

WOLF CREEK

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REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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CHAPTER 5.0

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

5.1.1 DESIGN BASES

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

- a. The RCS has the capability to transfer to the steam and power conversion system the heat produced during power operation and when the reactor is subcritical, including the initial phase of plant cooldown.
- b. The RCS has the capability to transfer to the residual heat removal system the heat produced during the subsequent phase of plant cooldown and cold shutdown.
- c. The RCS heat removal capability under power operation and normal operational transients, including the transition from forced to natural circulation, assures no fuel damage within the operating bounds permitted by the reactor control and protection systems.
- d. The RCS provides the water used as the core neutron moderator and reflector and as a solvent for chemical shim control.
- e. The RCS maintains the homogeneity of the soluble neutron poison concentration and the rate of change of the coolant temperature, so that uncontrolled reactivity changes do not occur.
- f. The RCS pressure boundary is capable of accommodating the temperatures and pressures associated with operational transients.
- g. The reactor vessel supports the reactor core and control rod drive mechanisms.
- h. The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, the pressurizer accommodates volume changes in the reactor coolant.

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- i. The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.
- j. The steam generators provide high quality steam to the turbine. The tube and tubesheet boundary are designed to prevent the transfer of radioactivity generated within the core to the secondary system.
- k. The RCS piping contains the coolant under operating temperature and pressure conditions and limits 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.
- l. The RCS is monitored for loose parts, as described in Section 4.4.6.

5.1.2 DESIGN DESCRIPTION

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

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

The RCS pressure boundary is 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.

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.

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Spring-loaded safety valves and power-operated relief valves from the pressurizer provide for steam discharge from the RCS. Discharged steam is piped to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

The extent of the RCS is defined as:

- a. The reactor vessel, including control rod drive mechanism housings
- b. The portion of the steam generators containing reactor coolant
- c. Reactor coolant pumps
- d. The pressurizer
- e. Safety and relief valves
- f. The interconnecting piping, valves, and fittings between the principal components listed above
- g. The 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

The RCS is shown schematically in Figure 5.1-2. Included on this figure is a tabulation of principal pressures and temperatures and the flow rate of the system under normal steady state full power operating conditions. These parameters are based on the best estimate flow at the pump discharge. RCS volume under the above conditions is presented in Table 5.1-1.

A piping and instrumentation diagram of the RCS is shown in Figure 5.1-1. 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. Figure 1.2-9 and Figures 1.2-11 through 1.2-18 provide plan and elevation views of the reactor building. These figures show principal dimensions of reactor coolant system components in relationship with supporting and surrounding steel and concrete structures and demonstrate the protection provided to the reactor coolant system by its physical layout.

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5.1.3 SYSTEM COMPONENTS

The major components of the RCS are as follows:

a. Reactor vessel

The reactor vessel is cylindrical and has a welded, hemispherical bottom head and a removable, flanged, 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.

b. Steam generators

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

c. Reactor coolant pumps

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

d. Piping

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

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e. 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 and relief and safety valve connections are located in the top head of the vessel.

f. Safety and relief valves

The pressurizer safety valves are of the totally enclosed pop-type. The valves are spring loaded and self activated with back pressure compensation. The power-operated relief valves have electric solenoid actuators. They are operated automatically based on RCS pressure or by remote manual control. Remotely operated valves are provided to isolate the inlet to the power-operated relief valves if excessive leakage occurs. These valves will automatically isolate if the RCS pressure drops below a predetermined value, indicative of a stuck-open, power-operated relief valve.

Steam from the pressurizer safety and relief valves is discharged into the pressurizer relief tank through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature.

5.1.4 SYSTEM PERFORMANCE CHARACTERISTICS

Design and performance characteristics of the RCS are provided in Table 5.1-1.

a. 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 analysis, and 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.

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

b. 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 flow resistances in the reactor vessel, steam generator, and piping and on the best estimate of the reactor coolant pump 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.

c. Thermal design flow

Thermal design flow is the flow rate used as a basis for the reactor core thermal performance, the steam generator thermal performance, and the nominal plant parameters used throughout the design. The thermal design flow accounts for the uncertainties in flow resistances (reactor vessel, steam generator, and piping), reactor coolant pump head, and the methods used to measure flow rate. The thermal design flow is approximately 9.0 percent less than the best estimate flow. The thermal design flow is confirmed when the plant is placed in operation. Tabulations of important design and performance characteristics of the RCS, as provided in Table 5.1-1, are based on the thermal design flow.

d. Mechanical design flow

Mechanical design flow is a conservatively high flow used in the mechanical design of the reactor vessel internals and fuel assemblies. The mechanical design flow is based on a reduced system resistance and on increased pump head capability. The mechanical design flow is approximately 2.0 percent greater than the best estimate flow.

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Pump overspeed due to a turbine generator 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.

e. Flows with one pump shut down

The design procedure for calculation of flows with one pump shut down is similar to the procedure described above for calculating flows with all pumps operating.* For the case where reverse flow exists in the idle loop, the system resistance incorporates the idle loop reverse flow resistance with a stationary pump impeller as a flow path in parallel with the reactor vessel internals.

The thermal design flow uncertainty includes a conservative application of parallel flow uncertainties (reactor internals high, idle loop low) as well as the usual component, pump, and flow measurement uncertainties, thereby resulting in a conservatively low reactor flow rate for the thermal design. The mechanical design flow uncertainty is increased slightly to account for the slightly higher uncertainties at the higher pump flows.

* In reality, WCGS Technical Specifications require a shutdown to hot standby (Mode 3) within 6 hours of a shutdown of a reactor coolant pump when in Mode 1 or 2. Continuous 3 pump operation is not permitted.

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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, ft ³	12,135 ±100*
System liquid volume, including pressurizer water at maximum guaranteed power, ft ³	11,393
Pressurizer spray rate, maximum, gpm	900
Pressurizer heater capacity, kW	1,800

System Thermal and Hydraulic Data

	4 Pumps <u>Running</u>	Uprated NSSS Power <u>of 3,651 MWt</u>
NSSS power, MWt	3,579	3,651
Reactor power, MWt	3,565	3,637
Thermal design flows, gpm		
Active loop	90,300** (10% SGTP)	90,300**
Reactor (core flow only)	330,859*** (10% SGTP)	330,859***
Total reactor flow, 10 ⁶ lb/hr	134.7	134.9
Temperatures, °F		
Reactor vessel outlet	621.0	621.6
Reactor vessel inlet	555.8	555.2
Steam generator outlet	555.5	554.9
Steam generator steam	537.5	536.4
Feedwater	446.0	448.6

*at a nominal T_{avg} of 557°F

**Exact value of 90,324 gpm rounded to the nearest hundred gpm.

***Value reflects a TDF of 90,300 gpm/loop and a design core bypass flow of 8.4%.

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TABLE 5.1-1 (Sheet 2)

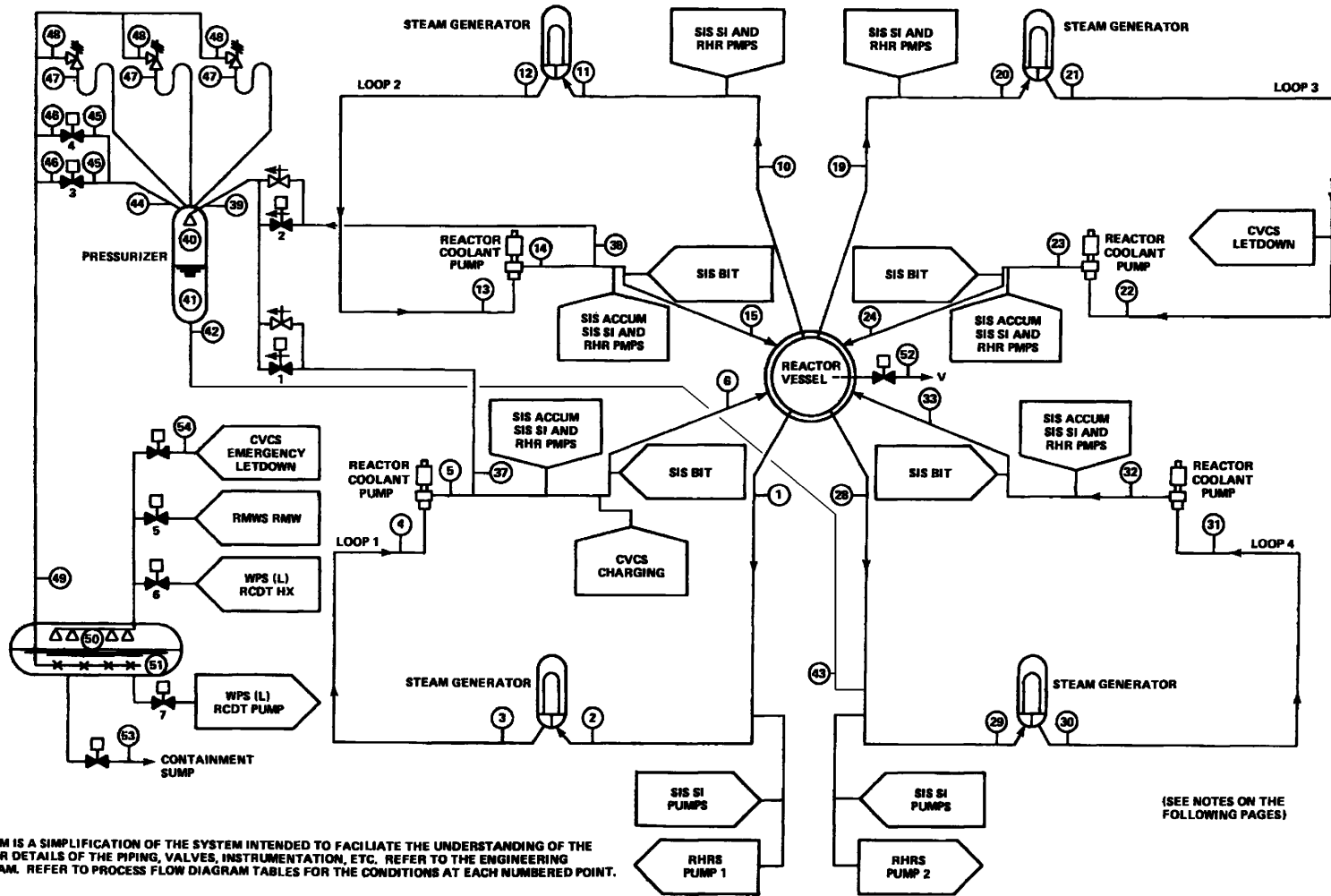
System Thermal and Hydraulic Data

	4 Pumps Running	Uprated NSSS Power of 3,651 MWt
Steam pressure, psia	943	934
Total steam flow, 10 ⁶ lb/hr	15.91	16.29
Best estimate flows, gpm ⁺		
Active loop	102,200 (0% SGTP)	
Reactor (core flow only)	99,200 (10% SGTP)	
	374,461 (0% SGTP)	
	363,469 (10% SGTP)	
Mechanical design flows, gpm ⁺		
Active loop	104,200 (0% SGTP)	
Reactor (core flow only)	381,789 (0% SGTP)	
System Pressure Drops ⁺		
	(T _{avg} = 570.7°F)	(T _{avg} = 588.4°F)
Reactor vessel ΔP, psi	48.7	47.4
Steam generator ΔP, psi	41.3	45.5
Hot leg piping ΔP, psi	1.3	1.2
Crossover leg piping ΔP, psi	3.4	3.1
Cold leg piping ΔP, psi	3.6*	3.3*
Pump head, ft	298	312

+Values applicable to both current power and uprated power of 3,651 MWt

*Includes pump weir ΔP of 2.0 psi.

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NOTES:

THIS DIAGRAM IS A SIMPLIFICATION OF THE SYSTEM INTENDED TO FACILITATE 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.

(SEE NOTES ON THE FOLLOWING PAGES)

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NOTES TO FIGURE 5.1-2

Mode A Steady State Full Power Operation

Key: Basis numbers NSSF 3579 MWt for T_{hot} Maintained @ 10% SG Tube Plugging
 () numbers NSSF 3579 MWt for 15°F T_{hot} Reduction @ 10% SG Tube Plugging

Location	Fluid	Pressure ⁽²⁾ (psig)	Temperature (°F)	Flow gpm ⁽¹⁾	Volume (cu.ft.)
1	Reactor	2,236.2	618.3	110,871	-
	Coolant	(2,236.2)	(601.4)	(109,522)	
2	Reactor	2,235.0	618.3	110,875	-
	Coolant	(2,235.0)	(601.4)	(109,526)	
3	Reactor	2,189.5	558.2	99,310	-
	Coolant	(2,188.4)	(539.7)	(99,294)	
4	Reactor	2,186.4	558.2	99,315	-
	Coolant	(2,185.2)	(539.7)	(99,298)	
5	Reactor	2,286.9	558.5	99,200	-
	Coolant	(2,288.2)	(540.0)	(99,200)	
6	Reactor	2,283.6	558.5	99,205	-
	Coolant	(2,284.8)	(540.0)	(99,204)	
10-15	Reactor Coolant	See Loop #1 Specifications			
19-24	Reactor Coolant	See Loop #1 Specifications			
28-33	Reactor Coolant	See Loop #1 Specifications			
37	Reactor	2,286.9	558.5	1.0	-
	Coolant	(2,288.2)	(540.0)	(1.0)	
38	Reactor	2,286.9	558.5	1.0	-
	Coolant	(2,288.2)	(540.0)	(1.0)	
39	Reactor	2,286.9	558.5	2.0	-
	Coolant	(2,288.2)	(540.0)	(2.0)	

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NOTES TO FIGURE 5.1-2 (Sheet 2)

Mode A Steady State Full Power Operation

Location	Fluid	Pressure ⁽²⁾ (psig)	Temperature (F)	Flow gpm ⁽¹⁾	Volume (cu.ft.)
40	Steam	2,235.0	652.7		720
41	Reactor coolant	2,235.0	652.7		1,080
42	Reactor coolant	2,235.0	652.7	2.5	-
43	Reactor coolant	2,235.0	652.7	2.5	-
44	Steam	2,235.0	652.7	0	-
45	Reactor coolant	2,235.0	<652.7	0	-
46	N ₂	3.0	120	0	-
47	Reactor coolant	2,235.0	<652.7	0	-
48	N ₂	3.0	120	0	-
49	N ₂	3.0	120	0	-
50	N ₂	3.0	120	-	450
51	Pres- surizer relief tank water	3.0	120	-	1,350
52	Steam/H ₂	2,235.0	559	0	-
53	Reactor coolant	3.0	120	0	-
54	Reactor coolant	50	170	0	-

(1) At the conditions specified.

(2) Pressures reflect nonrecoverable losses only (Elevation ΔPs are not included)

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5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

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

- a. Section 6.3 - RCPB components which are part of the emergency core cooling system.
- b. Section 9.3.4 - RCPB components which are part of the chemical and volume control system.
- c. Section 3.9(N).1 - Design loadings, stress limits, and analyses applied to the RCS and ASME Code Class 1 components.
- d. Section 3.9(N).3 - Design loadings, stress limits, and analyses applied to ASME Code Class 2 and 3 components.

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

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10 CFR 50.55a

RCS components are designed and fabricated in accordance with 10 CFR 50, Section 50.55a, "Codes and Standards" except as described below. The addenda of the ASME Code applied in the design of each component are listed in Table 5.2-1.

All components located within the reactor coolant pressure boundary (as defined by 10CFR50.2) are classified as required by 10CFR50.55a with the exception of the pressurizer upper level instrument lines, the pressurizer safety valve loop seal drain lines, 3/4" and smaller branch lines connected to the pressurizer relief lines, and the associated components. These lines are Safety Class 2 although a rupture of one of these lines may result in a rapid depressurization of the reactor coolant system and ECCS actuation on low pressurizer pressure. Relief from the requirements of 10CFR50.55a was authorized by the NRC in accordance with 10CFR50.55a(a)(3)(ii) to allow these lines to remain Safety Class 2 (Reference 11).

5.2.1.2 Applicable Code Cases

Regulatory Guides 1.84 and 1.85 are discussed in Appendix 3A.

Code Case 1528 (SA-508, Class 2a) material was used in the manufacture of the WCGS steam generators and pressurizer. At the time of initial application, Regulatory Guide 1.85 reflected a conditional NRC approval of Code Case 1528. Westinghouse conducted a test program which demonstrated the adequacy of Code Case 1528 material. The results of the test program are documented in Reference 1.

* A component is considered to be any piece or portion of equipment below the system level but above the part level.

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Reference 1 was submitted to the NRC by Reference 2.

The specific code cases used for Wolf Creek are:

Steam Generator: 1484 and 1528
Pressurizer: 1528-3
Piping: 1423-2, N-411, N-391, N-392 & N-318-3*, 1606-1 (N-53)
Valves: 1649, 1769, 1567 and N-3-10

5.2.2 OVERPRESSURE PROTECTION

RCS overpressure protection is accomplished by the utilization of pressurizer safety valves along with the reactor protection system and associated equipment. Combinations of these systems provide compliance with the overpressure requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized water reactor systems.

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

Selected overpressure protection measures for the secondary side are also described in these sections.

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 by means of a common header. The transient which established the design requirements for the primary system overpressure protection is a complete loss of steam flow to the turbine with operation of the steam generator safety valves and maintenance of main feedwater flow. However, for the sizing of the pressurizer safety valves, no credit is taken for reactor trip nor the operation of the following:

- a. Pressurizer power-operated relief valves
- b. Steam line atmospheric relief valve
- c. Steam dump system
- d. Reactor control system
- e. Pressurizer level control system
- f. Pressurizer spray valve

For this transient, the peak RCS and peak steam system pressure are limited to 110 percent of their respective design values.

* Code Case N-318-3 provides several conditions for lug attachment evaluation snubber reduction program (Ref. 13) has listed all stress calculation numbers that used N-318-3 in class 2 and 3 pipe lines. Lug locations are available in pipe support drawings.

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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. These assumptions are the most limiting with respect to system overpressure.

Overpressure protection for the steam system is provided by steam generator safety valves. The steam system safety valve capacity is based on providing enough relief to remove 105 percent of the engineered safeguards design steam flow. This relief capacity may be provided while limiting the maximum steam system pressure to less than 110 percent of the steam generator shell side design pressure.

Blowdown and heat dissipation systems of the NSSS connected to the discharge of pressure relieving devices are discussed in Section 5.4.11, pressurizer relief discharge system.

Steam generator blowdown systems for the balance-of-plant are discussed in Section 10.4.8.

Postulated events and transients on which the design requirements of the overpressure protection system are based are discussed in Reference 3.

5.2.2.2 Design Evaluation

The relief capacities of the pressurizer and steam generator safety valves are determined from the postulated overpressure transient conditions in conjunction with the action of the reactor protection system. An evaluation of the functional design of the system to perform its function is presented in Reference 3. The results of the analysis performed at the uprated power condition also confirm that the design of the overpressure protection system will continue to perform its function under uprated power condition. The analysis showed that when the first reactor protection system trip signal (following a direct reactor trip signal on turbine trip) was ignored, the primary and secondary coolant overpressure protection systems provided sufficient pressure relief to ensure that the peak pressure of both coolant systems remained below the Technical Specification limit of 110% of their respective design pressures. The analysis further demonstrated that, when the second and third trip signals were ignored, the overpressure protection systems maintained the primary and secondary coolant pressures below 110% of their design pressures and thus confirmed adequate safety valve sizing exists under uprated power conditions.

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Reference 3 describes in detail the types and number of pressure relief devices employed, relief device description, locations in the systems, reliability history, and the details of the methods used for relief device sizing based on typical worst-case transient conditions and analysis data for each transient condition. The description of the analytical model used in the analysis of the overpressure protection system and the basis for its validity are discussed in Reference 8. An evaluation of the overpressure protection system's design was performed to ensure that the conclusions presented in Reference 3 remain valid under the uprated power conditions. The evaluation followed the methodology presented in Reference 3 utilizing the analytical model described in Reference 8.

A description of the pressurizer safety valves performance characteristics along with the design description of the incidents, assumptions made, method of analysis, and conclusions are discussed in Chapter 15.0.

5.2.2.3 Piping and Instrumentation Diagrams

Overpressure protection for the RCS is provided by pressurizer safety valves shown in Figure 5.1-1, Sheet 2.

These discharge to the pressurizer relief tank by means of a common header.

The steam system safety valves are discussed in Section 10.3 and are shown on Figure 10.3-1, Sheet 2.

5.2.2.4 Equipment and Component Description

The operation, significant design parameters, number and types of operating cycles, and environmental conditions of the pressurizer safety valves are discussed in Sections 5.4.13, 3.9(N).1, and 3.11(N).

Section 10.3 contains a discussion of the equipment and components of the steam system overpressure system.

5.2.2.5 Mounting of Pressure-Relief Devices

The design bases for the assumed loads for the primary and secondary side pressure relief devices of the steam generator are described in Paragraph 3.9(B).3.3.

5.2.2.5.1 Location of Pressure Relief Devices

Figure 5.2-1 provides typical design and installation details for pressure relief devices mounted on the secondary side of the steam generator. Pressure relief devices for the reactor coolant system are three pressurizer safety relief valves and two power-operated relief valves. These valves discharge to the pressurizer relief tank via a common header.

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5.2.2.5.2 Pressurizer Safety Relief Valves

The pressurizer safety valve discharge piping system is a closed system in which no sustained reaction force from a free discharging jet of fluid can exist. However, transient hydraulic forces are imposed at various points in the piping system from the time a safety valve begins to open until a steady flow is completely developed. Since a water loop seal is applied, transient hydraulic forces caused by the liquid being forced through the safety valve and then accelerated down the piping system does occur.

The pressurizer relief devices are mounted and installed as follows:

- a. Each straight leg of the discharge pipe is supported to take the valve discharge transient force along that leg.
- b. The supports at the valve discharge piping are connected to the adjacent structure.
- c. Snubbers are used to restrain the valve discharge transient forces when thermal movements are of a high magnitude.

Subprogram RVDFT (relief valve discharge flow transients) was used to predict the transient flows resulting from actuation of a safety relief valve under normal operating conditions. It also predicted the resulting piping loads as a function of time to be used as dynamic forcing functions for structural design of discharge piping and its supporting components. The computation was based on finite difference solutions by the method of effluent characteristics. The computed transient forces were then used to calculate loads on pipe bends and on pipe runs.

