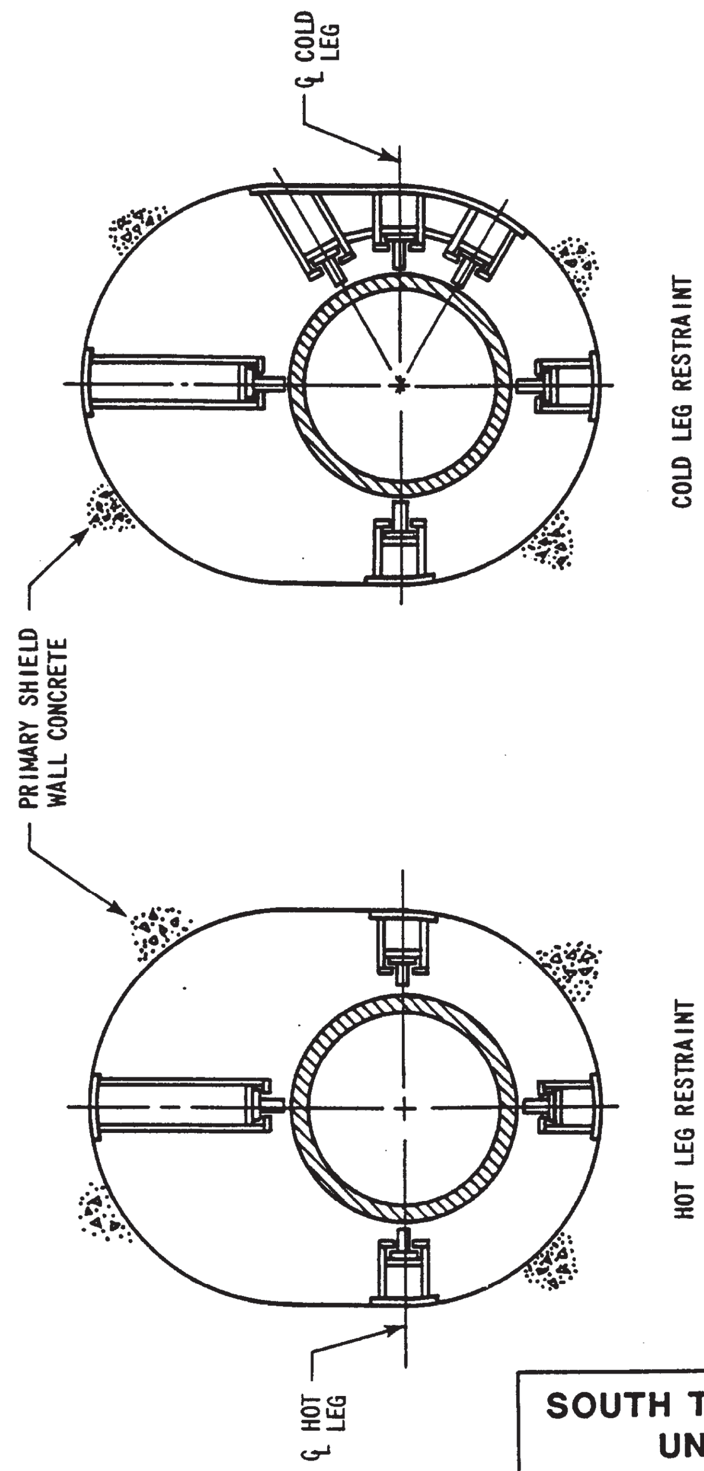
Figure 5.4-17 Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

HOT LEG PIPE WHIP RESTRAINT

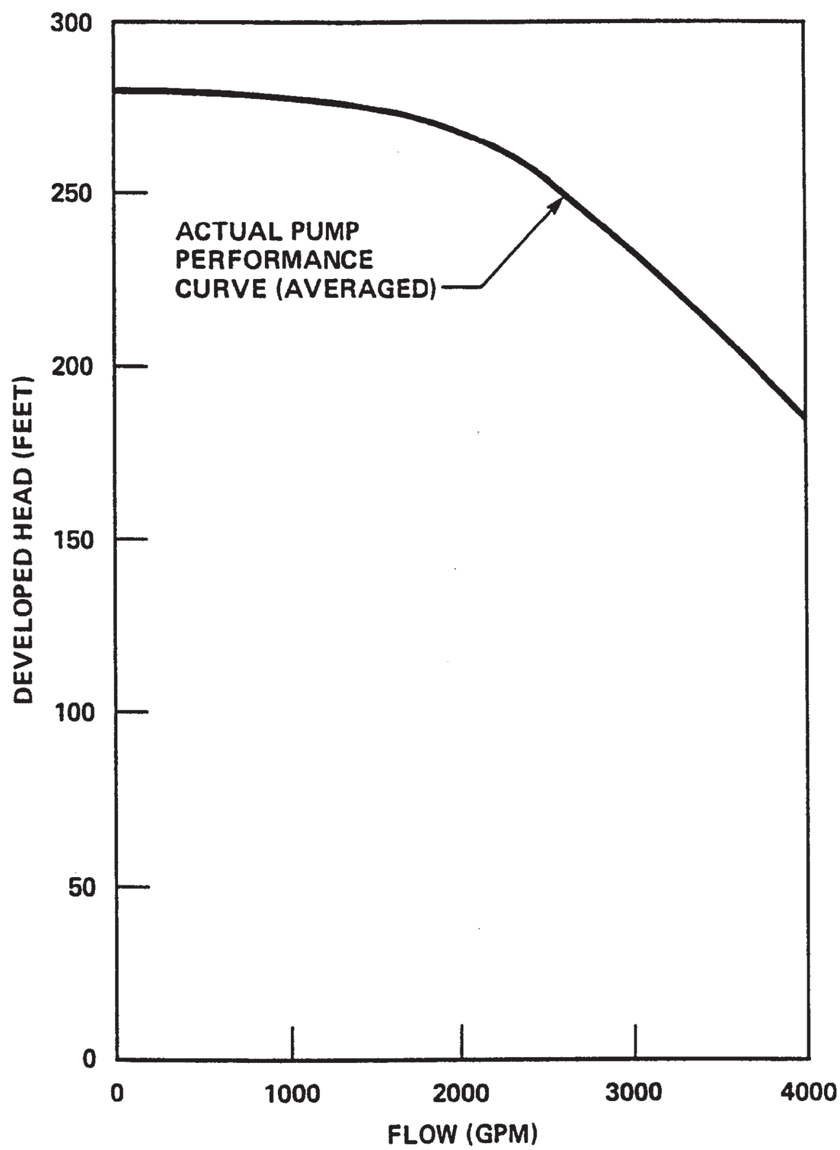
Figure 5.4-18 Revision 0



SOUTH TEXAS PROJECT
UNITS 1 & 2

HOT AND COLD LEG PRIMARY SHIELD
WALL PIPE WHIP RESTRAINTS

Figure 5.4-19 Revision 0



Residual Heat Removal Performance Curves

SOUTH TEXAS PROJECT
UNITS 1 & 2

RESIDUAL HEAT REMOVAL PUMP
PERFORMANCE CURVES

Figure 5.4-20 Revision 0