A static analysis was performed for thermal, weight, and seismic anchor movement loadings on the discharge piping. A dynamic analysis for seismic and valve discharge loadings was also performed to verify the design of the support configuration. The results of these analyses are described below:

- a. For loading combinations see Table 3.9(B)-2.
- b. Material Type

Class I Piping	3" Sch. 160, SA-312, TP-304
	6" Sch. 160, SA-312, TP-304
B31.1 Piping	3" Sch. 80S, SA-312, TP-304
	6" Sch. 80S, SA-312, TP-304
	12" Sch. 80S, SA-312, TP-304

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c. Maximum stress points within piping system

Class I Piping	Node point - 405	
	Type - reducer	
	Max. primary stress	18,092 psi
	Allowable primary stress	24,282 psi
B31.1 Piping	Node point - 310	
	Max. primary stress	14,811 psi
	Allowable primary stress	22,560 psi
	Node point - 555	
	Max. primary + secondary stress	33,995 psi
	Allowable primary + secondary stress	43,375 psi

5.2.2.5.3 Main Steam Safety Relief Valves

Figure 5.2-1 provides design and installation details.

The steady-state flow condition reached after the valve has opened and is exhausting into the stack was considered in the stress analysis of the safety valve installation. With these conditions, the valve moments are balanced due to the split valve discharge design, and the vertical discharge thrust force is reacted by the header supports via the header. The discharge force from the vent stack is reacted by an in-line anchor and the supports near the top of the stack. The effects of thermal expansion, pipe weight, seismic anchor movements, seismic occurrence, and relief valve discharge thrust forces were considered in the stress analysis of the vent stack piping. These effects were also considered in the stress analysis of the main steam header piping in addition to the water hammer effects caused by fast valve closure of the main steam isolation valves.

A 10 percent unbalanced discharge from the two split discharge ports of each safety valve was assumed for the stress analysis of the header piping. Therefore, one discharge port had an assumed vertical thrust load of 13,574 pounds and the other an assumed thrust load of 12,227 pounds. These values are based on a relief valve discharge from a line pressure of 1,185 psi and a dynamic load factor of 1.2. It was conservatively assumed that each valve opened simultaneously, resulting in the following header stresses and support loads:

- a. For loading combinations see Table 3.9(B)-2 and Table 3.9(B)-10.

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b. Material type
 28-inch OD wall thickness of 1.5 inch, SA 106, Gr C.

c. Maximum stress points within system
 Node point - 83
 Maximum primary stress 9,287 psi
 Allowable primary stress 21,000 psi
 Node point - 5
 Maximum secondary stress 4,112 psi
 Allowable secondary stress 26,250 psi

d. Support loads

Node Point	Header Support Loads (vertical supports and loads only)
5	21,942 lbs
33	187,800 lbs
83	112,700 lbs
85	166,300 lbs
300	33,347 lbs
294	187,800 lbs
282	112,700 lbs
281	166,300 lbs
347	33,362 lbs
341	187,800 lbs
329	112,700 lbs
328	166,300 lbs
397	10,100 lbs
391	184,400 lbs
380	112,800 lbs
379	166,300 lbs

5.2.2.6 Applicable Codes and Classification

The requirements of ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 (Overpressure Protection Report) and NC-7300 (Overpressure Protection Analysis), are followed and complied with for pressurized water reactor systems.

Piping, valves, and associated equipment used for overpressure protection are classified in accordance with ANS-N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." These safety class designations are delineated on Table 3.2-1 and shown on Figure 5.1-1.

For further information, refer to Section 3.9(N).

5.2.2.7 Material Specifications

Refer to Section 5.2.3 for a description of material specifications.

5.2.2.8 Process Instrumentation

Each pressurizer safety valve discharge line incorporates a control board temperature indicator and alarm to notify the operator of steam discharge due to either leakage or actual valve operation. Safety-related control room positive position indication is provided for the PORVs and safety valves. For a further discussion on process instrumentation associated with the system, refer to Chapter 7.0.

5.2.2.9 System Reliability

The reliability of the pressure relieving devices is discussed in Section 4 of Reference 3.

5.2.2.10 RCS Pressure Control During Low Temperature Operation

Administrative procedures were developed to aid the operator in controlling RCS pressure during low temperature operation. However, to provide a back-up to the operator and to minimize the frequency of RCS overpressurization, an automatic system is provided to maintain pressures within allowable limits.

Analyses have shown that one pressurizer power-operated relief valve is sufficient to prevent violation of these limits due to anticipated mass and heat input transients. However, redundant protection against an overpressurization event is provided through the use of two pressurizer power-operated relief valves to mitigate any potential pressure transients. The mitigation system is required only during low temperature water solid operation when it is manually armed and automatically actuated.

5.2.2.10.1 System Operation

Two pressurizer power-operated relief valves are supplied with actuation logic to ensure that a redundant and independent RCS pressure control back-up feature is provided for the operator during low temperature operations. This system provides the capability for RCS inventory letdown, thereby maintaining RCS pressure within allowable limits. Refer to Sections 5.4.7, 5.4.10, 5.4.13, 7.6.6, and 9.3.4 for additional information on RCS pressure and inventory control during other modes of operation.

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The basic function of the system logic is to continuously monitor RCS temperature and pressure conditions whenever plant operation is at low temperatures. An auctioneered system temperature is continuously converted to an allowable pressure and then compared to the actual RCS pressure. The system logic first annunciates a main control board alarm whenever the measured pressure approaches within a predetermined amount of the allowable pressure thereby indicating that a pressure transient is occurring. On a further increase in measured pressure, an actuation signal is transmitted to the pressurizer power-operated relief valves when required to mitigate the pressure transient.

5.2.2.10.2 Evaluation of Low Temperature Overpressure Transients

The ASME Code (Section III, Appendix G) establishes guidelines and upper limits for RCS pressure primarily for low temperature conditions less than approximately 350 F. The mitigation system discussed in Section 5.2.2.10.1 addresses these conditions as discussed in the following paragraphs.

Two specific transients: mass input and heat input, with the RCS in a water-solid condition; have been considered as the design basis for the Low Temperature Overpressure Protection (LTOP) system. Each of these scenarios assumes as an initial condition that the RHRS is isolated from the RCS, and thus the relief capability of the RHRS relief valves is not available. Transient analyses have been performed to determine the maximum pressure for the postulated mass input and heat input events.

The LTOP PORV setpoint limit curve (PTLR Figure 2.2-1) is determined based on the updated heatup and cooldown limit curves, and the analysis results of limiting Low Temperature Over-Pressure (LTOP) transients. The methodology for this determination is given in Reference 10. The limiting LTOP mechanisms analyzed for WCGS under water solid conditions were:

a. FOR LIMITING MASS ADDITION LTOP MECHANISM

Operation of one Centrifugal Charging Pump (CCP) and the Normal Charging Pump (NCP) with instrument air failure resulting in the flow control valve in the letdown line failing closed (letdown isolation) and the flow control valve in the charging line failing open (maximum charging flow), and

b. FOR LIMITING HEAT ADDITION LTOP MECHANISM

Inadvertent start-up of a reactor coolant pump with a maximum 50°F temperature mismatch between the RCS and the hotter steam generators.

These analyses, using the LOFTRAN computer code, take into consideration pressure overshoot and undershoot beyond the PORV open and close setpoints, which can occur as a result of time delays in signal processing and valve stroke times. The maximum expected pressure overshoot and undershoot calculated from the limiting mass input and heat input transients, in conjunction with the 10 CFR 50, Appendix G, pressure limits and reactor coolant pump No. 1 seal pressure limit, are utilized in the selection of the pressure setpoints for the PORV. The mass injection rate assumed in the design basis mass input transient is based on 100% flow capacity of the NCP and one CCP. The maximum combined pump flow has been assumed in order to envelop the maximum flow possible by the operational configuration that uses the NCP for charging with one CCP remaining operable, or the use of one CCP for charging with the NCP remaining operable, during shutdown modes.

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Both the heat input and mass input analyses take into account the single failure criteria and therefore, only one pressurizer power-operated relief valve was assumed to be available for pressure relief. The above events have been evaluated considering the allowable pressure/temperature limits established by the Appendix G guidelines. The evaluation of the transient results concluded that reactor vessel integrity is not impaired.

5.2.2.10.3 Operating Basis Earthquake Evaluation

A fluid systems evaluation has been performed considering the potential for overpressure transients following an operating basis earthquake.

The pressurizer power-operated relief valves have been designed in accordance with the ASME Code and seismically qualified under the Westinghouse valve operability program which is discussed in Section 3.9(N).3.2.

Therefore, the overpressurizer mitigation system is available to provide pressure relief following an operating basis earthquake.

5.2.2.10.4 Administrative Procedures

Although the system described in Section 5.2.2.10.1 was installed to maintain RCS pressure within allowable limits, administrative procedures minimize the potential for and the consequences of any transient that could actuate the over-pressure relief system. The following discussion highlights these procedural controls, listed in hierarchy of their function in mitigating RCS cold overpressurization transients.

5.2.2.10.4.1 Normal and Transitional Operation

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

An adequate cushion substantially reduces the severity of potential pressure transients, such as reactor coolant pump 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 further highlight precautions that minimize the severity of, or the potential for, developing an overpressurization transient. The following precautions or measures were considered in developing the operating procedures:

- a. The residual heat removal inlet lines from the reactor coolant loop are normally open when the RCS pressure is less than 425 psig. This precaution assures that there

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is a relief path from the reactor coolant loop to the residual heat removal suction line relief valves when the RCS is at low pressure and is water solid.

- b. Whenever the plant is water solid and the reactor coolant pressure is being maintained by the low pressure letdown control valve, letdown flow normally bypasses the normal letdown orifices. In addition, all three letdown orifices may be open.
- c. If all reactor coolant pumps have stopped for more than 5 minutes during plant heatup and the reactor coolant temperature is greater than the charging and seal injection water temperature, a steam bubble is formed in the pressurizer prior to restarting a reactor coolant pump. This precaution minimizes the pressure transient when the pump is started and the cold water previously injected by the charging pumps is circulated through the warmer reactor coolant components. The steam bubble accommodates the resultant expansion as the cold water is rapidly warmed.
- d. If all reactor coolant pumps are stopped and the RCS is being cooled down by the residual heat exchangers, a nonuniform temperature distribution may occur in the reactor coolant loops. Prior to restarting a reactor coolant pump, a steam bubble is formed in the pressurizer or an acceptable temperature profile is demonstrated.
- e. During plant cooldown, all steam generators are normally connected to the steam header to assure a uniform cooldown of the reactor coolant loops.
- f. At least one reactor coolant pump normally remains in service until the reactor coolant temperature is reduced to 160 F.

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

The specific plant configurations of emergency core cooling system testing and alignment also highlight procedural recommendations to prevent developing cold overpressurization transients.

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During these limited periods of plant operation, the following precautions/measures were considered in developing the operating procedures:

- a. To preclude inadvertent emergency core cooling system actuation during heatup and cooldown, procedures require blocking the low pressurizer pressure, and low steam line pressure signal actuation logic at 1,900 psig.
- b. During further cooldown, closure and power lockout of the accumulator isolation valves with one centrifugal charging pump and both safety injection pumps rendered incapable of injecting into the RCS in accordance with WCGS Technical Specifications, provide additional back-up to item a above.
- c. The recommended procedure for periodic emergency core cooling system pump performance testing is to test the pumps during normal power operation or at hot shutdown conditions. This precludes any potential for developing a cold overpressurization transient.

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

If CSD testing with the vessel closed is necessary, the procedures require emergency core cooling system pumps discharge valve closure and RHRS alignment to isolate potential emergency core cooling system pump input and to provide back-up benefit of the RHRS relief valves.

- d. SIS circuitry testing, if done during CSD, requires RHRS alignment and one centrifugal charging pump and both safety injection pumps rendered incapable of injecting into the RCS to preclude developing cold overpressurization transients.

The above procedural precautions covering normal operations with a steam bubble, transitional operations where potentially water solid, and specific testing operations provide in-depth cold overpressure preventions or reductions, augmenting the installed overpressure relief system.

5.2.2.10.4.2 Failure of Both PORVs

Should both of the PORVs fail closed at a time when the RHR letdown isolation valves for either or both RHR loops are open, the RCS is protected from overpressurization by the RHR inlet relief

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valves. Although the valves are only required to relieve the flow of a single centrifugal charging pump delivering at its maximum rate, the valves are each conservatively sized to relieve the combined flow of both centrifugal charging pumps at a setpoint of 450 psig.

During normal startup and shutdown, a pressurizer bubble is maintained whenever the RHR system is isolated. The normal steam bubble volume in this condition would be approximately 1350 ft³. Should normal letdown be isolated, the maximum makeup rate imbalance would be determined by the head/flow curve of the centrifugal charging pump, which could be in operation. This rate would actually be much less as the transient progressed, since the charging flow control system would throttle the flow to try to maintain pressurizer level. However, even if no credit is taken for the charging control system, and assuming that the pressurizer level is initially at the high level alarm setpoint (i.e., approximately 567 ft³ steam bubble), the plant operator would have greater than 10 minutes to terminate the event to prevent overflow of the pressurizer.

5.2.2.11 Testing and Inspection

Testing and inspection of the overpressure protection components are discussed in Section 5.4.13.4 and Chapter 14.0.

5.2.3 MATERIALS SELECTION, FABRICATION, AND PROCESSING

5.2.3.1 Material Specifications

Material specifications used for the principal pressure retaining applications in components of 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. Tables 5.2-2 and 5.2-3 also include the material specifications of unstabilized austenitic stainless steel used for components in systems required for reactor shutdown and for emergency core cooling.

The material specifications of unstabilized austenitic stainless steel used for reactor vessel internals which are essential for emergency core cooling and for core structural support are listed in Table 5.2-4.

Table 5.2-3 is not totally inclusive of the material specifications used in the listed applications. However, the listed specifications are representative.

The materials utilized conform to the applicable ASME Code rules.

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

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

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

5.2.3.2 Compatibility With Reactor Coolant

5.2.3.2.1 Chemistry of Reactor Coolant

The RCS chemistry specifications are given in Table 5.2-5.

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

The chemical and volume control system provides a means for adding chemicals to the RCS which perform the following functions: 1) control the pH of the coolant during pre-startup testing and subsequent operation, 2) scavenge oxygen from the coolant during heatup, and 3) control radiolysis reactions involving hydrogen, oxygen, and nitrogen during all power operations subsequent to startup. The normal limits for chemical additives and reactor coolant impurities for power operation are shown in Table 5.2-5.

The pH control chemical utilized is lithium hydroxide monohydrate, enriched in the lithium-7 isotope to 99.9 percent. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/inconel systems. In addition, lithium-7 is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The lithium-7 hydroxide is introduced into the RCS via the charging flow. The solution is prepared in the laboratory and transferred to the chemical additive tank. Reactor makeup water is then used to flush the solution to the suction header of the charging pumps. The concentration of lithium-7 hydroxide in the RCS is maintained in the range specified for pH control and is controlled by administrative procedures. If the concentration exceeds this range, the cation bed demineralizer is employed in the letdown line in series operation with the mixed bed demineralizer.

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During reactor startup from the cold condition, hydrazine is employed as an oxygen scavenging agent. The hydrazine solution is introduced in accordance with plant operating procedures.

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. This can be adjusted to provide the correct equilibrium hydrogen concentration.

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

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

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

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

Ferritic low alloy and carbon steel nozzles have safe ends of 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 post-weld heat treatment when the nozzle is larger than a 4-inch nominal inside diameter and/or the wall thickness is greater than 0.531 inches.

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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 material specifications.

During subsequent fabrication, these materials are not heated above 800 F other than locally by welding operations. The solution annealed surge line material is subsequently formed by hot bending followed by a resolution 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 pressurized water reactor systems because chlorides, fluorides, and oxygen are controlled to very low levels.

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

In general, all of the materials listed in 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 the reflective stainless steel type or made of compounded materials which yield low leachable chloride and/or fluoride concentrations. The compounded materials in the form of blocks, boards, cloths, tapes, adhesives, cements, etc., are silicated to provide protection of austenitic stainless steels against stress corrosion which may result from accidental wetting of the insulation by spillage, minor leakage, or other contamination from the environmental atmosphere. Appendix 3A includes a discussion which indicates the degree of conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is anticipated from service experience, such as valve packing, pump seals, etc., only materials which are compatible with the coolant are used. These are as shown in 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.

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5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

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

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

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

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

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

The fracture toughness tests for WCGS reactor coolant pressure boundary components were performed by qualified operators in accordance with written procedures.

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.

Appendix 3A includes discussions which indicate the degree of conformance of the ferritic materials components of the RCPB with Regulatory Guides 1.34, "Control of Electroslag Weld Properties,"

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1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,"
1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and
1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

Sections 5.2.3.4.1 through 5.2.3.4.5 address Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and present the methods and controls utilized by Westinghouse to avoid sensitization and prevent intergranular attack of austenitic stainless steel components. Also, Appendix 3A includes a discussion which indicates the degree of conformance with Regulatory Guide 1.44.

5.2.3.4.1 Cleaning and Contamination Protection Procedures

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

The process specifications which define these requirements and which follow the guidance of the American National Standards Institute N-45 Committee specifications are as follows:

Process Specification Number

82560HM	Requirements for Pressure Sensitive Tapes for Use on Austenitic Stainless Steels
83336KA	Requirements for Thermal Insulation Used on Austenitic Stainless Steel Piping and Equipment
83860LA	Requirements for Marking of Reactor Plant Components and Piping
84350HA	Site Receiving Inspection and Storage Requirements for Systems, Material, and Equipment

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84351NL	Determination of Surface Chloride and ^o Fluoride on Austenitic Stainless Steel Materials
85310QA	Packaging and Preparing Nuclear Components for Shipment and Storage
292722	Cleaning and Packaging Requirements of Equipment for Use in the NSSS
597756	Pressurized Water Reactor Auxiliary Tanks Cleaning Procedures
597760	Cleanliness Requirements During Storage Construction, Erection and Start-Up Activities of Nuclear Power System

Appendix 3A includes a discussion which indicates the degree of conformance of the austenitic stainless steel components of the RCPB with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of^oFluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

5.2.3.4.2 Solution Heat Treatment Requirements

The austenitic stainless steels listed in Tables 5.2-2, 5.2-3, and 5.2-4 are utilized in the final heat treated condition required by the respective ASME Code, Section II materials specification for the particular type of grade of alloy.

5.2.3.4.3 Material Testing Program

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

5.2.3.4.4 Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels

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

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

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- b. A sensitized steel
- c. 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, reliance is placed on the elimination of conditions a and b to prevent intergranular attack on wrought stainless steel components.

This is accomplished by:

- a. Control of primary water chemistry to ensure a benign environment.
- b. Utilization of materials in the final heat treated condition and the prohibition of subsequent heat treatments in the 800 and 1,500°F temperature range.
- c. Control of welding processes and procedures to avoid heat affected zone sensitization.
- d. 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.

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

The water chemistry in the RCS is controlled by the Technical Requirements Manual and plant procedures to prevent the intrusion of aggressive species. Reference 5 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 laboratory tests and operating experience. The long-time exposure of severely sensitized stainless in early Westinghouse pressurized water reactors to reactor coolant environments has not resulted in any sign of intergranular attack. Reference 5 describes the laboratory experimental findings and reactor operating experience. The additional years of operations since the issuance of Reference 5 have provided further confirmation of the earlier conclusions that severely sensitized stainless steels do not undergo any intergranular attack in Westinghouse pressurized water reactor coolant environments.

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In spite of the fact that there never has been any evidence that pressurized reactor coolant water attacks sensitized stainless steels, Westinghouse considers it good metallurgical practice to avoid the use of sensitized stainless steels in the nuclear steam supply system components. Accordingly, measures are taken to prohibit the purchase of sensitized stainless steels and to prevent sensitization during component fabrication. Wrought austenitic stainless steel stock used for components that are part of: 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 under postulated accident conditions) is utilized in one of the following conditions:

- a. Solution annealed and water quenched, or
- b. Solution annealed and cooled through the sensitization temperature range within less than approximately 5 minutes

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

The heat-affected zones of welded components must, of necessity, be heated into the sensitization temperature range, 800 to 1,500°F. However, severe sensitization, i.e., continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion, can be avoided by controlling welding parameters and welding processes. The heat input* and associated cooling rate through the carbide precipitation range are of primary importance. Westinghouse has demonstrated this by corrosion testing a number of weldments.

*Heat input is calculated according to the formula:

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

Where:

- H = joules/in.
- E = volts
- I = amperes
- S = travel speed, in./min.

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Of 25 production and qualification weldments tested, representing all major welding processes, and a variety of components, and incorporating base metal thicknesses from 0.10 to 4.0 inches, only portions of two were severely sensitized. Of these, one involved a heat input of 120,000 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. In only the socket weld did the sensitized condition exist at the surface, where the material is exposed to the environment. The component has been redesigned, and a material change has been made to eliminate this condition.

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

- a. Prohibiting the use of block welding
- b. Limiting the maximum interpass temperature to 350°F
- c. Westinghouse exercising approval rights on all welding procedures

5.2.3.4.5 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures

As described in the previous section, it is not normal Westinghouse practice to expose unstabilized austenitic stainless steels to the sensitization range of 800 to 1,500°F during fabrication into components. If, during the course of fabrication, the steel was inadvertently exposed to the sensitization temperature range, 800 to 1,500°F, the material could be tested in accordance with ASTM A 262, as amended by Westinghouse Process Specification 84201MW, to verify that it is not susceptible to intergranular attack, except that testing is not required for:

- a. Cast metal or weld metal with a ferrite content of 5 percent or more,
- b. Material with a carbon content of 0.03 percent or less that is subjected to temperatures in the range of 800 to 1,500°F for less than 1 hour,
- c. Material exposed to special processing provided the processing is properly controlled to develop a uniform product and provided that adequate documentation exists of service experience and/or test data to demonstrate that the processing will not result in increased susceptibility to intergranular stress corrosion.

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If it was not verified that such material is not susceptible to intergranular attack, the material would have been resolution annealed and water quenched or rejected.

5.2.3.4.6 Control of Welding

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

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

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

The fabrication and installation specifications require welding procedure and welder qualification in accordance with Section III, and include the delta ferrite determinations for the austenitic stainless steel welding materials that are used for welding qualification testing and for production processing. Specifically, the undiluted weld deposits of the "starting" welding materials are required to contain a minimum of 5-percent delta ferrite* as determined by chemical analysis and calculation, using the appropriate weld metal constitution diagrams in Section III. When new

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

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welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, they are performed in accordance with the requirements of Section III and Section IX.

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

The "starting" welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of Section III. The austenitic stainless steel welding material conforms to ASME weld metal analysis A-7 (designated A-8 in the 1974 Edition of the ASME Code), Type 308 or 308L for all applications. Bare weld filler metal, including consumable inserts, used in inert gas welding processes conform to ASME SFA 5.9, and are procured to contain not less than 5-percent delta ferrite according to Section III. Weld filler metal materials used in flux shielded welding processes conform to ASME SFA 5.4 or 5.9 and are procured in a wire-flux combination to be capable of providing not less than 5-percent delta ferrite in the deposit according to Section III. Welding materials are tested, using the welding energy inputs to be employed in production welding.

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

Welding systems are also subject to 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 requirements. Fabrication and installation welds are inspected using nondestructive examination methods according to Section III rules.

To assure the reliability of these controls, Westinghouse has completed a delta ferrite verification program, described in Reference 6, which has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative for conformance with the NRC Interim Position on Regulatory Guide 1.31. The Regulatory Staff's acceptance letter and topical report evaluation were received on December 30, 1974. The program

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results, which do support the hypothesis presented in Reference 6, are summarized in Reference 7.

Appendix 3A includes discussions which indicate the degree of conformance of the austenitic stainless steel components of the RCPB with Regulatory Guides 1.34, "Control of Electroslag Properties," and 1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.4 INSERVICE INSPECTION AND TESTING OF THE REACTOR COOLANT PRESSURE BOUNDARY

Inservice inspection, inservice testing, repair and replacement of pressure-retaining components, such as vessels, piping, pumps, valves, and bolting and supports within the reactor coolant pressure boundary, comply with Section XI of the ASME Code, including addenda, per 10 CFR 50.55a(f) for testing and 10 CFR 50.55a(g) for inspection, repair and replacement, with certain exceptions and alternatives whenever specific written relief is granted by the NRC per 10 CFR 50.55a, or when Section XI or OM Code Cases are used which either have been reviewed by the NRC and found acceptable as documented in 10CFR50.55a(b)(5) or (6) and Regulatory Guide 1.147 or 1.192 or approved for use by the granting of relief requests. The conditions for use of Regulatory Guide 1.147 or 1.192 approved Code Cases are discussed in Appendix 3A. The inservice testing of pumps and valves are discussed in Section 3.9(B).6. The limitations and modifications that the NRC places on the ASME Code in paragraph (b) of 10 CFR 50.55a are adhered to.

In addition, WCGS initially prepared separate preservice and inservice inspection program documents, which complied with "NRC Staff Guidance for Complying with Certain Provisions of 10CFR50.55a(g)--Inservice Inspection Requirements." A description of the preservice inspection program was submitted to the NRC by SNUPPS letter dated May 26, 1981. The initial inservice inspection program document was submitted to the NRC by letter dated December 11, 1985. Subsequent inservice inspection program documents are prepared in accordance with the 10 year update requirements in 10 CFR 50.55a and submitted to the NRC for initial approval. The inspection program documents identify the applicable Section XI edition and addenda and provide the details to the areas subject to examination, method of examination, extent and frequency of examination, and applicable Code Cases. 'Relief Requests' seeking relief from applicable code requirements are submitted to the NRC and become part of the inservice inspection program upon approval by the NRC. The repair and replacement program identifies the applicable Section XI edition and addenda, applicable Code Cases and relief requests, and provides the administrative controls for performing repairs and replacements.

Since the plant is required to meet the requirements of future editions of Section XI, insofar as practicable, an attempt was made during design to allow access for inspections and coverage's anticipated to be required by later editions of the Code. The result of this effort increased the areas on RPV available to mechanized inservice inspection. WCGS has attempted to create an inservice inspection program and plant design which concur with the 10 CFR 50 philosophy of upgrading inspections.

5.2.4.1 Inspection of Class 1 Components

The system boundary subject to inspection includes all piping and components in quality Group A (ASME Boiler and Pressure Vessel Code, Section III, Class 1). The reactor pressure vessel (RPV), pressurizer, Class 1 portion of the steam generators, and all Class 1 piping, pumps, and valves are examined except for items exempt from examination in accordance with ASME Section XI IWB-1200 and for those areas where relief has been requested and granted.

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The scope of examinations, inspections, and acceptance criteria for initial preservice inspections met the requirements outlined in Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1977 Edition up to and including the Summer 1978 Addenda. The scope of examinations, inspections, and acceptance criteria for inservice inspections and preservice inspections following repair and replacement meet the applicable Edition and Addenda of Section XI, as described at the beginning of section 5.2.4 and documented in the inservice inspection program. In addition, the RPV is examined in accordance with the recommendations of Regulatory Guide 1.150, Rev. 1 (Alternative Method), except for the components required to be examined to Appendix VIII. The ultrasonic examination of ferritic, austenitic, and dissimilar metal piping welds are performed in accordance with IWA-2232. The ultrasonic examination of cast austenitic stainless steel (centrifugal and static cast) piping and component welds may be performed in accordance with IWA-2240.

The extent of selection of piping welds for PSI examination were determined by the requirements of the 1974 Edition of Section XI with Addenda through Summer 1975. The extent of selection of piping welds for ISI examination is determined by the requirements of the applicable Edition and Addenda of ASME Section XI as described at the beginning of section 5.2.4 and documented in the inservice inspection program. Beginning in ISI interval 2, the selection of piping welds for examination is determined under a risk-informed ISI program as an NRC approved alternative to the Section XI requirements. This program is implemented under the 'Relief Request' process as described at the beginning of 5.2.4.

The Inservice Inspection Program requirements are specified in the Technical Requirements Manual.

5.2.4.2 Arrangement and Accessibility

5.2.4.2.1 General

Access for the purpose of inservice inspection is defined as the design of the plant with the proper clearances for examination personnel and/or equipment to perform inservice examinations during a nuclear unit shutdown. During system and component arrangement design, careful attention was given to physical clearances to allow personnel and equipment to perform required inservice examinations. Access requirements of the Code were considered in the design of components, weld joint configuration, and system arrangement. An inservice inspection program design review was undertaken to identify any exceptions to the access requirements of the code with subsequent design modifications and/or inspection technique development to ensure Code compliance, as required. Additional exceptions may be identified and reported to the NRC after plant operations, as specified in 10 CFR 50.55a(g) (5) (iv). Space has been provided to handle and store insulation, structural members, shielding, calibration blocks, and similar material related to the inspection. Suitable hoists and other handling equipment are also provided. Lighting, sources of power, and services for the inspection equipment are provided at appropriate locations.

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Access is provided for volumetric examination of the pressure-containing welds from the external surfaces of components and piping by means of removable insulation, removable shielding, and permanent tracks for remote inspection devices in areas where personnel access is restricted. Provisions for suitable access for inservice inspection examinations minimize the time required for these inspections to be performed. Therefore, they reduce the amount of radiation exposure to both plant and examination personnel. Working platforms have been provided at strategic locations in the plant to permit ready access to those areas of the reactor coolant pressure boundary which are designated as inspection points in the inservice inspection program. Areas without permanent platforms are provided with temporary platforms and/or scaffolding, as required.

5.2.4.2.2 Access to Reactor Pressure Vessel

Access for inspection of the RPV was provided as follows:

- a. Access to the exterior surface of the RPV below the 2,011-foot-6-inch cavity shelf elevation for inservice inspection is available since an annular space has been provided between the vessel exterior surface and the insulation interior surface. This was designed to permit the insertion of remotely operated inspection devices, if used, between the insulation and the reactor vessel. Examination personnel could enter the area below the RPV through one approximately 3-foot-square access port in the insulation to install the pole track remote examination device. The bottom head insulation is designed to allow an examiner to walk on the insulation while installing the examination device. Access to the window is provided through the in-core instrumentation tunnel. Use of the remotely operated external inspection devices was abandoned in favor of the standard industry approach of remotely operated internal inspection devices.
- b. A 3-foot annular space between the exterior surface of the RPV and the interior surface of the insulation has been provided from the vessel closure flange elevation to the cavity shelf elevation. The clearance area provides sufficient access for examination personnel and equipment to perform preservice and, if used, inservice examinations on the exterior surfaces of the nozzle-to-shell, safe end, pipe-to-elbow, flange-to-shell, and vertical welds in the upper shell course of the vessel. These welds may also be examined from the inside surface of the vessel using remotely operated inspection devices.
- c. The vessel flange seal surface is accessible during refueling outages when the closure head is removed. The vessel-to-flange weld can be examined manually or mechanically from the flange seal surface, using ultrasonic techniques. The inside surface of the RPV is

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available for a mechanized examination of the vessel-to-flange weld from the vessel side during refueling outages when the core barrel is removed. If examination of the vessel-to-flange weld from the vessel side is required when the core barrel has not been removed, the weld can be examined from the exterior surface of the vessel.

- d. Access to the inner surface of the RPV is available during refueling outages when the portions of vessel core structure are removed. A remotely operated examination device designed to perform ultrasonic examinations from the inner surface of the vessel is used to examine the vessel-to-flange weld, nozzle-to-shell welds, and the vertical, circumferential, and meridional welds of the vessel.

Selected areas of reactor cladding and the internal support attachments welded to the vessel wall are accessible for remote visual examination when the core barrel is removed at the end of the 10-year inspection interval. A camera capable of remote positioning can be inserted into the RPV.

- e. The closure head is dry stored during refueling, which facilitates direct manual examination. Removable insulation allows examination of the head welds from the outside surface. All reactor vessel studs which can be removed, nuts, and washers are removed to dry storage during refueling and are examined as required at that time. Studs which can not be removed are covered with a protective cover. Any stud that cannot be removed is cleaned and visually inspected, in-situ, to the extent possible, prior to placement in service for the next power operation cycle.

5.2.4.2.3 Pressurizer

The external surface is accessible for visual and volumetric inspection by removing the external insulation. Manways are provided to allow access for internal visual inspection. The permanent insulation around the pressurizer heaters is provided with a means to identify component leakages during system pressure testing as described in section 5.2.4.7.

5.2.4.2.4 Heat Exchangers and Steam Generators

The external surface is accessible for volumetric and visual inspection by removing portions of the vessel insulation. Manways in the steam generator channel head provide access for internal visual examinations and eddy current tests of steam generator tubes.

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5.2.4.2.5 Piping Pressure Boundary

The physical arrangement of piping, pumps, and valves has been designed to allow personnel access to welds requiring inservice inspection. Modifications to the initial plant design have been incorporated where practical to provide proper inspection access. Removable insulation has been provided where required by the Code on those piping systems requiring ultrasonic and/or surface examinations. In addition, the placement of pipe hangers and supports with respect to these welds has been reviewed and modified where necessary to reduce the amount of plant support required in these areas during inspection. Working platforms are provided in areas required to facilitate the servicing of pumps and valves.

Temporary or permanent platforms and ladders are provided, as necessary, to gain access to piping welds. A conscientious effort has been made to minimize the number of fitting-to-fitting welds within the inspection boundary. Welds requiring inspection have been located to permit ultrasonic examinations from at least one side, but, where component geometries permit, access from both sides of the weld is provided. The surfaces of the welds requiring ultrasonic examination by the Code have been prepared to permit effective examination. Vertical runs of piping are provided with removable insulation or catch basins at the low point for leakage surveillance during system pressure testing as described in section 5.2.4.7.

5.2.4.2.6 Pump Pressure Boundaries

The internal pressure-retaining surfaces of the pumps are accessible for visual inspection by removing the pump internals. External surfaces of the pump casing are accessible for visual and volumetric examination by removing component insulation. Internal examinations, when required by ASME Section XI, are performed when the pumps are disassembled for maintenance purposes.

5.2.4.2.7 Valve Pressure Boundaries

Class 1 valves over 4-inch nominal size are accessible for disassembly for visual examination of internal pressure boundary surfaces.

5.2.4.3 Examination Techniques and Procedures

Techniques and procedures, including any special technique and procedure for visual, surface, and volumetric examinations were written in accordance with the requirements of Subarticle IWA-2200 and Table IWB-2500-1 of Section XI of the ASME Code, applicable year and addenda. The liquid penetrant, magnetic particle or eddy current methods are utilized for surface examinations, radiographic (RT), and/or ultrasonic (UT) methods (either automated or manual) for volumetric examinations.

5.2.4.3.1 Equipment for Inservice Inspection

Procedures governing the use of the following examination devices are qualified prior to examinations in the plant.

5.2.4.3.1.1 Ultrasonic Equipment

Although the SNUPPS design provided for remotely operated external inspection equipment for examination of the reactor pressure vessel, such external equipment was abandoned in favor of the standard industry approach of remotely operated internal inspection equipment. The remotely operated device for examination of the vessel and connected piping from their inner surfaces is attached to the RPV at the flange surface. The device is capable of moving the transducers over the surface of the components in any direction.

An electronic system with a receiver or data channel for each ultrasonic transducer is used for acquiring and storing data when using remote automated examination equipment. Reflected signals may be transmitted through an ultrasonic instrument, gated, and multiplexed to initiate a digital recording. Scanning position is indicated by encoders and subsequently logged by the data acquisition system. The key parameters of each reflector recorded include location, maximum signal amplitude, depth below the scanning surface, and length of reflector. However, similar or compatible systems of data acquisition may be utilized.

5.2.4.3.1.2 Surface Examination Equipment

Mechanized surface examination techniques provide results which are at least equivalent to those obtainable by manual surface techniques.

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5.2.4.3.1.3 Visual Examination Equipment

Remote visual examination techniques will be in accordance with ASME Section XI requirements.

5.2.4.3.2 Coordination of Inspection Equipment with Access Provisions

Access to areas of the plant requiring inservice inspection is provided to allow the use of existing equipment, wherever practicable.

5.2.4.3.3 Manual Examination

In areas where manual ultrasonic examination is performed, reportable indications are mapped and records made of maximum signal amplitude, depth below the scanning surface, and length of the reflector. The data compilation format is such as to provide for comparison of data from subsequent examinations. Radiographic techniques may be used where ultrasonic techniques are not applicable. In areas where manual surface or direct visual examinations are performed, reportable indications are mapped with respect to size and location in a manner to allow comparison of data from subsequent examinations.

5.2.4.4 Inspection Intervals

The inspection interval, as defined in Subarticle IWA-2400 of Section XI, is a 10 year interval of service. These inspection intervals represent calendar years after the reactor facility has been placed into commercial service. The interval may be extended by as much as one year to permit inspections to be concurrent with plant outages. The inspection schedule is in accordance with IWB-2400. Inservice examinations are performed during normal plant outages, such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval. However, inservice examinations may be performed while the unit is on-line, if radiological and operational conditions permit access to the components. No examinations are performed which require draining of the reactor vessel further than just below the nozzles or removal of the core solely for the purpose of accomplishing the examinations.

5.2.4.5 Examination Categories and Requirements

The extent of the examinations performed and the examination methods utilized shall be in accordance with the applicable Edition and Addenda of Section XI, as described at the beginning of section 5.2.4 and documented in the inservice inspection program.

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In addition, preservice inspections comply with IWB-2200.

5.2.4.6 Evaluation of Examination Results

Evaluation of examination results for Class 1 components preservice inspections were conducted in accordance with the requirements of Article IWB-3000 of the ASME Code, Section XI, 1977 Edition with Addenda through the Summer of 1978. Evaluation of examination results for Class 1 inservice inspections are conducted in accordance with IWB-3000 in the applicable Edition and Addenda of Section XI, as described at the beginning of section 5.2.4 and documented in the inservice inspection program. In addition, the recording and evaluation of examinations results for reactor pressure vessel (RPV) are done as per Regulatory Guide 1.150, except for the components required to be examined to Appendix VIII.

5.2.4.7 System Leakage and Hydrostatic Tests

System pressure tests of the reactor pressure vessel and reactor coolant pressure boundary are conducted in accordance with the requirements of Articles IWA-5000 and IWB-5000. System leakage tests are conducted prior to startup following each reactor refueling outage, in accordance with Paragraph IWB-5221, as required by Article IWB-5000. The system leakage test performed during Inspection Period 3 at or near the end of each 10-year interval is in accordance with the provisions of ASME Code, Section XI, or approved ASME Code Cases, as documented in the ISI program plan.

5.2.5 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS

5.2.5.1 Design Bases

5.2.5.1.1 Safety Design Bases

There is no safety design basis for the reactor coolant pressure boundary leakage detection system.

5.2.5.1.2 Power Generation Design Bases

POWER GENERATION DESIGN BASIS ONE - For leaks of 1 gpm or greater, other than identified leakage sources, the reactor coolant boundary leakage detection systems are designed to detect leaks and determine the leakage rate (in accordance with Regulatory Guide 1.45 and 10 CFR 50, Appendix A, General Design Criterion 30). A comparison with the Regulatory Guide requirements is provided in Table 5.2-6.

POWER GENERATION DESIGN BASIS TWO - The leakage detection equipment is designed to continuously monitor the environmental conditions within the containment so that a background level is identified which is indicative of the normal level of leakage from

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primary systems and components. Significant upward deviation from normal containment environmental conditions provides positive indication in the control room of increases in leakage rates.

5.2.5.2 System Description

5.2.5.2.1 General Description

IDENTIFIED LEAKAGE DETECTION - Certain components of the reactor coolant pressure boundary may have small amounts of leakage and cannot, from a practical standpoint, be made leaktight. These identified sources of leakage are piped to the reactor coolant drain tank whose level is indicated and alarmed in the control room. The annular gap between the O-rings in the reactor vessel head flange is tapped and piped to a temperature indicator and then to the reactor coolant drain tank. Reactor coolant leakage gives a high temperature indication and alarm. Additionally, the controlled leakage shaft seal system for the reactor coolant pumps is monitored by reactor coolant drain tank level indication and alarm.

UNIDENTIFIED LEAKAGE DETECTION - The reactor coolant pressure boundary leakage detection system consists of the sump level and flow monitoring system, the containment air particulate monitoring system, the containment cooler condensate measuring system, and the containment humidity monitoring system. The sump level and flow monitoring system indicates leakage by monitoring increases in sump level. The containment cooler condensate measuring system and the containment humidity measuring system detect leakage from the release of steam or water to the containment atmosphere. The air particulate gas monitoring system detects leakage from the release of radioactive materials to the containment atmosphere. The containment gaseous radioactivity monitor could provide additional indication of leakage if significant reactor coolant gaseous activity is present from fuel cladding defects.

Primary-to-secondary reactor coolant leakage, if it occurs, is detected by the following radioactivity monitors: the main condenser evacuation, the steam generator liquid, the steam generator blowdown processing, and the steam generator blowdown discharge (Section 11.5.2).

Reactor coolant pressure boundary leakage is also indicated by increasing charging pump flow rate compared with reactor coolant system inventory changes and by unscheduled increases in reactor makeup water usage.

INTERSYSTEM LEAKAGE - Leakage to any significant degree into the auxiliary systems connected to the RCPB is not expected to occur. Design and administrative provisions which serve to limit leakage

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include isolation valves designed for low seat leakage, periodic testing of RCPB check valves (see Section 6.3.4.2), and inservice inspection (see Section 6.6). Leakage is detected by increasing the auxiliary system level, temperature, and pressure indications or lifting of the relief valves accompanied by increasing values of monitored parameters in the relief valve discharge path. These systems are isolated from the RCS by normally closed valves and/or check valves.

- a. Residual Heat Removal System (Suction Side) - The RHR system is isolated from the RCS on the suction side by motor-operated valves 8701A/B and 8702A/B. Leakage past these valves is detected by lifting of relief valves 8708A or 8708B, accompanied by increasing pressurizer relief tank level, pressure, and temperature indications and alarms on the main control board.
- b. Safety Injection System/Accumulators - The accumulators are isolated from the RCS by check valves 8948A/B/C/D and 8956A/B/C/D. Leakage, past these valves and into the accumulator subsystem, is detected by redundant control room accumulator pressure and level indications and alarms.
- c. Safety Injection System/RHR Discharge Subsystem - The RHR pump portion of the safety injection system is isolated from the RCS by check valves 8948A/B/C/D, 8818A/B/C/D, 8949B/C, 8841A/B, and normally closed motor-operated valve 8840. Leakage past these valves eventually pressurizes the RHR discharge header and result in lifting of the relief valves 8856A and 8856B or 8842. Relief valve lifting is detected by increasing levels of boron recycle holdup tanks which indicate and alarm in the radwaste control room and provide a general system alarm in the main control room.
- d. Safety Injection System/SI Pump Subsystem - The safety injection pump portion of the safety injection system is isolated from the RCS by check valves 8948A/B/C/D; EP-V010, V020, V030, V040; 8949A/B/C/D; EM-V001, V002, V003, V004; and normally closed motor-operated valves 8802A/B. Leakage past these valves pressurizes the safety injection pump discharge header, resulting in control room indication of increasing pressure and eventually lifting of relief valve 8851 or 8853A/B. Relief valve lifting is detected by increasing levels of boron recycle holdup tanks which indicate and alarm in the radwaste control room and provide a general system alarm in the main control room.

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- e. Safety Injection System/Charging Pump Subsystem - The charging pump subsystem is isolated from the RCS by check valves BB-V001, V022, V040, V059; and EM-8815; and motor-operated valves EM-8801A/B. Leakage past these valves eventually pressurizes the boron injection tank, resulting in a control room indication of increasing tank pressure. The BIT and associated piping form a closed volume which is designed for charging flow pressure. Lower pressure portions of the SIS are protected by double valve isolation, while single valves isolate the higher pressure charging flow piping. Leakage past valves EM-V151, V246, and V247 is not possible, since the inlet of each of these valves is pressurized by the operating charging pump.
- f. Waste Processing System - The waste processing system is isolated from the RCS by manual valves BB-V008, V028, V047, V066 and BB-V009, V029, V048, V067. Leakage past these valves results in increasing the control room indication of reactor coolant drain tank level and reactor coolant drain tank pump flow.
- g. Head Gasket Monitoring Connections - Leakage past the reactor vessel head gasket(s) result in temperature indication and alarm in the control room.
- h. Component Cooling Water - Leakage from the reactor coolant system to the component cooling water system, which services all components of the reactor coolant pressure boundary that require cooling, is detected by the component cooling water radioactivity monitoring system and/or increasing surge tank level. (Section 11.5.2).

Leakage to the containment atmosphere from the reactor coolant pressure boundary would cause a change in the containment airborne radioactivity which would be detected by the air particulate monitors. If the reactor is operating with a known rate of leakage, at a constant power level, with a constant reactor coolant activity and a constant purge rate, both the gross particulate and gross noble gas activities reach an equilibrium level. Under these conditions, an abnormal increase in monitored activity are the results of increased leakage. Such leakage is classified as unidentified until its source is determined.

During the expected modes of operation, the reactor coolant activity level fluctuates due to power variations and variations in letdown flow rate. However, significant increases in leakage can be detected.

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Leakage detection systems have been designed to aid operating personnel, to the extent possible, in differentiating between possible sources of detected leakage within the containment and identifying the physical location of the leak.

The containment atmosphere particulate monitoring system provides the primary means of remotely determining the presence of reactor coolant leakage within the containment. Increases in containment airborne activity levels detected by either of the monitors indicate the reactor coolant pressure boundary as the source of leakage. Conversely, if the humidity detector or condensate measuring system detects increased containment moisture without a corresponding increase in airborne activity level, the indicated source of leakage would be judged to be a non-radioactive system, except during times when reactor coolant activity may be low.

Less sensitive methods of leakage detection, such as unexplained increases in reactor plant makeup requirements to maintain pressurizer level, also provide indication of the reactor coolant pressure boundary as a potential leakage source. Increases in the frequency of a particular containment sump pump operation or increases in the level in a particular sump facilitate localization of the source to components whose leakage would drain to that sump. Leakage rates of the magnitude necessary to be detectable by these latter methods are expected to be noted first by the more sensitive radiation and moisture detection equipment.

Normally, unidentified leakage from the reactor coolant pressure boundary is essentially zero. The reactor coolant system is an all welded system, with the exception of the connections on the pressurizer safety valves, reactor vessel head, the pressurizer and steam generator manways, which are flanged, and encapsulation clamps at the capped flange on CRDM penetrations 10, 13, 17, 20, 22, 24, 25, 27, 28, 29. In addition, encapsulation clamps are authorized to be installed on any of the remaining CRDM penetrations. Connections to the reactor coolant system are welded. Isolation or check valves between the reactor coolant system and other systems have been designed for low seat leakage, and reactor coolant pressure boundary check valve backleakage is checked periodically. In general, valves in the reactor coolant system 2 inches and under are of the packless type. Valves larger than 2 inches have graphite packing.

The plant containment has the capability for a continuous purge of 4,000 cfm. The time to recirculate one containment free air volume through the containment air coolers is 4.57 minutes. The component operation for various leak detection systems, as discussed in Section 5.2.5.2.3, is based on this containment purge and recirculation time.

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MAXIMUM ALLOWABLE TOTAL LEAKAGE - The limits for the reactor coolant pressure boundary leakage are: identified, 10 gpm and unidentified, 1 gpm. When leakage is identified, it is evaluated by the operating staff to determine if operation can safely continue. Under these conditions, an allowable total leakage from known sources of 10 gpm has been established. Continued operation of the reactor with identified or unidentified leakage shall be in accordance with the Technical Specifications.

Normal chemical and volume control system operation can consist of either 75 gpm or 120 gpm letdown. This is determined by either operator preference or plant conditions. For example, 120 gpm letdown would normally be employed during periods of increased RCS activity. An additional 12 gpm reactor coolant pump seal return during normal plant operation results in a total flow leaving the reactor of either 87 gpm (75 gpm letdown) or 132 gpm (120 gpm letdown). Based on the above conditions, the charging pump flow rates of 87 gpm or 132 gpm would be required to makeup for flow leaving the reactor. Considering a normal seal injection flow of 32 gpm; 55 gpm (75 gpm letdown) or 100 gpm (120 gpm letdown) would be supplied through the normal charging line. A single centrifugal charging pump with a 150 gpm rated capacity at 5800 ft of head or the Normal Charging Pump which has a capability of 150 gpm as shown in the preoperational test provides an adequate reserve capacity at normal RCS pressures to easily accommodate a 10 gpm maximum limit on reactor coolant pressure boundary leakage.

The reactor coolant pressure boundary leakage detection system provides ample protection to assure that, in the unlikely event of a failure of the reactor coolant pressure boundary, small cracks are detected prior to becoming large leaks. In particular:

- a. The sensitivity of the detection equipment is such that leaks can be identified when small, and the plant can be shut down. The limit on continued operation for unidentified leakage is 1 gpm. This is well within the detection capability of the reactor coolant pressure boundary leakage detection system.
- b. The time span for a crack to go from detectable size to critical size varies from 5 to more than 40 years. This assures adequate safety from a major loss-of-coolant accident. Actual conditions are addressed in Reference 9.

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The above methods are supplemented by visual and ultrasonic inspections of the reactor coolant pressure boundary during plant shutdown periods, in accordance with the inservice inspection program (Section 5.2.4).

5.2.5.2.2 Component Description

CONTAINMENT AIR PARTICULATE MONITOR - An air sample is drawn outside the containment into a closed system by a sample pump and is then consecutively passed through a particulate filter with detectors, an iodine filter with detector, and a gaseous monitor chamber with detector. The sample transport system includes:

- a. A pump to obtain the air sample
- b. A flow control valve to provide flow adjustment
- c. A flow meter to indicate the flow rate
- d. A flow alarm assembly to provide high and low flow alarm signals

The particulate filter is continuously monitored by a scintillation crystal with a photo multiplier tube which provides an output signal proportional to the activity collected on the filter. The particulate monitor has a range of 10^{-12} to 10^{-7} $\mu\text{Ci}/\text{cc}$ and a minimum detectable concentration of 10^{-11} $\mu\text{Ci}/\text{cc}$. The containment and particulate monitoring system is capable of performing its radioactive monitoring functions following an SSE. More details concerning the particulate monitors can be found in Section 11.5.2.3.2.2.

CONTAINMENT GASEOUS RADIOACTIVITY MONITOR - The containment gaseous radioactivity monitor determines gaseous radioactivity in the containment by monitoring continuous air samples from the containment atmosphere. After passing through the gas monitor, the sample is returned via the closed system to the containment atmosphere.

Each sample is continuously mixed in a fixed, shielded volume where its activity is monitored. The monitor has a range of 10^{-7} to 10^{-2} $\mu\text{Ci}/\text{cc}$ and a minimum detectable concentration of 2×10^{-7} $\mu\text{Ci}/\text{cc}$.

The containment gaseous radioactivity monitors are fully described in Section 11.5.2.3.2.2.

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The containment gaseous radioactivity monitoring system is capable of performing its radioactivity monitoring functions following an SSE.

CONTAINMENT PURGE MONITORS - The containment purge system radioactivity monitors (Section 11.5.2.3.2.3) serve as a backup to the containment air particulate and gaseous airborne radioactivity monitoring system while the purge is in operation.

CONTAINMENT COOLER CONDENSATE MONITORING SYSTEM - The condensate monitoring system permits measurements of the liquid runoff from the containment cooler units. It consists of a containment cooler drain collection header, a vertical standpipe, valving, and standpipe level instrumentation for each cooler.

The condensation from the containment coolers flows via the collection header to the vertical standpipe. A differential pressure transmitter provides standpipe level signals. The system provides measurements of low leakages by monitoring standpipe level increase versus time.

CONTAINMENT HUMIDITY MONITORING SYSTEM - The containment humidity monitoring system, utilizing temperature compensated humidity detectors, is provided to determine the water vapor content of the containment atmosphere.

An increase in the humidity of the containment atmosphere indicates release of water within the containment. The range of the containment humidity measuring system is 10 to 98-percent relative humidity at 80°F with a temperature range of 40 to 120°F.

CONTAINMENT SUMP LEVEL AND FLOW MONITORING SYSTEM - Since a leak in the primary system would result in reactor coolant flowing into the containment normal or instrument tunnel sumps, leakage would be indicated by a level increase in the sumps. Indication of increasing sump level is transmitted from the sump to the control room level indicator by means of a sump level transmitter. The system provides measurements of low leakages by monitoring level increase versus time.

CHARGING PUMP OPERATION - During normal operation, either the normal or other centrifugal charging pump is in operation. If a gross loss of reactor coolant occurs which is not detected by the methods previously described, the flow rate of the operating charging pump indicates the leakage from the reactor coolant system. This leakage must be sufficient to cause a decrease in pressurizer or volume control

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tank level that is within the sensitivity range of the level indicators. The charging pump flow would automatically increase to try to maintain pressurizer level. Charging pump discharge flow indication is provided in the control room.

SUMP PUMP OPERATION - Since a leak in the primary system may result in reactor coolant flowing into the containment normal or instrument tunnel sumps, gross leakage can be indicated by an increase in the frequency of operation of the containment normal or the containment instrument tunnel sump pumps. Pump operation can be monitored from the control room.

LIQUID INVENTORY - Larger leaks may also be detected by unscheduled increases in the amount of reactor coolant makeup water which is required to maintain the normal level in the pressurizer. Pressurizer level can be monitored in the control room. Total makeup water flow is also available from the plant computer.

5.2.5.2.3 Component Operation

CONTAINMENT AIR PARTICULATE MONITOR - Particulate activity is determined from the containment free volume and the coolant fission and corrosion product particulate activity concentrations. Any increase of more than two standard deviations above the count rate for background would indicate a possible leak. The total particulate activity concentration above background, due to an abnormal leak and natural decay, increases almost linearly with time for the first several hours after the beginning of a leak. As shown in Figure 5.2-2, with 0.1-percent failed fuel, containment background airborne particulate radioactivity equivalent to 10⁻⁴ percent/day, and a partition factor equal to 0.01 (NUREG-0017 assumptions), a 1-gpm leak would be detected in 1 hour. Larger leaks would be detected in proportionately shorter times (exclusive of sample transport time, which remains constant). The detection capabilities and response times are shown on Figure 5.2-2.

In the discussions with the NRC and in NUREG/CR-6582, the gaseous particulate monitors cannot readily determine the leakage rate because the activity is determined by unsteady conditions, background level, reactor coolant activity and partition factors for particulates. The background activity is dependent upon the power level, percent failed fuel, crud bursts, iodine spiking, and natural radioactivity brought in by the containment purge.

CONTAINMENT GASEOUS RADIOACTIVITY MONITOR - This monitor is less sensitive than the containment air particulate monitors but gives a positive indication of leakage in the event that reactor coolant gaseous activity exists as a result of fuel-cladding defects. Gaseous radioactivity is determined from the containment free volume and the gaseous activity concentration of the reactor coolant. Any increase more than two standard deviations above the count rate for background would indicate a possible leak. The total gaseous activity level above background (after 1 year of normal operation) increases

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almost linearly for the first several hours after the beginning of the leak. As specified in Figure 5.2-2, with 0.1-percent failed fuel, containment background airborne gaseous radioactivity equivalent to 1 percent/day, and a partition factor equal to 1 (NUREG-0017 assumptions), a 1-gpm leak would be detected within 1 hour. Larger leaks would be detected in proportionately shorter times (exclusive of the sample transport time which remains constant). The detection capabilities and response times are shown on Figure 5.2-2.

Evaluations have shown that the pre-existing containment radioactive gaseous background levels for which reliable detection is possible is dependent upon the reactor power level, percent failed fuel and natural radioactivity brought in by the containment purge. With primary coolant concentrations less than equilibrium levels, such as during reactor startup and operation with no fuel defects, the increase in detector count rate due to leakage will be partially masked by 1) the statistical variation of the minimum detector background count rate, and 2) the Ar-41 activation activity rendering reliable detection of a 1 gpm leak uncertain. The containment atmosphere gaseous radioactivity monitors were designed in accordance with the sensitivities specified in Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," with the alarm setpoint set to indicate a 1 gpm RCS leak based on Regulatory Guide 1.45 assumptions. The monitors are fully functioning in accordance with its design requirements, however they have been removed as part of the reactor coolant pressure boundary leakage detection system due to the inability to promptly detect a 1 gpm RCS leak within 1 hour with reduced radioactivity levels in the reactor coolant system. (Reference 12)

CONTAINMENT PURGE MONITORS - The containment purge monitors function the same as the containment air particulate and gaseous radioactivity monitors, except that the purge monitors sample from the containment purge exhaust line.

CONTAINMENT COOLER CONDENSATE MONITORING SYSTEM - The condensate flow rate is a function of containment humidity, essential service water temperature leaving the coolers, and containment purge rate. The water vapor dispersed by a 1 gpm leak is much greater than the water vapor brought in with the outside air. Air brought in from the outside is heated to 50°F before it enters the containment.

After the air enters the containment, it is heated to 100-120°F so that the relative humidity drops. The water vapor brought in with the outside air does not build up in the containment since it is continually purged. The most important factor in condensing the water vapor is the temperature of the essential service water which is provided to the containment coolers. This water can vary between 38 - 100°F on the outlet of the coolers, depending on seasonal conditions.

Level changes of as little as 0.25 inches in the cooler condensate standpipes can be detected. Increases in the condensation rates over normal background are monitored by the plant computer based upon level checks in order to determine the unidentified leakage. Figure 5.2-2 shows the detection capabilities of the system for various seasonal conditions with no airborne identified leakage. Normal background leakage will increase containment humidity to the point where condensation will more readily occur and, thereby, will improve the detection capabilities of this system.

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As shown on Figure 5.2-2, a sensitivity of 1 gpm in 1 hour can be achieved with cold essential service water temperature to the containment coolers or with initial background leakage.

The rate of leakage can be determined when the precise essential service water, outside air, and containment air temperatures and the outside relative humidity are known by use of psychrometric charts.

CONTAINMENT HUMIDITY MONITORING SYSTEM - The maximum possible containment humidity under various outside air conditions and no leakage will fall within the extremes shown on Figure 5.2-2. Therefore, a 1-gpm leak can be detected within 1 hour by measuring the containment humidity.

The accuracy of the humidity detectors is ± 3 percent. A rapid increase of humidity over the background level by more than 10 percent can be taken as a probable indication of a leak.

The leak rate can be determined when the outside air temperature and humidity and the containment temperature are known by use of psychrometric charts.

CONTAINMENT SUMP LEVEL AND FLOW MONITORING SYSTEM - The detection capabilities of the containment normal sump and instrument tunnel sump are shown in Figure 5.2-2, assuming that the water from the leak is collected in the sump.

The minimum detectable change in the containment normal sump level is 3 gallons and in the instrument tunnel sump level is 15 gallons.

The actual reactor coolant leakage rate can be established from the increase above the normal rate of change of sump level after consideration of 35 percent of the high temperature leakage which initially evaporates but may be condensed by the containment coolers and then is routed to the sump. A check of other instrumentation would be required to eliminate possible leakage from nonradioactive systems as a cause of an increase in sump level.

CHARGING PUMP OPERATION - The normal charging pump normally delivers 87 or 132 gpm to the reactor coolant system depending on the amount of letdown flow established. Any significant increase in the flow rate is a possible indication of a leak.

The leakage rate can be determined by the amount that the charging pump rate increases above 87 or 132 gpm to maintain constant pressurizer level.

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

The leakage rate can be determined from sump volumes and frequency of sump pump operation.

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

5.2.5.3 Safety Evaluation

Inasmuch as this system has no safety design basis, no safety evaluation is provided. Criteria for the selection of safety design bases are stated in Section 1.1.7.

5.2.5.4 Tests and Inspections

Periodic testing of leakage detection systems is conducted to verify the operability and sensitivity of detector equipment. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks. A description of calibration and maintenance procedures and frequencies for the containment radioactivity monitoring system is presented in Section 11.5.2.

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

Inservice inspection criteria, the equipment used, procedures involved, the frequency of testing, inspection, surveillance, and examination of the structural and leaktight integrity of reactor coolant pressure boundary components are described in detail in Section 5.2.4.

5.2.5.5 Instrumentation Applications

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

- a. Containment air particulate monitor - air particulate activity

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- b. Containment gaseous activity monitor - gaseous activity
- c. Containment cooler condensate monitoring system - standpipe level
- d. Containment normal sump level and instrument tunnel sump level
- e. Containment humidity measuring system - containment humidity
- f. Gross leakage detection methods - Charging pump flow rate, let-down flow rate, pressurizer level and reactor coolant temperatures are available for the charging pump flow method. Containment sump levels and pump operation are available for the sump pump operation method. Totalized makeup water flow is available from the plant computer for liquid inventory.

5.2.6 REFERENCES

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7. Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.
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11. NRC Letter dated May 31, 2005, from Robert A. Gramm to Rick A. Muench "Wolf Creek Generating Station - Request for Relief Regarding Classification of Pressurizer Upper level Instrument and Other Lines and Associated components for Wolf Creek Generating Station, Unit 1 (TAC No. MC5058)."

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12. NRC Letter dated May 16, 2006, from J. Donohew to R. Muench, "Wolf Creek Generation Station - License Amendment Request to change the Reactor Coolant System Leakage Detection Instrumentation Methodology (TAC No. MC8214).
13. Implementation of piping code cases in specification M-200. |
14. Letter 07-00401, dated July 19, 2007, from USNRC to WCNO, Authorization of Relief Request 13R-05, Alternatives to Structural Weld Overlay Requirements. |

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TABLE 5.2-1

APPLICABLE CODE ADDENDA FOR REACTOR COOLANT SYSTEM COMPONENTS

Reactor vessel	ASME III, 1971 Edition through Winter 1972
Steam generator	ASME III, 1971 Edition through Summer 1973
Pressurizer	ASME III, 1974 Edition
CRDM housing	ASME III, 1974 Edition through Winter 1974
CRDM head adapter	ASME III, 1971 Edition through Winter 1972
Reactor coolant pump	ASME III, 1971 Edition through Summer 1973
* Reactor coolant pipe	ASME III, 1974 Edition through Winter 1975
** Surge lines	ASME III, 1986 Edition
Valves	
Pressurizer safety	ASME III, 1974 Edition through Summer 1975
Motor operated	ASME III, 1974 Edition through Summer 1975
Manual (3 inch and larger)	ASME III, 1974 Edition through Summer 1975
Control	ASME III, 1974 Edition through Summer 1975

* The 1974 Edition and Addenda up to and including the Winter 1975 Addenda is the applicable version of the Code for Class 1 piping components designed / supplied by Westinghouse. In addition, the fatigue stress analysis uses the ASME Code Addend up to Summer 1979.

** The Class 1 piping fatigue stress analysis uses ASME Section III 1986 code.

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TABLE 5.2-2

CLASS 1 PRIMARY COMPONENTS
MATERIAL SPECIFICATIONS

Reactor Vessel Components

Shell and head plates (other than core region)	SA-533, Grade A, B or C, Class 1 or 2 (vacuum treated)
Shell plates (core region)	SA-533, Grade A or B, Class 1 (vacuum treated)
Shell, flange and nozzle forgings, nozzle safe ends	SA-508, Class 2 or 3; SA-182, Grade F304 or F316
CRDM and/or ECCS appurtenances, upper head	SB-166 or SB-167 and SA-182, Grade F304
Instrumentation tube appurtenances, lower head	SB-166 or SB-167 and SA-182, Grade F304, F304L or F316
Closure studs, nuts, washers, inserts, and adaptors	SA-540, Class 3, Grade B23 or B24 (as modified by Code Case 1605)
Core support pads	SB-166 with carbon less than 0.10 percent
Monitor tubes and vent pipe	SA-312 or SA-376, Grade TP304 or TP316 or SB-166 or SB-167 or SA-182, Grade F316
Vessel supports, seal ledge, and heat lifting lugs	SA-516, Grade 70 (quenched and tempered) or SA-533, Grade A, B or C, Class 1 or 2 (vessel supports may be of weld metal buildup of equivalent strength of the nozzle material)
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43

Steam Generator Components

Pressure Plates	SA-533, Grade A, Class 2
Pressure forgings (including nozzles and tube sheet)	SA-508, Class 2a

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TABLE 5.2-2 (Sheet 2)

Nozzle safe ends	Stainless Steel Weld Metal Analysis A-8
Channel heads	SA-533, Grade A, B or C, Class 1 or 2 or SA-216, Grade WCC
Tubes	SB-163 (Ni-Cr-Fe annealed)
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Closure bolting	SA-193, Grade B7
Pressurizer Components	
Pressure plates	SA-533, Grade A, Class 2
Pressure forgings *	SA-508, Class 2a
Nozzle safe ends *	SA-182, Grade F316L
Cladding and buttering *	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Closure bolting	SA-193, Grade B7
Reactor Coolant Pump	
Pressure forgings	SA-182, Grade F304, F316, F347 or F348
Pressure casting	SA-351, Grade CF8, CF8A or CF8M
Tube and pipe	SA-213; SA-376 or SA-312, Seam- less, Grade TP304 or TP316
Pressure plates	SA-240, Type 304 or 316
Bar material	SA-479, Type 304 or 316
Closure bolting	SA-193; SA-320; SA-540 or SA-453, Grade 660; SB-637 Gr. N07718
Flywheel	SA-533, Grade B, Class 1

* In order to mitigate primary water stress corrosion cracking concerns with the originally installed Alloy 600 (82/182) dissimilar metal welds, full structural weld overlays made of ERNiCrFe-7A (Alloy 52M/UNS N06054) have been installed to cover portions of the Pressurizer nozzles (Surge, Safety, Relief, and Spray), nozzle weld butter layers, dissimilar metal welds between the butter and the safe end, safe ends, safe end to stainless steel pipe welds, and connecting stainless steel piping.

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TABLE 5.2-2 (Sheet 3)

Reactor Coolant Piping	
Reactor coolant pipe	SA-351, Grade CF8A Centrifugal Casting
Reactor coolant fittings, branch nozzles	SA-351, Grade CF8A and SA-182, (Code Case 1423-2) Grade 316N
Surge line	SA-376, Grade TP304, TP316 or F304N
Auxiliary piping 1/2 through 12 inch 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
Full Length CRDM	
Latch housing	SA-182, Grade F304 or SA-351, Grade CF8
Rod travel housing	SA-182, Grade F304 or SA-336, Class F8
Cap	SA-479, Type 304
Welding materials	Stainless Steel Weld Metal Analysis A-8

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TABLE 5.2-3

CLASS 1 AND 2 AUXILIARY COMPONENTS
MATERIAL SPECIFICATIONS

Valves

Bodies	SA-182, Grade F316 or SA-351, Grade CF8 or CF8M
Bonnets	SA-182, Grade F316 or SA-351, Grade CF8 or CF8M
Discs	SA-182, Grade F316 or SA-564, Grade 630, or SA-351, Grade CF8 or CF8M
Stems	SA-182, Grade F316 or SA-564, Grade 630
Pressure-retaining bolting	SA-453, Grade 660
Pressure-retaining nuts	SA-453, Grade 660 or SA-194 Grade 6

Auxiliary Heat Exchangers

Heads	SA-240, Type 304
Nozzle necks	SA-182, Grade F304
Tubes	SA-213, Grade TP304
Tube Sheets	SA-182, Grade F304
Shells	SA-240 and SA-312, Grade TP304

Auxiliary Pressure Vessels, Tanks, Filters, etc.

Shells and heads	SA-240, Type 304 or SA-264 (consisting of SA-537, Class 1 with Stainless Steel Weld Meta Analysis A-8 Cladding)
Flanges and nozzles	SA-182, Grade F304 and SA-105 or SA-350, Grade LF2 or LF3 with Stainless Steel Weld Metal Analysis A-8 Cladding

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TABLE 5.2-3 (Sheet 2)

Piping	SA-312 and SA-240, Grade TP304 or TP316 Seamless
Pipe fittings	SA-403, Grade WP304 Seamless
Closure bolting and nuts	SA-193, Grade B7 and SA-194, Grade 2H/Grade 7
Auxiliary Pumps	
Pump casing and heads	SA-351, Grade CF8 or CF8M; SA-182, Grade F304 or F316
Flanges and nozzles	SA-182, Grade F304 or F316; SA-403, Grade WP316L Seamless
Piping	SA-312, Grade TP304 or TP316 Seamless
Stuffing or packing box cover	SA-351, Grade CF8 or CF8M; SA-240, Type 304 or 304L or 316
Pipe fittings	SA-403, Grade WP316L Seamless
Closure bolting and nuts	SA-193, Grade B6, B7 or B8M; SA-194, Grade 2H/Grade 7 or 8M; SA-453 Grade 660, and Nuts, SA-194, Grade 2H, 6 and 8 M

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TABLE 5.2-4

REACTOR VESSEL INTERNALS FOR EMERGENCY CORE COOLING SYSTEMS

Forgings	SA-182, Grade F304
Plates	SA-240, Type 304
Pipes	SA-312, Grade TP304 Seamless or SA-376, Grade TP304
Tubes	SA-213, Grade TP304
Bars	SA-479, Type 304 and 410
Castings	SA-351, Grade CF8 and CF8A
Bolting	SA-193, Grade B8M (65 MYS/90 MTS) Code Case 1618 Inconel-750; SA-461, Grade 688
Nuts	SA-193, Grade B8
Locking devices	SA-479, Type 304

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TABLE 5.2-5

RECOMMENDED REACTOR COOLANT WATER CHEMISTRY LIMITS (g)

Electrical conductivity	Determined by the concentration of boric acid and alkali present. Expected range is 1 to 40 μ mhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C. Values will be 5.0 or greater at normal operating temperatures.
Oxygen (a)	0.005 ppm, maximum
Chloride (b)	0.15 ppm, maximum
Fluoride (b)	0.15 ppm, maximum
Hydrogen (c)	25 to 50 cc (STP)/kg H ₂ O
Suspended solids (d)	1.0 ppm, maximum
pH control agent (Li7OH)	Lithium Control Program
Boric acid	Variable from 0 to ~4000 ppm as B
Silica (f)	1.0 ppm, maximum
Aluminum (f)	0.05 ppm, maximum
Calcium (f)	0.05 ppm, maximum
Magnesium (f)	0.05 ppm, maximum

NOTES:

- (a) Oxygen concentration should normally be controlled by scavenging with hydrazine to less than 0.1 ppm in the reactor coolant prior to exceeding a temperature of 250°F. During power operation with the specified hydrogen concentration maintained in the coolant, the residual oxygen concentration does not exceed 0.005 ppm.
- (b) Halogen concentrations are maintained below the specified values at all times regardless of system temperature.

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TABLE 5.2-5 (Sheet 2)

- (c) Hydrogen is maintained in the reactor coolant for all plant operations with nuclear power above 1 MWt. The normal operating range should be 30 to 40 cc/kg H₂O.
Twenty four hours prior to a scheduled shutdown, when the reactor coolant system is intended to be cooled down, the hydrogen concentration may be reduced below the normal operating range to facilitate degassification, but hydrogen levels of at least 15cc H₂/KgH₂O should be maintained.
- (d) Solids concentration determined by filtration through filter having 0.45 micron pore size.
- (e) Deleted
- (f) These limits are included in the table of reactor coolant specifications as recommended standards for monitoring coolant purity. Establishing coolant purity within the limits shown for these species is judged desirable with the current data base to minimize fuel clad crud deposition which affects the corrosion resistance and heat transfer of the clad.
- (g) Refer to the Technical Requirements Manual for required reactor coolant chemistry limits.

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TABLE 5.2-6

DESIGN COMPARISON WITH REGULATORY GUIDE 1.45, DATED MAY 1973, TITLED REACTOR
COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS

Regulatory Guide
1.45 Position

WCGS

C. REGULATORY POSTION

The source of reactor coolant leakage should be identifiable to the extent practical. Reactor coolant pressure boundary leakage detection and collection systems should be selected and designed to include the following:

1. Leakage to the primary reactor containment from identified sources should be collected or otherwise isolated so that:

- a. the flow rates are monitored separately from unidentified leakage, and
- b. the total flow rate can be established and monitored.

2. Leakage to the primary reactor containment from unidentified sources should be collected and the flow rate monitored with an accuracy of one gallon per minute (gpm) or better.

3. At least three separate detection methods should be employed and two of these methods should be (1) sump level and flow monitoring and

1. Complies. Flow to the RCDT can be established, is monitored, and is separated from unidentified leakage.

2. Complies. The instrumentation provided is such that over a period of time (1 hour or more), the collected flow rate can be determined with an accuracy of better than 1 gallon per minute.

3. Complies. The methods provided are sump-level and flow (level versus time) monitoring, airborne particulate

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TABLE 5.2-6 (Sheet 2)

Regulatory Guide
1.45 Position

WCGS

(2) airborne particulate radioactivity monitoring. The third method may be selected from the following:

- a. monitoring of condensate flow rate from air coolers,
- b. monitoring of airborne gaseous radioactivity.

Humidity, temperature, or pressure monitoring of the containment atmosphere should be considered as alarms or indirect indication of leakage to the containment.

4. Provisions should be made to monitor systems connected to the RCPB for signs of intersystem leakage. Methods should include radioactivity monitoring and indicators to show abnormal water levels or flow in the affected area.

5. The sensitivity and response time of each leakage detection system in regulatory position 3. above employed for unidentified leakage should be adequate to detect a leakage rate, or its equivalent, of one gpm in less than one hour.

6. The leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown. The airborne particulate radioactivity monitoring system should remain functional when subjected to the SSE.

Radioactivity monitoring, containment cooler condensate monitoring, and containment atmosphere humidity monitoring.

4. Complies. Refer to Sections 5.2.5.2.1, 9.3.3, and 11.5.

5. Complies, as described in Section 5.2.5.2.3 and as shown on Figure 5.2-2.

6. Complies. The airborne particulate radioactivity system is designed to remain functional when subjected to the SSE. Refer to Sections 11.5.2.3.2.2 and 11.5.2.3.2.3. The remaining leakage detection systems can reasonably be

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TABLE 5.2-6 (Sheet 3)

Regulatory Guide
1.45 Position

WCGS

7. Indicators and alarms for each leakage detection system should be provided in the main control room. Procedures for converting various indications to a common leakage equivalent should be available to the operators. The calibration of the indicators should account for needed independent variables.
8. The leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation.
9. The technical specifications should include the limiting conditions for identified and unidentified leakage and address the availability of various types of instruments to assure adequate coverage at all times.

expected to remain functional following seismic events of lesser severity than the SSE. However, no special qualification program is used to assure operability under such conditions.

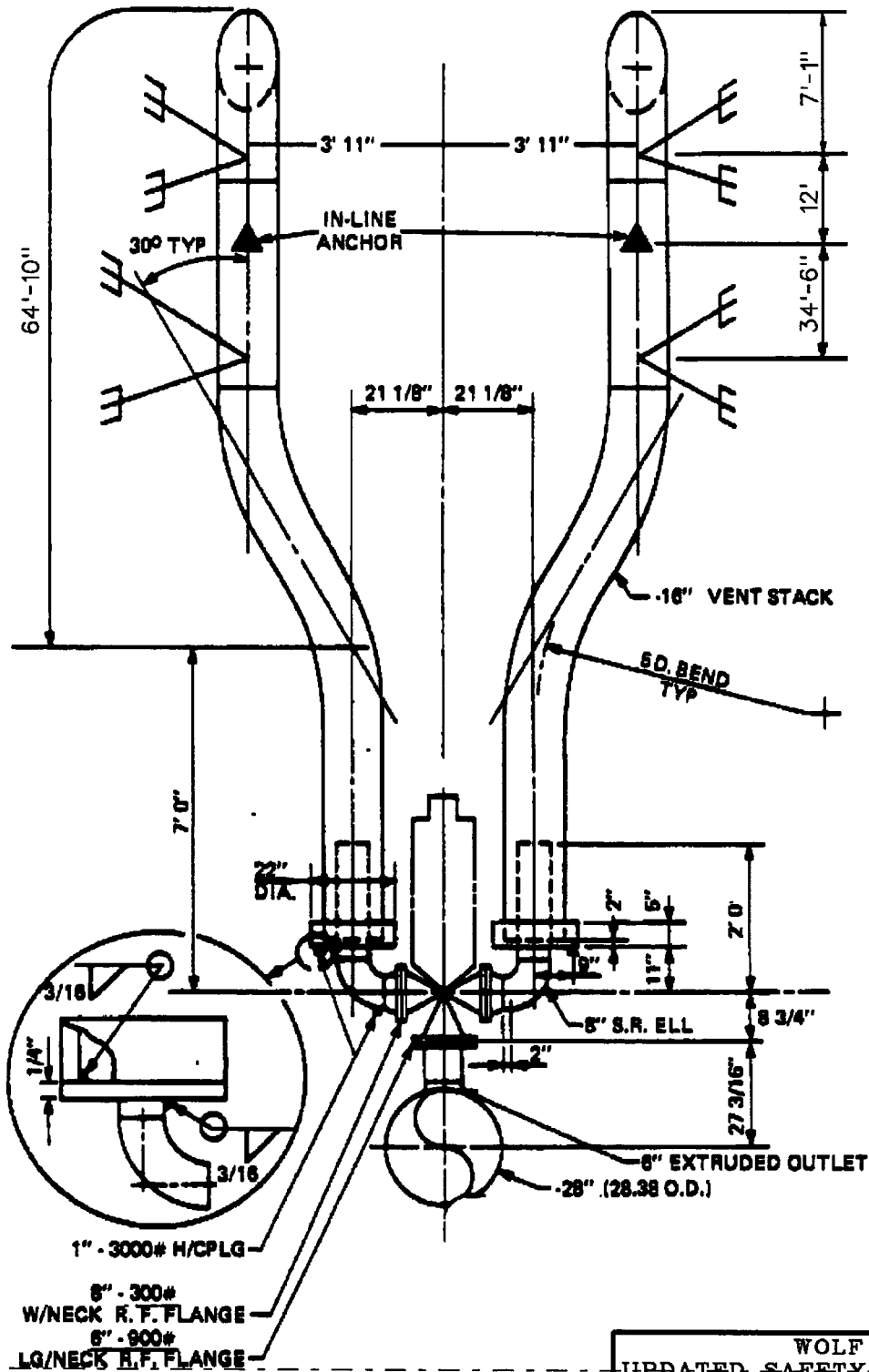
7. Complies, as described in Sections 5.2.5.2.3 and 5.2.5.5.
8. Complies. Refer to Section 5.2.5.4.
9. Complies. Refer to Technical Specifications. The Containment Atmosphere Particulate Radioactivity Monitor, Containment Sump Level and Flow Monitoring System, and the Containment Air Cooler Condensate Monitoring System are specified in the Limiting Conditions for operation to monitor and detect leakage from the reactor coolant pressure boundary.

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Table 5.2-7

This table has been deleted

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 UPDATED SAFETY ANALYSIS REPORT

Figure 5.2-1
 INSTALLATION DETAIL FOR THE MAIN
 STEAM PRESSURE RELIEF DEVICES

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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.

The ferritic materials of the reactor vessel beltline are restricted to the following maximum limits of copper, phosphorous, and vanadium to reduce sensitivity to irradiation embrittlement in service:

<u>Element</u>	<u>Base Metal (percent)</u>	<u>As Deposited Weld Metal (percent)</u>
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)

Figure 5.3-2 identifies the location of the beltline materials and welds for the WCGS reactor vessel. Table 5.3-7 contains weld identification information for these welds. Information concerning the fabrication and post-weld heat treatment of the surveillance test specimen weld is identified in WCAP-10015 for WCGS. The test weldment is fabricated as a separate weld, not as an extension of a longitudinal weld seam.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

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

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- b. The use of severely sensitized stainless steel as a pressure boundary material has been prohibited and has been eliminated by either choice of material or programming the method of assembly.
- c. The control rod drive mechanism head adapter threads and surfaces of the guide studs are chrome plated to prevent possible galling of the mated parts.
- d. At all locations in the reactor vessel where stainless steel and Inconel are joined, the final joining weld beads are Inconel weld metal in order to prevent cracking.
- e. The location of full penetration weld seams in the upper closure head and vessel bottom head are restricted to areas that permit accessibility during inservice inspection.
- f. The stainless steel clad surfaces are sampled to assure that material composition requirements are met.
- g. Freedom from underclad cracking is assured by special evaluation of the procedure qualification for cladding applied on low alloy steel (SA-508, Class 2).
- h. Minimum preheat requirements have been established for pressure boundary welds, using low alloy material. The preheat is maintained until either an intermediate or full post-weld heat treatment is completed or until the completion of welding.

5.3.1.3 Special Methods for Nondestructive Examination

The nondestructive examination of the reactor vessel and its appurtenances is conducted in accordance with ASME Code, Section III requirements; also numerous examinations are performed in addition to ASME Code, Section III requirements. Nondestructive examination of the vessel is discussed in the following paragraphs and the reactor vessel quality assurance program is given in Table 5.3-1.

5.3.1.3.1 Ultrasonic Examination

- a. In addition to the required ASME Code straight beam ultrasonic examination, angle beam inspection over 100 percent of one major surface of plate material is performed during fabrication to detect discontinuities that may be undetected by the straight beam examination.

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- b. In addition to the ASME Code, Section III nondestructive examination, all full penetration ferritic pressure boundary welds in the reactor vessel are ultrasonically examined during fabrication. This test was performed upon completion of the welding and intermediate heat treatment but prior to the final post-weld heat treatment.
- c. After hydrotesting, all full penetration ferritic pressure boundary welds in the reactor vessel, as well as the nozzle to safe end welds, are ultrasonically examined. These inspections are also performed in addition to the ASME Code, Section III nondestructive examinations.

5.3.1.3.2 Penetrant Examinations

The partial penetration welds for the control rod drive mechanism head adapters and the bottom instrumentation tubes were inspected by dye penetrant after the root pass, in addition to code requirements. Core support block attachment welds were inspected by dye penetrant after the first layer of weld metal and after each 1/2 inch of weld metal. All clad surfaces and other vessel and head internal surfaces were 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 were performed in accordance with the following:

- a. Prior to the final post-weld heat treatment - Only by the prod, coil, or direct contact method.
- b. After the final post-weld heat treatment - Only by the yoke method.

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

Surface Examinations

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

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- b. Magnetic particle examine all exterior closure stud surfaces and all nut surfaces after final machining or rolling. Continuous circular and longitudinal magnetization is used.
- c. Magnetic particle examine all inside diameter surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. This inspection is performed after forming and machining (if performed) and prior to cladding.

Weld Examination

Magnetic particle examination of the weld metal build-up for vessel support welds, the closure head lifting lugs, and the refueling seal ledge to the reactor vessel after the first layer and each 1/2 inch of weld metal is deposited. All pressure boundary welds are 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. Section 5.2.3 includes discussions which indicate the degree of conformance with Regulatory Guide 1.44. Appendix 3A discusses the degree of conformance with Regulatory Guides 1.43, 1.50, 1.71, and 1.99.

5.3.1.5 Fracture Toughness

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

The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline (including welds) are 75 foot-pounds, as required per Appendix G of 10 CFR 50. Materials having a section thickness greater than 10 inches with an upper shelf of less than 75 foot-pounds are evaluated with regard to effects of chemistry (especially copper content), initial upper shelf energy, and fluence to assure that a 50-foot-pound shelf energy, as required by Appendix G of 10 CFR 50 is maintained throughout the life of the vessel. The specimens are oriented as required by NB-2300 of Section III of the ASME Code. The vessel fracture toughness data is provided in Table 5.3-3.

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Charpy V-notch test data for the heat-affected zone of the limiting beltline region plate is presented in WCAP 10015 for WCGS. Complete Charpy test results for each weld and plate are provided in Tables 5.3-8 through 10. There are no other heat-affected zones which require impact testing per paragraph NB-4335.2 of the 1977 ASME Code. There are no ferritic base metals other than in the vessel in the reactor coolant pressure boundary.

5.3.1.6 Material Surveillance

In the surveillance program, the evaluation of radiation damage is based on preirradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch, tensile, and 1/2 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 conforms with ASTM E-185 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and 10 CFR 50, Appendix H.

The reactor vessel surveillance program prior to Refuel 14 used six specimen capsules. The capsules are located in guide baskets welded to the outside of the neutron shield pads and positioned directly opposite the center portion of the core. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The six capsules contain reactor vessel steel specimens, oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting base material located in the core region of the reactor vessel and associated weld metal and weld heat-affected zone metal. The six capsules contain 54 tensile specimens, 360 Charpy V-notch specimens (which include weld metal and weld heat-affected zone material), and 72 CT specimens. Archive material sufficient for two additional capsules is retained.

Dosimeters, as described below, are placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys are included to monitor the maximum temperature of the specimens. The specimens are enclosed in a tight-fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity. The complete capsule is helium leak tested. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent is made for surveillance material and as-deposited weld metal.

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Each of the six capsules contains the following specimens:

<u>Material</u>	<u>Number of Charpys</u>	<u>Number of Tensiles</u>	<u>Number of CTs</u>
Limiting base material [*]	15	3	4
Limiting base material ^{**}	15	3	4
Weld metal ^{***}	15	3	4
Heat-affected zone	15	-	-

* Specimens oriented in the major rolling or working direction.

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

*** Weld metal to be selected per ASTM E-185.

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 Pb, 2.5 percent Ag (579°F melting point)

97.5 percent Pb, 1.75 percent Ag, 0.75 percent Sn (590°F melting point)

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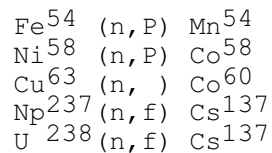
The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall, with the specimens being located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition temperature shift measurements are representative of the vessel at a later time in life. Data from CT fracture toughness specimens are expected to provide additional information for use in determining allowable stresses for irradiated material.

Correlations between the calculations and measurements of the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Section 5.3.1.6.1. The anticipated degree to which the specimens perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure is made by the use of data on all capsules withdrawn. The schedule for removal of the capsules for postirradiation testing is shown in Table 5.3-11 and conforms with ASTM E-185 and Appendix H of 10 CFR 50. Changes to the schedule for removal of the capsules is required to be approved by the NRC in accordance with appendix H of 10 CFR 50. The results of the reactor vessel material irradiation surveillance specimens are used to update the RCS pressure/temperature limits for heatup, cooldown, inservice hydrostatic and leak testing, criticality and PORV lift setting figures in the PTLR.

WCAP 10015 provides the location withdrawal schedule and lead factors for each capsule and the estimated reactor vessel end of life fluence at the 1/4 wall thickness as measured from the ID.

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

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



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

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

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

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

The Mn^{54} product of the $\text{Fe}^{54}(n,p)\text{Mn}^{54}$ reaction has a half-life of 314 days and emits gamma rays of 0.84 MeV energy, which are easily detected using a NaI scintillator. In irradiated steel samples, chemical separation of the Mn^{54} may be performed to ensure freedom from interfering activities. This separation is simple and very effective, yielding sources of very pure Mn^{54} activity. In some samples, all of the interferences may be corrected for by the gamma spectrometric methods without any chemical separation.

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

For this analysis, the DOT (Ref. 1), two-dimensional multigroup discrete ordinates transport code is employed to calculate the spectral data at the location of interest. Briefly, the DOT calculations utilize a 21 group energy scheme, an S_8 order of angular quadrature, and a P_1 expansion of the scattering matrix to compute neutron radiation levels within the geometry of interest. The reactor geometry employed here includes a description of the radial regions internal to the primary concrete (core barrel, neutron pad, pressure vessel, and water annuli) as well as the surveillance capsule and an appropriate reactor core fuel loading

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pattern and power distribution. Thus, distortions in the fission spectrum due to the attenuation of the reactor internals are accounted for in the analytical approach.

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

The induced Mn⁵⁴ activity in the iron flux monitors may be expressed as:

$$D = \frac{N_o}{A} f_i \int E^{\sigma(E)\phi(E)} \sum_{j=1}^n F_j (1 - e^{-\lambda t_j}) e^{-\lambda t_d}$$

where:

D	= induced Mn ⁵⁴ activity (dps/gm Fe)
N _o	= Avogadro's number (atoms/gm-atom)
A	= atomic weight of iron (gm/gm-atom)
f _i	= weight fraction of Fe ⁵⁴ in the detector
σ(E)	= energy dependent activation cross-section for the Fe ⁵⁴ (n,p)Mn ⁵⁴ reaction (barns)
φ(E)	= energy dependent neutron flux at the detector at full reactor power (n/cm ² sec)
λ	= decay constant of Mn ⁵⁴ (1/sec)
F _J	= fraction of full reactor power during the Jth time interval, J
τ _j	= length of the Jth irradiation period (sec)
τ _d	= decay time following the Jth irradiation period (sec)

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

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

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$$\int_{E_{TH}}^{\infty} \sigma(E) \phi(E) dE = \bar{\sigma} \bar{\phi}_{E_{TH}} = \frac{\int_{E_{TH}}^{\infty} \sigma_S(E) \phi_S(E) dE}{\int_{E_{TH}}^{\infty} \phi_S(E) dE} \bar{\phi}_{E_{TH}}$$

where:

- $\bar{\sigma}$ = effective spectrum average reaction cross-section for neutrons above energy, E_{TH}
- $\bar{\phi}_{E_{TH}}$ = average neutron flux above energy, E_{TH}
- $\sigma_S(E)$ = multigroup $Fe^{54}(n,p)Mn^{54}$ reaction cross-sections compatible with the DOT energy group structure
- $\phi_S(E)$ = multigroup energy spectra at the detector location obtained from the DOT analysis
- E_{TH} = threshold energy for damage correlation

Thus,

$$D = \frac{N_o}{A} \bar{\phi}_{E_{TH}} \bar{\sigma} \sum_{j=1}^n F_j (1 - e^{-\lambda \tau_j}) e^{-\lambda t_d}$$

or, solving for the threshold flux:

$$\bar{\phi}_{E_{TH}} = \frac{D}{\frac{N_o}{A} \bar{\sigma} \sum_{j=1}^n F_j (1 - e^{-\lambda \tau_j}) e^{-\lambda t_d}}$$

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

$$\Phi_{E_{TH}} = \bar{\phi}_{E_{TH}} \sum_{j=1}^n F_j \tau_j$$

where $\sum_{j=1}^n F_j \tau_j$ represents the total effective full power seconds of reactor operation up to the time of capsule removal.

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

No burnup correction was made to the measured activities, since burnout of the Mn^{54} product is not significant until the thermal flux level is about 10^{14} n/cm²-sec.

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

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

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

$$\phi(E, R, \theta, Z) = \bar{f}(E, R, \theta) F(Z)$$

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

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

The neutron flux at the surveillance capsule is given by:

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

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

$$\phi_{V-max} = \phi(E, R_{V-max}, \theta_{V-max}, Z_{V-max})$$

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The lead factor then becomes:

$$LF = \frac{\phi_c}{\phi_v - \max}$$

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

5.3.1.6.3 Ex-vessel surveillance program

The Reactor Cavity Neutron Measurement Program at Wolf Creek after Refuel 14 is designed to provide a verification of fast neutron exposure distributions within the reactor vessel wall and to establish a mechanism to enable long term monitoring of those portions of the reactor vessel and vessel support structure that could experience significant radiation induced increases in reference nil ductility transition temperature (RT_{NDT}) over the service lifetime of the plant. When used in conjunction with dosimetry from internal surveillance capsules and with the results of neutron transport calculations, the reactor cavity neutron measurements allow the projection of embrittlement gradients through the reactor vessel wall with a minimum uncertainty. Minimizing the uncertainty in the neutron exposure projections will, in turn, help to assure that the reactor can be operated in the least restrictive mode possible with respect to

- 10CFR50 Appendix G pressure/temperature limit curves for normal heat up and cool down of the reactor coolant system,
- Emergency Response Guideline (ERG) pressure/temperature limit curves, and
- Pressurized thermal shock (PTS) RT_{NDT} screening criteria.

In addition, an accurate measure of the neutron exposure of the reactor vessel and support structure can provide a sound basis for requalification should operation of the plant beyond the current design and/or licensed lifetime prove to be desirable.

The reactor cavity neutron dosimetry is installed in the annular air gap between the reactor vessel insulation and the primary concrete shield wall. The reactor cavity neutron dosimetry consist of aluminum dosimeter capsules connected to and supported by stainless steel bead chain. Each dosimetry chain is attached to and hangs from a stainless steel spring hook mounting plate. The local attachment plates are affixed to the horizontal portion of the reflective insulation below the reactor vessel nozzles (plant elevation 2012'+0.5") using four No. 14 × 3/4-long self-tapping sheet metal screws. The attachment plates are located in the vicinity of the Loop 1 outlet nozzle (at Reactor Angles of 5°, 15°, 30°, and 40°).

In some pressurized water reactor designs (like Wolf Creek) the neutron exposure rate at the surveillance capsule locations is much greater than that at the peak location on the reactor vessel. The ratio of these exposure rates is referred to as the surveillance capsule lead factor. Lead factors of three to five are not uncommon. With a high lead factor the reactor vessel material samples in a surveillance capsule may, if left in the reactor, receive neutron damage well beyond any projected end-of-life condition, thus rendering them useless. For example, a capsule with a lead factor of five would receive a 60-year exposure in as little as 12 years. This issue is particularly important for those plants planning for license renewal. The recently issued Generic Aging Lessons Learned (GALL) Report (NUREG-1801, April 2001), Section XI.M31 Reactor Vessel Surveillance, provides the following guidance for surveillance capsule management.

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A plant with a surveillance program containing capsules with projected fast neutron fluence exceeding a 60-year fluence at the end of 40 years, i.e., a lead factor greater than 1.5, should remove the capsules when they reach the 60-year exposure. One of these capsules should be tested to meet the requirements of ASTM E185 and the remaining capsules should be placed in storage without material testing. Subsequently, an alternative dosimetry program will need to be instituted to monitor reactor vessel neutron exposure through the license renewal period.

The NRC staff has recognized the importance of preserving the material specimens within the surveillance capsules. Any capsules that are to be left in the reactor vessel are to provide meaningful metallurgical data. For a high lead factor plant, if the remaining surveillance capsules are left in place, the material specimens will be irradiated well beyond the predicted end-of-life fast neutron exposure. At a projected end-of-life of 40 years, a surveillance capsule with a lead factor of three will have experienced the equivalent of a reactor vessel exposure of 120 years. Thus the material specimens would be damaged to such an extent that they would be unable to provide any useful data. With passive neutron sensors located in the reactor cavity the neutron exposure of the reactor vessel can be continuously monitored throughout plant life, as required by Appendix H, and the surveillance capsules can be removed and stored on site thus preserving this critical, irreplaceable material for future use. Thus the material specimens would:

- Monitor important azimuthal and axial exposure gradients over the entire beltline region of reactor vessel (unavailable with surveillance capsules) and provide measurements in proximity to critical areas on the reactor vessel.
- Provide long term monitoring that permits continuous evaluation of the effect of changes in reactor operation and changing fuel management schemes on the reactor vessel exposure, and
- Minimize the uncertainty in reactor vessel exposure projections using a combination of measurements and analytical predictions.

Within the nuclear industry it has been common practice to base estimates of the fast neutron exposure of reactor vessels either directly on the results of neutron transport calculations or on the analytical results normalized to measurements obtained from internal surveillance capsules. There are potential drawbacks associated with both of these approaches to exposure assessment.

In performing neutron transport calculations for pressurized water reactors, using the DORT code or similar NRC approved code, several design and operational variables have an impact on the magnitude of the analytical prediction of fast neutron exposure rates within the reactor vessel wall. Particularly important are cycle-to-cycle variations in core power distributions (especially with the implementation of low leakage loading patterns), variations of water temperature (density) in the peripheral fuel assemblies and the downcomer regions of the reactor internals, and deviations in as-built versus design dimensions for the reactor internals and vessel. Treatment of these important variables in the analysis leads to an increased uncertainty in the exposure predictions for the reactor vessel and may well result in the use of overly conservative estimates of reactor vessel embrittlement in the assessment of pressure / temperature limitations as well as of the expected lifetime of the components.

With the addition of supplementary passive neutron sensors in the reactor cavity annulus between the reactor vessel wall and the biological shield, the deficiencies in both surveillance capsule dosimetry and analytical prediction can be alleviated and the uncertainties associated with exposure estimates for the reactor vessel can be minimized. With state of the art neutron sensors deployed to establish the

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absolute magnitude of the azimuthal and axial exposure rate distributions in the reactor cavity, the burden placed on the neutron transport calculation is reduced.

An ex-vessel neutron dosimetry program can also provide additional data to support license renewal. As a comprehensive system to characterize the neutron exposure of the reactor vessel, it has the flexibility

Studies have shown that the operational and design variables cited above (that have a strong impact on the calculated magnitude of exposure rates) have only a minor effect on both the interpretation of reactor cavity measurements and on the extrapolation of measurement results to key reactor vessel locations. It is possible, therefore, to employ reactor cavity neutron measurements and plant specific calculations to produce reactor vessel exposure projections with a reduced uncertainty and without the excess conservatism inherent in an approach based on analysis alone. Furthermore, since the reactor cavity neutron measurements are not directly tied to the materials surveillance program, measurement intervals can be chosen to easily provide integral reactor vessel exposure over plant lifetime.

When the last surveillance capsule is removed for analysis, it is highly desirable to also analyze the Ex-Vessel Neutron Dosimetry. This provides a simultaneous in-vessel and ex-vessel measurement that results in the lowest uncertainty in the projected reactor vessel fluence and provides the most direct link between the existing in-vessel measurements and the ex-vessel measurements that will be used to monitor the neutron exposure of the vessel once the remaining surveillance capsules are withdrawn and placed in storage.

The use of fast ($E > 1.0$ MeV) neutron fluence to correlate measured materials properties changes to the neutron exposure of the material for light-water reactor applications has traditionally been accepted for the development of damage trend curves as well as for the implementation of trend curve data to assess reactor vessel condition. In recent years, however, it has been suggested that an exposure model that accounts for differences in neutron energy spectra between surveillance capsule locations and positions within the reactor vessel wall could lead to an improvement in the uncertainties associated with damage trend curves as well as to a more accurate evaluation of damage gradients through the reactor vessel wall.

Because of this potential shift away from a threshold fluence toward an energy dependent damage function for data correlation, ASTM Standard Practice E853, Analysis and Interpretation of Light-Water Reactor Surveillance Results, recommends reporting displacements per iron atom (dpa) along with fast neutron fluence ($E > 1.0$ MeV) to provide a data base for future reference. The energy dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693, Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom. The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the reactor vessel wall has already been promulgated in Revision 2 to Regulatory Guide 1.99, Radiation Damage to Reactor Vessel Materials.

With the aforementioned views in mind, the Ex-Vessel Neutron Dosimetry Program was established to meet the following objectives:

- Determine azimuthal and axial gradients of fast neutron exposure over the beltline region of the reactor vessel,
- Provide measurement capability sufficient to allow the determination of exposure parameters in terms of both fast ($E > 1.0$ MeV) neutron fluence and iron displacements per atom (dpa), and
- Provide a long-term monitoring capability for the beltline region of the reactor vessel and vessel support structure.

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Technical Description

To achieve the goals of the Ex-Vessel Neutron Dosimetry (EVND) Program, two types of measurements are made. Comprehensive sensor sets including radiometric monitors (RM) are employed at discrete locations within the reactor cavity to characterize the neutron energy spectrum variations axially and azimuthally over the beltline region of the reactor vessel. In addition, stainless steel gradient chains are used in conjunction with the encapsulated dosimeters to complete the mapping of the neutron environment between the discrete locations chosen for spectrum determinations.

In choosing sensor set locations for the Ex-Vessel Neutron Dosimetry Program, advantage is taken of the octant symmetry typical of pressurized water reactors. That is, subject to access limitations, spectrum measurements are concentrated to obtain azimuthal flux distributions in a single forty-five degree sector. Placement of the discrete sensor sets is such that spectrum determinations are made at various locations (5, 15, 30, and 40 degrees) on the midplane of the active core to measure the spectrum changes caused by the varying amounts of water located between the core and the reactor vessel. These thickness changes are due to the stair step shape of the reactor core periphery relative to the cylindrical geometry of the reactor internals and vessel and to the local nature of the neutron pads. The remaining sensor sets may be positioned opposite the top and bottom of the active core or opposite key reactor vessel welds at particular azimuthal angles of interest. Here the intent is to measure axial variations in neutron spectrum over the core height, particularly near the top of the fuel where back scattering of neutrons from primary loop nozzles and reactor vessel support structures can produce significant differences. At each of the azimuthal locations selected for spectrum measurements, stainless steel gradient chains extend over the full height of the active fuel.

Sensor Sets

The Ex-Vessel Neutron Dosimetry Program employs advanced sensor sets that are recommended by and are designed to the latest ASTM neutron dosimetry standards. The sensor sets consist of the following encapsulated dosimeters and gradient chains. Table 1 lists the neutron reactions that are of interest.

1. Radiometric Monitors (RM) - these include cadmium-shielded foils of the following metals: copper, titanium, iron, nickel, niobium, and cobalt-aluminum. Cadmium shielded fast fission reactions include ^{238}U and ^{237}Np in vanadium encapsulated oxide detectors. Bare iron and cobalt monitors are also included.

2. Gradient Chains - These stainless steel bead chains connect and support the dosimeter capsules containing the radiometric monitors. These segmented chains provide iron, nickel, and cobalt reactions that are used to complete the determination of the axial and azimuthal gradients. The high purity iron, nickel, and cobalt-aluminum foils contained in the multiple foil sensor sets provide a direct correlation with the measured reaction rates from these gradient chains. These crosscomparisons permit the use of the gradient measurements to derive neutron flux distributions in the reactor cavity with a high level of confidence.

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Material	Reaction of Interest	Neutron Energy Response (a)	Product Half-Life	Dosimeter Capsule Position (b)	Gradient Chain (c)
Copper	$^{63}\text{Cu}(n, \alpha) ^{60}\text{Co}$	4.53-11 MeV	5.271 yr	2-Cd	No
Titanium	$^{46}\text{Ti}(n, p) ^{46}\text{Sc}$	3.70-9.43 MeV	83.79 d	2-Cd	No
Iron	$^{54}\text{Fe}(n, p) ^{54}\text{Mn}$	2.27-7.54 MeV	312.3 d	1-B & 2-Cd	Yes
Nickel	$^{58}\text{Ni}(n, p) ^{58}\text{Co}$	1.98-7.51 MeV	70.82 d	2-Cd	Yes
$^{238}\text{U}(\alpha, e)$	$^{238}\text{U}(n, f) ^{137}\text{Cs}$	1.44-6.69 MeV	30.07 yr	3-Cd	No
Niobium	$^{93}\text{Nb}(n, n^1) ^{93m}\text{Nb}$	0.95-5.79 MeV	16.13 y	3-Cd	No
$^{237}\text{Np}(\alpha, e)$	$^{237}\text{Np}(n, f) ^{137}\text{Cs}$	0.68-5.61 MeV	30.07 yr	3-Cd	No
Cobalt-Al	$^{59}\text{Co}(n, \gamma) ^{60}\text{Co}$	Thermal	5.271 yr	1-B & 2-Cd	Yes

Notes:

- a) Energies between which 90% of activity is produced (^{235}U fission spectrum).
- b) B denotes bare and Cd denotes cadmium shielded
- c) Determined with additional radiochemical analysis
- d) For the fission monitors ^{95}Zr (64.02 d) and ^{103}Ru (39.26 d) activities are also reported
- e) Fission monitors have been discontinued and are replaced by niobium.

5.3.1.7 Reactor Vessel Fasteners

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

Refueling procedures require that the studs, nuts, and washers be removed from the reactor closure and be placed in storage racks or suspended in the reactor vessel head belt ring holes while the head is removed to its storage stand during preparation for refueling. The storage racks are then removed from the refueling cavity and stored at convenient locations in containment or their cleaning location prior to removal of the reactor closure head and refueling cavity flooding. When a stud cannot be removed from the reactor vessel flange, it is covered with a protective cover. Therefore, the reactor closure studs are never exposed to the borated refueling cavity water. Additional protection against the possibility of incurring corrosion effects is assured by the use of a manganese base phosphate surfacing treatment.

The stud holes in the reactor flange are sealed with special plugs prior to flooding the reactor cavity, thus preventing leakage of the borated refueling water into the stud holes. When a stud cannot be removed, the protective cover installed over the stud also protects the stud hole from the borated refueling water.

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5.3.2 PRESSURE - TEMPERATURE LIMITS

5.3.2.1 Limit Curves

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

Predicted ΔRT_{NDT} values are derived using two curves: the effect of fluence and copper content on the shift of RT_{NDT} for the reactor vessel steels exposed to 550°F temperature curve and the maximum fluence at 1/4 T (thickness) and 3/4 T location (tips of the code reference flaw when flaw is assumed at inside diameter and outside diameter locations, respectively) curve. These curves are presented in the PTLR. 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) is limiting in the analysis.

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

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

Compliance with Regulatory Guide 1.99 is discussed in Appendix 3A.

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(N).1.1. These transients are representative of the operating conditions that should prudently be considered to occur during plant operation. The transients selected form a conservative basis for evaluation of the RCS to insure the integrity of the RCS equipment.

Those transients listed as upset condition transients are given in Table 3.9(N)-1. None of these transients result in pressure-temperature changes which exceed the heatup and cooldown limitations, as described in Section 5.3.2.1 and in the PTLR.

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5.3.3 REACTOR VESSEL INTEGRITY

5.3.3.1 Design

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, bolted, flanged, and gasketed hemispherical upper head. The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leakoff connections: one between the inner and outer ring and one outside the outer O-ring. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains head adapters. These head adapters are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adapters contains Acme threads for the assembly of control rod drive mechanisms or instrumentation adapters. The seal arrangement at the upper end of these adapters consists of a welded flexible canopy seal. 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. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

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

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

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

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

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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 trips, and startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life. Vessel analysis results in a usage factor that is less than 1.

The design specifications require analysis to prove that the vessel is in compliance with the fatigue and stress limits of the ASME Code, Section III. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The analyzed heatup and cooldown rates imposed by plant operating limits are 100°F in any one hour except for cooldown of the pressurizer, which is limited to 200°F in any one hour. In practice, these operations occur more slowly. These rates are reflected in the vessel design 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 WCGS reactor vessel manufacturer is Combustion Engineering Corporation.

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, except for the vessel support surfaces and the top surface of the external seal ring, are painted with a heat-resistant paint before shipment.

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

5.3.3.6 Operating Conditions

Operating limitations for the reactor vessel are presented in Section 5.3.2, as well as in the PTLR.

In addition to the analysis of primary components discussed in Section 3.9(N).1.4, the reactor vessel is further qualified to ensure against unstable crack growth under faulted conditions. Actuation of the emergency core cooling system (ECCS) following a loss-of-coolant accident produces relatively high thermal stresses in regions of the reactor vessel which come into contact with ECCS water. Primary consideration is given to these areas, including the reactor vessel beltline region and the reactor vessel primary coolant nozzle, to ensure the integrity of the reactor vessel under this severe postulated transient.

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

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

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses

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perpendicular to the major plane of the crack), the stress intensity factor is designated as K_I and the critical stress intensity factor is designated K_{IC} . Commonly called the fracture toughness, K_{IC} is an inherent material property which is a function of temperature and strain rate. Any combination of applied load, structural configuration, crack geometry, and size which yields a stress intensity factor K_{IC} for the material will result in crack instability.

The criterion of the applicability of LEFM is based on plasticity considerations at the postulated crack tip. Strict applicability (as defined by ASTM) of LEFM to large structures where plane strain conditions prevail requires that the plastic zone developed at the tip of the crack does not exceed 2.25 percent of the crack depth. In the present analysis, the plastic zone at the tip of the postulated crack can reach 20 percent of the crack depth. However, LEFM has been successfully used to provide conservative brittle fracture prevention evaluations, even in cases where strict applicability of the theory is not permitted due to excessive plasticity. Recently, experimental results from the Heavy Section Steel Technology (HSST) Program intermediate pressure vessel tests have shown that LEFM can be applied conservatively as long as the pressure component of the stress does not exceed the yield strength of the material. The addition of the elastically calculated thermal stresses, which results in total stresses in excess of the yield strength, does not affect the conservatism of the results, provided that these thermal stresses are included in the evaluation of the stress intensity factors. Therefore, for faulted conditions analyses, LEFM is considered applicable for the evaluation of the vessel inlet nozzle and beltline region.

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

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

5.3.3.7 Inservice Surveillance

The internal and external surfaces of the reactor vessel are accessible for periodic inspection. Visual and/or nondestructive techniques are used. 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.

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The closure head is examined visually during each refueling, Optical devices permit a selective inspection of the cladding, control rod drive mechanism nozzles, and the gasket seating surface. The knuckle transition piece, which is the area of highest stress of the closure head, is accessible on the outer surface for visual inspection, dye penetrant or magnetic particle, and ultrasonic testing. The closure studs and nuts can be inspected periodically using visual, magnetic particle, and ultrasonic techniques.

The closure studs, nuts, washers, and the vessel flange seal surface, as well as the full penetration welds in the following areas of the installed reactor vessel, are available for nondestructive examination:

- a. Vessel shell - from the inside and outside surfaces
- b. Primary coolant nozzles - from the inside and outside surfaces*
- c. Closure head - from the inside and outside surfaces.
Bottom head - from the inside and outside surfaces.
- d. Field welds between the reactor vessel nozzle safe ends and the main coolant piping - from the inside and outside surfaces.

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

- a. All reactor internals are completely removable. The tools and storage space required to permit these inspections are provided.
- b. The closure head is stored dry on the reactor operating deck during refueling to facilitate direct visual inspection.
- c. Reactor vessel studs, nuts, and washers can be removed to dry storage during refueling. Studs which cannot be removed are covered to protect from borated refueling pool water, subsequently cleaned and inspected in-situ.
- d. Access is provided to the reactor vessel nozzle safe ends. The insulation covering the nozzle-to-pipe welds may be removed.

*Only partial outside diameter coverage is provided.

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- e. Reactor cavity is designed to allow access to the outside surface of the vessel. Tracks are installed to allow mechanical equipment to inspect the vessel surface.

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:

- a. 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 bond defect allowed is 1/4 inch by 3/4 inch 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 0.442 square inches (3/4 inch diameter) in all other regions are rejected.
- b. The design of the reactor vessel shell is an uncluttered cylindrical surface to permit future positioning of the test equipment without obstruction.
- c. The weld deposited clad surface on both sides of the welds to be inspected is specifically prepared to assure meaningful ultrasonic examinations.
- d. During fabrication, all full penetration ferritic pressure boundary welds are ultrasonically examined in addition to Code examinations.
- e. After the shop hydrostatic testing, all full penetration ferritic pressure boundary welds, as well as the nozzle to safe end welds, are ultrasonically examined from both the inside and outside diameters in addition to ASME Code, Section III requirements.

The vessel design and construction enable inspection in accordance with the ASME Code, Section XI. The reactor vessel inservice inspection program is in accordance with ASME Section XI as described in the Inservice Inspection Program and PTLR.

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5.3.4 REFERENCES

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2. Bachalet, C., Bamford, W. H., and Chirigos, J. N., "Method for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients," WCAP-8510, December 1975.
3. Singer, L. R. "Kansas Gas and Electric Company Wolf Creek Generating Station Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-10015, June 1982.

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TABLE 5.3-1

REACTOR VESSEL QUALITY ASSURANCE PROGRAM

	<u>RT*</u>	<u>UT*</u>	<u>PT*</u>	<u>MT*</u>
Forgings				
Flanges		Yes		Yes
Studs and nuts		Yes		Yes
CRD head adapter flange		Yes	Yes	
CRD head adapter tube		Yes	Yes	
Instrumentation tube		Yes	Yes	
Main nozzles		Yes		Yes
Nozzle safe ends		Yes	Yes	
Plates		Yes		Yes
Weldments				
Main seam	Yes	Yes		Yes
CRD head adapter to closure head connection			Yes	
Instrumentation tube to bottom head connection			Yes	
Main nozzle	Yes	Yes		Yes
Cladding		Yes	Yes	
Nozzle to safe ends	Yes	Yes	Yes	
CRD head adapter flange to CRD head adapter tube	Yes		Yes	
All full penetration ferritic pressure boundary welds accessible after hydrotest		Yes		Yes
Full penetration nonferritic pressure boundary welds accessible after hydrotest				
a. Nozzle to safe ends		Yes	Yes	
b. CRD head adapter flange to CRD head adapter tube			Yes	

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TABLE 5.3-1 (Sheet 2)

	<u>RT*</u>	<u>UT*</u>	<u>PT*</u>	<u>MT*</u>
Seal ledge				Yes
Head lift lugs				Yes
Core pad welds			Yes	

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

NOTE:

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

In addition, UT examination per the in-process/post-hydro UT requirements is performed on the following:

1. Base metal repairs in the core region.
2. Base metal repairs in the ISI zone (1/2 T).

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TABLE 5.3-2

REACTOR VESSEL DESIGN PARAMETERS

Design/operating pressure, psig	2,485/2,317*
Design temperature, F	650
Overall height of vessel and closure head, bottom head outside diameter to top of control rod mechanism adapter, ft-in.	43-10
Thickness of RPV head insulation, minimum, in.	3
Number of reactor closure head studs	54
Diameter of reactor closure head/studs, minimum shank, in.	6-13/16
Outside diameter of flange, in.	205
Inside diameter of flange, in.	167
Outside diameter at shell, in.	190-1/2
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 beltline thickness, minimum, in.	8-5/8
Closure head thickness, in.	7
Nominal water volume, ft ³	3,700

* The operating pressure used to control the plant is 2,235 psig and is measured in the pressurizer.

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TABLE 5.3-3
 REACTOR VESSEL MATERIAL PROPERTIES

COMPONENT	CODE NO.	MATERIAL SPEC. NO.	Cu (%)	P (%)	TNDT (F)	RTNDT (F)	Avg. Upper Shelf	
							NMWD** (FT-LB)	MWD* (FT-LB)
Closure Head Dome	R2516-1	A533B, CL.1	0.12	0.010	-40	0	112	-
Closure Head Torus	R2515-1	A533B, CL.1	0.11	0.009	-20	-20	119	-
Closure Head Flange	R2504-1	A508 CL. 2	-	0.013	20	20	139	-
Vessel Flange	R2501-1	A508 CL. 2	-	0.012	20	20	102	-
Inlet Nozzle	R2502-1	A508 CL. 2	-	0.010	-20	-20	147	-
Inlet Nozzle	R2502-2	A508 CL. 2	-	0.009	-20	-20	137	-
Inlet Nozzle	R2502-3	A508 CL. 2	0.11	0.010	-20	-20	156	-
Inlet Nozzle	R2502-4	A508 CL. 2	0.11	0.010	-30	-30	156	-
Outlet Nozzle	R2503-1	A508 CL. 2	-	0.006	-10	-10	126	-
Outlet Nozzle	R2503-2	A508 CL. 2	-	0.009	0	0	129	-
Outlet Nozzle	R2503-3	A508 CL. 2	-	0.007	0	0	136	-
Outlet Nozzle	R2503-4	A508 CL. 2	-	0.007	0	0	114	-
Nozzle Shell	R2004-1	A533B, CL. 1	0.05	0.010	-40	10	118	-
Nozzle Shell	R2004-2	A533B, CL. 1	0.04	0.011	-40	20	121	-
Nozzle Shell	R2004-3	A533B, CL. 1	0.04	0.008	-50	0	133	-
Inter. Shell	R2005-1	A533B, CL. 1	0.04	0.008	-20	-20	127	156
Inter. Shell	R2005-2	A533B, CL. 1	0.04	0.007	-30	-20	127	143
Inter. Shell	R2005-3	A533B, CL. 1	0.05	0.007	-30	-20	135	164
Lower Shell	R2508-1	A533B, CL. 1	0.09	0.009	-40	0	87	118
Lower Shell	R2508-2	A533B, CL. 1	0.06	0.008	-10	10	100	127
Lower Shell	R2508-3	A533B, CL. 1	0.07	0.008	-20	40	86	127
Bottom Head Torus	R2517-1	A533B, CL. 1	0.11	0.010	-80	-30	92	-
Bottom Head Dome	R2518-1	A533B, CL. 1	0.12	0.009	-60	-60	154	-
Inter. and lower shell long. weld seams	G2.06	SAW	0.04	0.006	-50	-50	150	-
Inter. to lower shell girth weld seam	E3.16	SAW	0.05	0.007	-50	-50	98	-
Weld HAZ	-	-	-	-	-80	-80	171	-

*Major working direction

**Normal to major working direction

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TABLE 5.3-4 HAS BEEN DELETED

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TABLE 5.3-5 IS DELETED

TABLE 5.3-6

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TABLE 5.3-7

VESSEL BELTLINE REGION WELD METAL IDENTIFICATION INFORMATION

<u>Weld Seam Identification</u>	<u>Weld Control No.</u>	<u>Weld Procedure Qual. No.</u>	<u>Weld Wire</u>		<u>Flux</u>	
			<u>Type</u>	<u>Heat No.</u>	<u>Type</u>	<u>Lot No.</u>
Int. shell long weld seam 101-124A, B, and C	G2.06	SAA-SMA-12.12-102	B4	90146	Linde 0091	0842
Lower shell long weld seam 101-142A, B, and C	G2.06	SAA-SMA-12.12-102	B4	90146	Linde 0091	0842
Inter. to lower shell girth seam 101-171	E3.16	SAA-SMA-3.3-118	B4	90146	Linde 124	1061
Surveillance test weld	E3.16	SAA-SMA-3.3-118	B4	90146	Linde 124	1061

<u>Weld Control No.</u>	<u>Weld Metal Chemical Composition (Wt. %)</u>									
	<u>C</u>	<u>Mn</u>	<u>P</u>	<u>S</u>	<u>Si</u>	<u>Cr</u>	<u>Ni</u>	<u>Mo</u>	<u>Cu</u>	<u>V</u>
G2.06	.15	1.16	.006	.011	.18	.05	.04	.51	.04	.005
E3.16	.097	1.27	.007	.011	.52	.09	.05	.50	.05	.004

NOTES

1. The test weld was fabricated from plates R2508-1 and R2508-3.
2. The test weldment was stress relieved at 1150°F for 10.25 hours - furnace cooled.

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TABLE 5.3-8

BELTLINE REGION INTERMEDIATE SHELL PLATE TOUGHNESS

<u>Plate R2005-1</u>				<u>Plate R2005-2</u>				<u>Plate R2005-3</u>			
<u>Temp.</u> <u>(F)</u>	<u>Energy</u> <u>(ft lb)</u>	<u>Shear</u> <u>(%)</u>	<u>Lat. Exp.</u> <u>(mils)</u>	<u>Temp.</u> <u>(F)</u>	<u>Energy</u> <u>(ft lb)</u>	<u>Shear</u> <u>(%)</u>	<u>Lat. Exp.</u> <u>(mils)</u>	<u>Temp.</u> <u>(F)</u>	<u>Energy</u> <u>(ft lb)</u>	<u>Shear</u> <u>(%)</u>	<u>Lat. Exp.</u> <u>(mils)</u>
-60	6	0	2	-60	10	0	4	-60	6	0	2
-60	7	0	3	-60	11	0	6	-60	7	0	3
-60	7	0	2	-60	8	0	3	-60	6	0	2
-20	20	5	14	-20	25	5	18	-20	20	5	10
-20	27	10	17	-20	48	20	32	-20	11	0	4
-20	14	0	8	-20	37	15	24	-20	12	0	4
40	72	30	48	30	39	15	28	30	49	20	35
40	62	25	40	30	65	30	46	30	55	25	38
40	58	25	39	30	70	35	48	30	59	30	41
60	73	30	46	40	69	35	49	40	65	30	44
60	56	25	36	40	79	40	55	40	58	30	40
60	69	30	44	40	82	40	56	40	84	40	58
100	95	40	65	60	78	30	51	60	65	25	45
100	96	50	64	60	89	30	53	60	87	30	52
100	89	50	63	60	80	30	52	60	86	30	57
160	122	100	77	100	105	70	69	100	94	40	62
160	126	100	76	100	102	70	70	100	97	40	61
160	132	100	80	100	108	70	75	100	108	50	72
				160	128	100	84	160	140	100	81
				160	125	100	78	160	136	100	74
				160	127	100	79	160	129	100	77
	T_{NDT}	-20°F			T_{NDT}	-30°F		T_{NDT}	-30°F		
	RT_{NDT}	-20°F			RT_{NDT}	-20°F		RT_{NDT}	-20°F		

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TABLE 5.3-9
BELTLINE REGION LOWER SHELL PLATE TOUGHNESS

<u>Plate R2508-1</u>				<u>Plate R2508-2</u>				<u>Plate R2508-3</u>			
<u>Temp.</u> <u>(F)</u>	<u>Energy</u> <u>(ft lb)</u>	<u>Shear</u> <u>(%)</u>	<u>Lat. Exp.</u> <u>(mils)</u>	<u>Temp.</u> <u>(F)</u>	<u>Energy</u> <u>(ft lb)</u>	<u>Shear</u> <u>(%)</u>	<u>Lat. Exp.</u> <u>(mils)</u>	<u>Temp.</u> <u>(F)</u>	<u>Energy</u> <u>(ft lb)</u>	<u>Shear</u> <u>(%)</u>	<u>Lat. Exp.</u> <u>(mils)</u>
-40	12	0	5	-30	22	5	11	-40	5	0	2
-40	12	0	6	-30	17	0	8	-40	4	0	1
-40	13	0	6	-30	23	5	13	-40	5	0	1
0	28	10	22	10	28	10	15	0	19	5	15
0	29	10	22	10	31	10	19	0	15	0	12
0	27	10	22	10	29	10	17	0	16	0	12
20	32	10	26	50	41	15	26	40	34	15	23
20	37	15	30	50	52	25	36	40	29	10	19
20	40	20	32	50	49	20	32	40	27	10	16
50	53	30	40	60	48	20	34	90	54	25	44
50	52	35	38	60	47	20	34	90	48	25	38
50	46	30	33	60	45	20	33	90	53	25	42
60	58	40	42	70	56	25	39	100	52	25	41
60	65	50	51	70	55	25	40	100	57	30	43
60	56	40	41	70	60	30	42	100	58	30	47
100	84	80	61	100	63	40	45	160	93	100	71
100	74	70	58	100	59	30	42	160	79	100	68
100	78	70	60	100	76	50	53	160	86	100	74
160	87	100	62	160	96	90	68	212	85	100	66
160	88	100	65	160	96	90	64	212	80	100	64
160	87	100	66	160	97	90	68	212	87	100	66
				212	100	100	68				
				212	96	100	64				
				212	104	100	71				
	T _{NDT}	-40°F			T _{NDT}	-10°F			T _{NDT}	-20°F	
	RT _{NDT}	0°F			RT _{NDT}	10°F			RT _{NDT}	40°F	

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TABLE 5.3-10

BELTLINE REGION WELD METAL TOUGHNESS

Weld Control No. G2.06				Weld Control No. E3.16			
Temp. (°F)	Energy (ft lb)	Shear (%)	Lat. Exp. (mils)	Temp. (°F)	Energy (ft lb)	Shear (%)	Lat. Exp. (mils)
-60	20	0	12	-80	11	0	9
-60	23	5	10	-80	8	0	4
-60	26	5	14	-80	7	0	6
-40	39	20	23	-40	45	20	33
-40	31	15	16	-40	42	20	30
-40	43	20	26	-40	32	15	27
-20	75	40	50	10	58	30	41
-20	108	60	63	10	52	25	37
-20	58	30	38	10	60	40	46
10	102	60	61	60	106	80	69
10	128	80	79	60	92	90	64
10	120	70	71	60	97	90	61
20	125	80	77	100	97	95	73
20	119	70	78	100	95	95	68
20	123	70	68	100	103	95	72
60	151	100	88	160	99	100	71
60	150	100	87	160	96	100	72
60	148	100	87	160	95	100	79
100	148	100	80				
100	155	100	85				
100	145	100	81				
	T _{NDT}	-50°F			T _{NDT}	-50°F	
	R _T T _{NDT}	-50°F			R _T T _{NDT}	-50°F	

TABLE 5.3-11

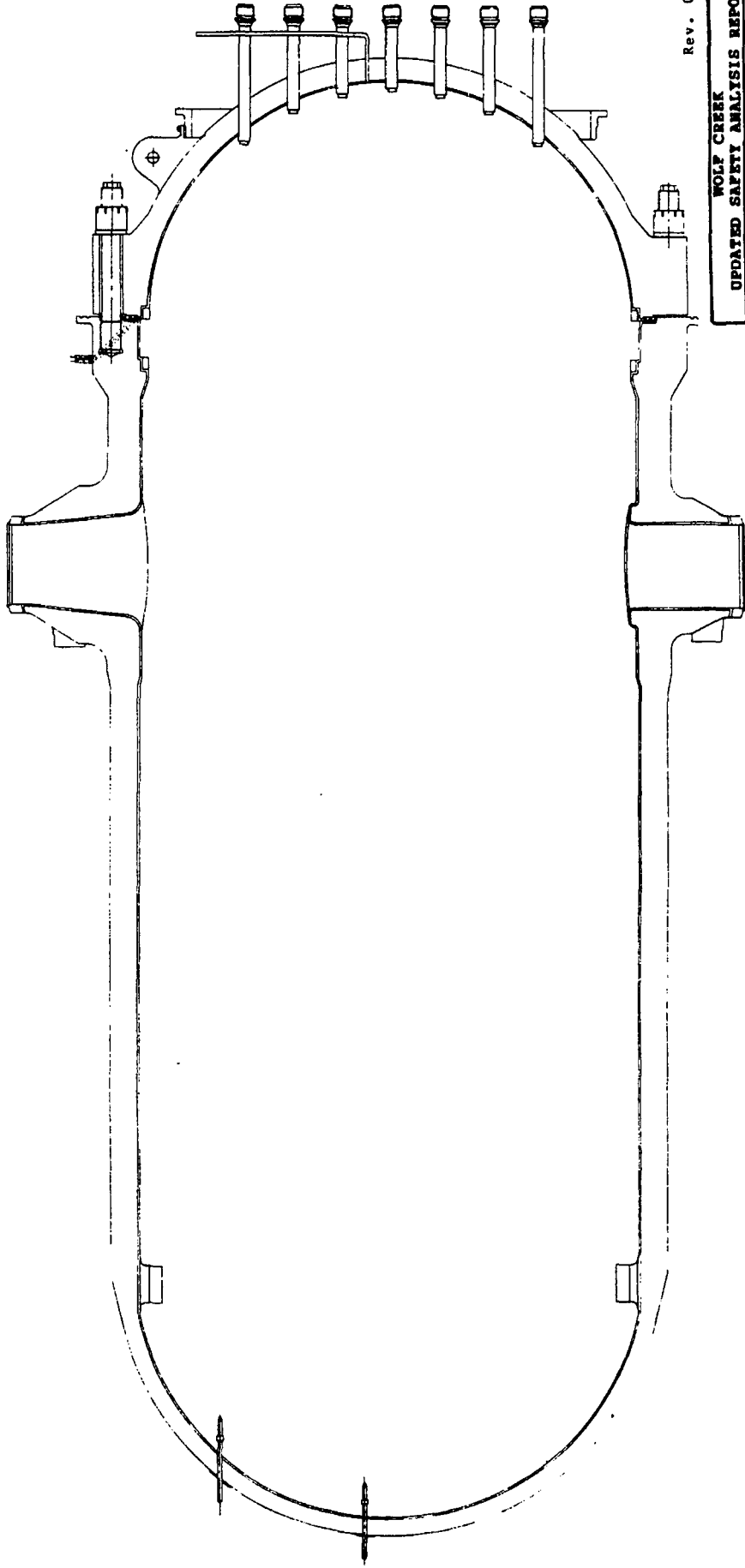
REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

<u>CAPSULE NUMBER</u>	<u>VESSEL LOCATION</u>	<u>LEAD FACTOR</u>	<u>WITHDRAWAL TIME</u>
U	58.5°	4.25	1.07 EFPY ^(b)
Y	241°	3.93	4.79 EFPY ^(b)
V	61°	4.02	9.78 EFPY ^(b)
X	238.5°	4.30	13.83 EFPY ^(b)
W	121.5°	4.11	14 th Refueling (Storage)
Z	301.5°	4.11	14 th Refueling (Storage)

(a) Updated in Capsule X dosimetry analysis.

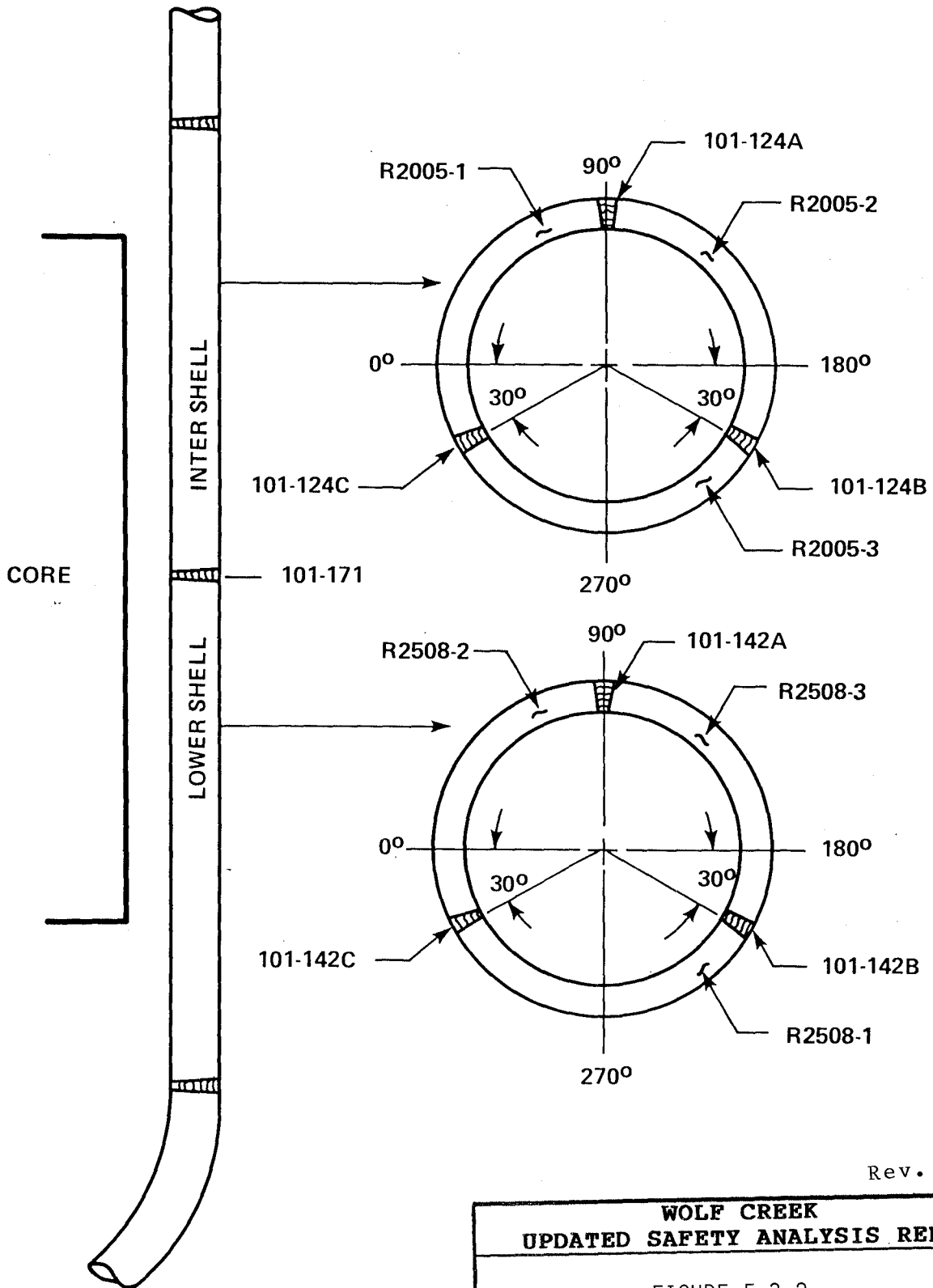
(b) Capsule withdrawn and analyzed.

NOTE: Changes to the schedule for removal of the capsules is required to be approved by the NRC in accordance with Appendix H of 10CFR50.



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FIGURE 5.3-1
REACTOR VESSEL



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FIGURE 5.3-2

WOLF CREEK UNIT 1 REACTOR VESSEL
BELTLINE REGION MATERIAL
IDENTIFICATION AND LOCATION

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5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

5.4.1.1 Design Bases

The reactor coolant pump provides an adequate core cooling flow rate for heat transfer to maintain a departure from nucleate boiling ratio (DNBR) greater than the Safety Analysis Limit DNBR as defined in the COLR within the parameters of operation. The required net positive suction head 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 flow.

The reactor coolant pump motor is tested, without mechanical damage, at overspeeds up to and including 125 percent of normal speed. The retention of integrity of the flywheel during a LOCA is demonstrated in Reference 1.

Steam/water tests planned jointly by Westinghouse, Framatome, and the French Atomic Energy Commission (CEA) are discussed in Reference 2. The ultimate use of the data from this testing will be to develop an empirical two-phase flow pump performance model. It is expected that this new model will confirm that the present pump model conservatively predicts performance in all LOCA conditions and thus increase the safety margin available in the emergency core cooling system (ECCS) and reactor coolant pump overspeed analyses.

The pump/motor system is designed for the SSE at the site.

5.4.1.2 Pump Description

5.4.1.2.1 Design Description

The reactor coolant pump is shown in Figure 5.4-1. The reactor coolant pump design parameters are given in Table 5.4-1. Code and material requirements are provided in Section 5.2.

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

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

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- a. The hydraulic section consists of the casing, thermal barrier, flange, impeller/diffuser, and diffuser adapter.
- b. The shaft seal section consists of three primary devices. They are the number 1 controlled leakage, film riding face seal, and the number 2 and number 3 rubbing face seals. These seals are contained within the thermal barrier heat exchanger assembly and seal housing. Collectively, they provide a pressure breakdown from the reactor coolant system (RCS) pressure to ambient conditions. A fourth sealing device called a shutdown seal is housed within the No. 1 seal area and is passively actuated by high temperature if seal cooling is lost.
- c. The motor is a drip-proof squirrel cage induction motor with a vertical solid shaft, an oil lubricated double-acting Kingsbury type thrust bearing, upper and lower oil lubricated radial guide bearings, and a flywheel.

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

5.4.1.2.2 Description of Operation

Reactor coolant enters the suction nozzle, is directed to the impeller by the diffuser adapter, is pumped through the diffuser, and exits through the discharge nozzle.

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

Component cooling water is provided to the thermal barrier heat exchanger. During normal operation, the thermal barrier limits the heat transfer from hot reactor coolant to the radial bearing and to the seals. In addition, if a loss of seal injection flow should occur, the thermal barrier heat exchanger cools reactor coolant to an acceptable level before it enters the bearing and seal area.

Reactor coolant pump operation with either seal water injection or component cooling water alone is acceptable for an unlimited time. As described in Sections 9.2.2 and 9.3.4 the component cooling water and the injection paths provide diverse cooling means which precludes seal failures due to any single failure or due to the effects of an SSE.

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The reactor coolant pump motor bearings are of conventional design. The radial bearings are the segmented pad type, and the thrust bearing is a double-acting Kingsbury type. All are oil lubricated. Component cooling water is supplied to the external upper bearing oil cooler and to the integral lower bearing oil cooler. The reactor coolant pump motor bearings are qualified for 10 minutes operation without component cooling water with no resultant damage.

The motor is a water/air cooled, Class F thermalastic epoxy insulated, squirrel cage induction motor. The rotor and stator are of standard construction and are cooled by air.

Six resistance temperature detectors are imbedded in the stator windings to sense stator temperature. The top of the motor consists of a flywheel and an antireverse 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 heat exchangers, which are supplied with component cooling water. Each motor has two such coolers, mounted opposite each other. In passing through the coolers, the air is cooled to below 122°F so that little heat is rejected to the containment from the motors.

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

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 cover.

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

5.4.1.3 Design Evaluation

5.4.1.3.1 Pump Performance

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

The estimated performance characteristic is shown in Figure 5.4-2. The "knee" at about 45-percent design flow introduces no operational restrictions, since the pumps operate at full flow.

The reactor trip system assures that pump operation and core cooling capability are within the assumptions used for loss of flow analyses (See Chapter 15.0). In addition, in the event that a reactor coolant pump is taken out of service during operation, adequate core cooling is provided, and continued plant operation without a reactor trip can be accommodated if the reactor coolant pump is stopped following an orderly reduction in power. The WCGS Technical Specifications require shutdown to hot standby within six hours after a reactor coolant pump stops.

Long-term tests have been conducted on less than full scale prototype seals, as well as on full size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled leakage shaft seal pump design.

The support of the stationary member of the number 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 number 1 seal entirely bypassed (full system pressure on the number 2 seal) shows that small (approximately 4 to 12 gpm) leakage rates would be maintained for a period of time sufficient to secure the pump. Even if the number 1 seal were to fail entirely during normal operation, the number 2 seal would maintain these small leakage rates if the proper action is

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taken by the operator. An increase in number 1 seal leakoff rate will warn the plant operator of number 1 seal damage. Following warning of excessive seal leakage conditions, the plant operator will take corrective actions. Gross leakage from the pump does not occur if these procedures are followed.

Loss of offsite power causes loss of power to the pump and causes a temporary stoppage in the supply of seal injection flow to the pump and also of the component cooling water flow to the pump and motor. The emergency diesel generators are started automatically due to loss of offsite power so that seal injection flow is provided by the charging pumps. Component cooling water flow is subsequently restored automatically, within 2 minutes. Load shedding and sequencing is discussed in Section 8.3.

In the event of a loss of all AC power and/or loss of all seal cooling, the shutdown seal (SDS) will actuate on high seal cooling temperature to limit leakage from the RCP seal package. Leakage is limited when a thermal actuator retracts and causes the SDS piston ring and polymer ring to clamp down around the pump shaft

5.4.1.3.2 Coastdown Capability

It is important to reactor protection that the reactor coolant flow is maintained for a short time after a pump trip in order to remove heat stored in the fuel elements of the core. In order to provide this flow after interruption of power to the pumps, each reactor coolant pump is provided with a flywheel. The rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow. An inadvertent early actuation of the SDS on the pump shaft, with the shaft still rotating, will not adversely impact RCP coastdown. The coastdown flow transients are provided in the figures in Section 15.3. The coastdown capability of the pumps is maintained even under the most adverse case of a blackout coincident with the SSE. Core flow transients 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 reactor coolant pump bearings are primarily aimed at giving an accurate alignment and smooth operation over long periods of time in order to ensure a long life with negligible wear. The surface-bearing stresses are held at a very low value, and even under the most severe seismic transients remain below stress values that can 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.

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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 temperature detectors and so initiation of failure, separate from loss of oil, is indicated and alarmed in the control room as a high bearing temperature. This, again, requires pump shutdown. If these indications are ignored, and the bearing proceeded 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.

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. This constitutes a loss-of-coolant flow in the loop. 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, thus disengaging the flywheel and motor from the shaft. Following such a postulated seizure, the motor continues to run without any overspeed, and the flywheel maintains its integrity, as it is still supported on a shaft with two bearings. Flow transients are provided in 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 seals results in a shearing of the antirotation pin in the seal ring. An inadvertent actuation of the shutdown seal on the shaft will not interrupt core cooling flow provided by the RCP. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions 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 shut down for investigation.

5.4.1.3.5 Critical Speed

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

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5.4.1.3.6 Missile Generation

Precautionary measures taken to preclude missile formation from primary coolant pump components assure that the pumps do 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 in the heavy casing. Further discussion and analysis of missile generation is contained in Reference 1 and Section 3.5.

5.4.1.3.7 Pump Cavitation

The minimum net positive suction head required by the reactor coolant pump at running speed is approximately a 192-foot head (approximately 85 psi). In order for the controlled leakage seal to operate correctly, it is necessary to require a minimum differential pressure of approximately 200 psi across the number 1 seal. This corresponds to a primary loop pressure at which the minimum net positive suction head is exceeded, and no limitation on pump operation occurs.

5.4.1.3.8 Pump Overspeed Considerations

For turbine trips actuated by either the reactor trip system or the turbine protection system, the generator and reactor coolant pumps are maintained connected to the external network for 30 seconds to prevent any pump overspeed condition. The overspeed condition is prevented by the dynamic braking action of the pump motor. In case a generator trip de-energizes the pump busses, the reactor coolant pump motors are transferred to offsite power within 6 to 10 cycles. Further discussion of pump overspeed considerations and missile generation is contained in Reference 1 and Section 3.5.

5.4.1.3.9 Antireverse Rotation Device

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

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At an approximate forward speed of 70 rpm, the pawls drop and bounce across the ratchet plate; as the motor continues to slow, the pawls drag across the ratchet plate. After the motor has 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. While the motor is running at speed, there is no contact between the pawls and ratchet plate.

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

5.4.1.3.10 Shaft Seal Leakage

During normal operation, leakage along the reactor coolant pump shaft is controlled by three shaft seals arranged in series so that reactor coolant leakage to the containment is essentially zero. Injection flow is directed to each reactor coolant pump via a seal water injection filter. It enters the pumps through a connection of the thermal barrier flange and flows to an annulus around the shaft inside the thermal barrier. Here the flow splits: a portion flows down the shaft to cool the bearing and enters the RCS; the remainder flows up the shaft through the seals. This flow provides a backpressure on the number 1 seal and a controlled flow through the seal. Above the seal, most of the flow leaves the pump via the number 1 seal discharge line. Minor flow passes through the number 2 seal and leakoff line. A back flush injection from a head tank flows into the number 3 seal between its "double dam" seal area. At this point, the flow divides with half flushing through one side of the seal and out the number 2 seal leakoff while the remaining half flushes through the other side and out of the number 3 seal leakoff. This arrangement assures essentially zero leakage of reactor coolant or trapped gases from the pump.

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

5.4.1.3.11 Seal Discharge Piping

The number 1 seal reduces the coolant pressure to that of the volume control tank. Water from each pump number 1 seal is piped to a common manifold, through the seal water return filter, and through the seal water heat exchanger where the temperature is

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reduced to that of the volume control tank. The number 2 and number 3 leakoff lines dump number 2 and 3 seal leakage to the reactor coolant drain tank and the containment sump, respectively.

5.4.1.4 Tests and Inspections

The reactor coolant pumps can be inspected in accordance with the ASME Code, Section XI, for inservice inspection of nuclear reactor coolant systems.

The pump casing is cast in one piece, eliminating welds in the casing. Support feet are cast integral with the casing to eliminate a weld region.

The design enables disassembly and removal of the pump internals for usual access to the internal surfaces of the pump casing.

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

5.4.1.5 Pump Flywheels

5.4.1.5.1 Pump Flywheel Integrity

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

5.4.1.5.2 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 primary coolant pumps run at approximately 1,190 rpm and may operate briefly at overspeeds up to 109 percent (1,295 rpm) during loss of load. For conservatism, however, 125 percent of operating speed was selected as the design speed for the primary coolant pumps. The flywheels were given a preoperational test of 125 percent of the maximum synchronous speed of the motor.

5.4.1.5.2.1 Fabrication and Inspection

The flywheel consists of two thick plates bolted together. The flywheel material is produced by a process that minimizes flaws in the material and improves its fracture toughness properties, such as vacuum degassing, vacuum melting, or electroslag remelting. Each plate is fabricated from SA-533, Grade B, Class 1 steel.

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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 1.14.

Flywheel blanks are flame-cut from the SA-533, Grade B, Class 1 plates with at least 1/2 inch of stock left on the outer and bore radii for machining to final dimensions. The flywheel plates, both before and after assembly, are subjected to magnetic particle or liquid penetrant examination. Included in this examination are all surfaces within a minimum radial distance of 4 inches beyond the final machined bore. This includes the bore surface and the keyways. 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.

5.4.1.5.2.2 Material Acceptance Criteria

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

- a. The nil-ductility transition temperature (NDTT) of the flywheel material is obtained by two drop weight tests (DWT) which exhibit "no-break" performance at 20°F in accordance with ASTM E-208. The above drop weight tests demonstrate that the NDTT of the flywheel material is no higher than 10°F.
- b. A minimum of three Charpy V-notch impact specimens from each plate are tested at ambient (70°F) temperature in accordance with the specification ASME SA-370. 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, and, therefore, RT_{NDT} of 10°F can be assumed. An evaluation of flywheel overspeed has been performed which concludes that flywheel integrity will be maintained (Ref. 1).

As stated in reference 1, the normal operating temperature is 120°F. The Charpy V-notch and dropweight tests confirm that the normal operating temperature is in excess of 100°F above the RT_{NDT} of the flywheel material.

Thus, it is concluded that flywheel plate materials are suitable for use and can meet Regulatory Guide 1.14 acceptance criteria on the bases of the suppliers' certification data. The degree of compliance with Regulatory Guide 1.14 is further discussed in Appendix 3A.

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5.4.1.5.2.3 Accessibility

The reactor coolant pump motors are designed so that, by removing the cover to provide access, the flywheel is available to allow an inservice inspection program in accordance with requirements of Section XI of the ASME Code and the recommendations of Regulatory Guide 1.14.

5.4.1.5.2.4 Spin Testing

Each flywheel assembly is spin tested at the design speed of the flywheel, i.e., 125 percent of the maximum synchronous speed of the motor.

5.4.1.5.3 Preservice Inspection

Post spin testing of reactor coolant pump flywheels is discussed in Appendix 3A under the response to Regulatory Guide 1.14.

5.4.1.5.4 Inservice Inspection

The reactor coolant pump flywheels are inservice inspected in accordance with the recommendations given in Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1, August 1975. The Administrative Controls portion of the Technical Specifications provides specific information on the commitment to the inspection requirements of Regulatory Guide 1.14.

5.4.2 STEAM GENERATORS

5.4.2.1 Design Bases

Steam generator design data are given in Table 5.4-3. Code classifications of the steam generator components are given in Section 3.2. Although the ASME classification for the secondary side is specified to be Class 2, all pressure-retaining parts of the steam generator, and thus both the primary and secondary pressure boundaries, are designed to satisfy the criteria specified in Section III of the ASME Code for Class 1 components. The design stress limits, transient conditions, and combined loading conditions applicable to the steam generator are discussed in Section 3.9(N).1. Estimates of radioactivity levels anticipated in the secondary side of the steam generators during normal operation and the bases for the estimates are given in Chapter 11.0. The accident analysis of a steam generator tube rupture is discussed in Chapter 15.0.

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The internal moisture separation equipment is designed to ensure that moisture carryover does not exceed 0.25 percent by weight under the following conditions:

- a. Steady state operation up to 100 percent of full load steam flow, with water at the normal operating level.
- b. Loading or unloading at a rate of 5 percent of full power steam flow per minute in the range from 15 to 100 percent of full load steam flow.
- c. A step load change of 10 percent of full power in the range from 15 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 10.0. Compatibility of steam generator tubing with both primary and secondary coolants is discussed further in Section 5.4.2.3.2.

The steam generator is designed to prevent unacceptable damage from mechanical or flow-induced vibration. Tube support adequacy is discussed in Section 5.4.2.5.3. The tubes and tube sheet are analyzed and confirmed to withstand the maximum accident loading conditions as they are defined in Section 3.9(N).1. Further consideration is given in Section 5.4.2.5.4 to the effect of tube wall thinning on accident condition stresses.

Access is provided to the primary side channel heads of the steam generator in order to permit inservice inspection and tube plugging, when required. Access is provided to the shell side of the steam generator in the region of the tube sheet and flow distribution baffle in order to permit inservice inspection and removal of accumulated sludge.

5.4.2.2 Design Description

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

The Model F steam generator is similar in configuration to the Model 51 steam generators in Westinghouse-supplied plants that are in operation. The Model F incorporates several improved features that have been developed through modification programs in operating steam generators. These features are illustrated in Figure

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5.4-4 and include: preferential distribution of feedwater to the hot leg portion of the tube bundle, removal of downcomer resistance, blockage of the tube lane, and improvements to the primary and secondary steam separators. The net effect of these changes, as has been demonstrated with the use of special instrumentation at Prairie Island, is to increase the flow velocities within the tube bundle, to reduce the tendency for deposition of sludge where it cannot be removed by the continuously operating blowdown system, to reduce the tendency for vapor generation at the tube sheet, and, to reduce moisture carryover with the steam.

The Model F steam generator incorporates several other improved features. These features are illustrated in Figure 5.4-5. A sealed thermal sleeve and J-nozzles on the feedring prevent the draining of water from the feedring inside the steam generator, and, together with a short horizontal length of feedwater piping to the feedring, have been incorporated to prevent water hammer.

The holes in the tube support plates of the Model F generator have a four-lobe shape that provides four lands to support the tube laterally. The holes are fabricated by drilling, followed by broaching. Figure 5.4-6 is an illustration of the "quatrefoil" broached holes.

The tubes are seal welded to the tube sheet cladding. Fusion welds are performed in compliance with Sections III and IX of the ASME Code and are dye penetrant inspected and leakproof tested. After welding, each tube is hydraulically expanded for the full depth of the tube sheet to the secondary surface to eliminate crevices between the tube and tube sheet.

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 steam generator. The head is divided into inlet and outlet chambers by a vertical divider plate extending from the apex of the head to the tube sheet.

Steam is generated on the shell side, flows upward, and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes, through a feedwater nozzle. The water is distributed circumferentially around the steam generator by means of a feedwater ring and then flows downward through an annulus between the tube wrapper and shell. The feedwater enters the ring via a welded thermal sleeve connection and leaves it through inverted "J" tubes located at the flow holes, which are at the top of the ring. These features are designed to prevent a condition which

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can result in water hammer occurrences in the feedwater piping. At the bottom of the wrapper, the water is directed toward the center of the tube bundle by a flow distribution baffle. This baffle arrangement serves to minimize the tendency of relatively low velocity fluid to deposit sludge in the tube bundle. Flow blockers, installed on the tube lane, restrict feedwater from flowing through the tube lane and bypassing the tubes. The steam-water mixture from the tube bundle rises into the steam drum section, where 16 individual centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary separators, which remove most of the remaining moisture and provide a quality of at least 99.75 percent. The separated water is combined with entering feedwater to flow back down the annulus between the wrapper and shell for recirculation through the steam generator. The dry steam exits from the steam generator through the outlet nozzle which is provided with a steam flow restriction, described in Section 5.4.4.

5.4.2.3 Steam Generator Materials

5.4.2.3.1 Selection and Fabrication of Materials

Pressure boundary materials used in the steam generator are selected and fabricated in accordance with the requirements of Section III of the ASME Code. A general discussion of materials specifications is given in Section 5.2.3, with types of materials listed in Tables 5.2-2 and 5.2-3. Fabrication of reactor coolant pressure boundary materials is also discussed in Section 5.2.3, particularly in Sections 5.2.3.3 and 5.2.3.4.

The steam generator materials are carbon steel, except for the U and J tubes, tube support plates, flow distribution baffle, antivibration bars, and the channel head divider plate. The interior surfaces of the reactor coolant channel head, nozzles, and manways are clad with austenitic stainless steel. The primary side of the tube sheet is weld clad with Inconel (ASME SFA-5.14). The U and J tubes are Inconel-600, a nickel-chromium-iron alloy (ASME SB-163). The channel head divider plate is Inconel (SB-168). Tube support plates and the flow distribution baffle are ferritic stainless steel (Type 405). The antivibration bars are Inconel-600, which is chrome plated to improve wear resistance.

The Inconel tubing has been subjected to a thermal treatment process, which has been defined on the basis of laboratory tests and which provides increased resistance to stress corrosion cracking.

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Code cases used in material selection are discussed in Section 5.2.1. The extent of conformance with Regulatory Guides 1.84 and 1.85 is discussed in Appendix 3A.

During manufacture, cleaning is performed on the primary and secondary sides for the steam generator, in accordance with written procedures which follow the guidance of Regulatory Guide 1.37 and the ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants." Onsite cleaning and cleanliness control also follow the guidance of Regulatory Guide 1.37, as discussed in Appendix 3A. Cleaning process specifications are 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 reactor coolant pressure boundary is provided by compliance with Appendix G of 10 CFR 50 and with Paragraph NB-2300 of Section III of the ASME Code. As discussed in Section 5.4.2.1, consideration of fracture toughness is only necessary for materials in Class 1 components.

5.4.2.3.2 Compatibility of Steam Generator Tubing with Primary and Secondary Coolants

As mentioned in Section 5.4.2.3.1, corrosion tests, which subjected the steam generator tubing material, Inconel-600 (ASME SB-163), to simulated steam generator water chemistry, have indicated that the loss due to general corrosion over the 40-year plant life is insignificant, compared to the tube wall thickness. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel-600 has excellent resistance to general and pitting type corrosion in severe operating water conditions. Many reactor years of successful operation have shown the same low general corrosion rates as indicated by the laboratory tests.

Recent operating experience, however, has revealed areas on secondary surfaces where localized corrosion rates were significantly greater than the low general corrosion rates. Both intergranular stress corrosion and tube wall thinning were experienced in localized areas, although not at the same location nor under the same environmental conditions (water chemistry, sludge composition).

Adoption of the all volatile treatment (AVT) chemistry control program eliminates the possibility for recurrence of the tube wall thinning phenomenon related to phosphate chemistry control. Successful AVT operation requires maintenance of low concentration of impurities in the steam generator water, thus reducing the potential for formation of highly concentrated solutions in low

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flow zones, which is the precursor of corrosion. By restriction of the total alkalinity in the steam generator and prohibition of extended operation with free alkalinity, the AVT control program minimizes the possibility for occurrence of intergranular corrosion in localized areas due to excessive levels of free caustic.

Laboratory testing has shown that the Inconel-600 tubing is compatible with the AVT environment. Isothermal corrosion testing in high purity water has shown that commercially produced Inconel-600 exhibiting normal microstructures tested at normal engineering stress levels does not suffer intergranular stress corrosion cracking in extended exposure to high temperature water. These tests also showed that no general type of corrosion occurred. A series of autoclave tests in reference secondary water with planned excursions have produced no corrosion attack after 1,938 days of testing on any as produced Inconel-600 tube samples.

AVT chemistry control has been employed successfully in plant operations for considerable periods. Plants with stainless steel tubes which have demonstrated successful AVT operation include Selni, Sena, and Yankee-Rowe. Selni has operated with AVT since 1964, Sena since 1966, and Yankee-Rowe since 1967. Approximately 20 plants with Inconel tubes have operated with AVT or limited phosphate exposure for periods up to 4 to 4-1/2 years. There have been only a few tube leaks, and annual eddy current inspections have revealed no tube thinning and virtually no corrosion-induced cracking.

Additional extensive operating data are presently being accumulated with the conversion to AVT chemistry. A comprehensive program of steam generator inspections, including the recommendations of NEI 97-06, with the exceptions as stated in Appendix 3A, will ensure detection and correction of any unanticipated degradation that might occur in the steam generator tubing.

Another corrosion-related phenomenon, termed tube denting, was first discovered during the April 1975 steam generator inspection at the Surry Unit No. 2 plant. This discovery was evidenced by eddy current signals resembling those produced by scanning dents and by difficulty in passing the standard eddy current probe through the tubes at the intersections with the support plates. Subsequent to the initial finding, steam generator inspections at other operating plants revealed indications of denting to various degrees.

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An intensive program of investigations, which has included removal of dented tubes and tube/support plate samples from affected steam generators and laboratory tests of heated crevices and model boilers, has revealed that the source of tube denting is corrosion of the carbon steel tube support plate (TSP) in the crevices between the tube and TSP. The corrosion rate in these locations is apparently accelerated by deposition of impurities from the secondary fluid, caused by low flow velocity and superheated fluid in the crevice. The corrosion product has a larger volume than the base metal. The results are simultaneous reduction of the tube diameter, dilation of the hole in the TSP, and secondary effects (e.g., TSP distortions) related to dilation of the TSP holes. Denting has been most pronounced in plants having a history of chloride contamination resulting from condenser leakage. The presence of acid chloride has been found to be a common factor in tube denting produced in laboratory tests. Measures to inhibit denting concentrate on providing a more corrosion resistant TSP material and on eliminating conditions conducive to corrosion at the tube support locations (e.g., chemical impurities in the secondary fluid and localized superheat).

The tube support plates and flow distribution baffle used in the Model F steam generator are Type 405 ferritic stainless steel which has been shown in laboratory tests to be resistant to corrosion in the AVT environment. When corrosion of ferritic stainless steel does occur, the volume of the corrosion products is equivalent to the volume of the parent material. Thus, substitution of Type 405 ferritic stainless steel for carbon steel used in previous steam generators substantially reduces the potential for tube denting.

Other features of the Model F generator further reduce the potential for tube denting. The quatrefoil geometry of the tube support plates is less susceptible to the accumulation of corrosion products which cause tube denting. The quatrefoil geometry also results in a reduced fluid pressure drop across the tube support plates and, therefore, a higher recirculation ratio and higher fluid velocities in the tube bundle. The flow distribution baffle serves to provide higher cross-flow velocity immediately above the tube sheet and to sweep sludge to the center of the tube bundle, where the intakes to the blowdown pipes are located. Increased capacity (90 gpm per steam generator) blowdown pipes have been added. High volume blowdown provides protection against inleakage of impurities from the condenser and feedwater system.

Blocking devices located adjacent to the downcomer region and at the innermost U-bend tube row, at the tube sheet, minimize bypass flow, promoting flow into the central regions of the bundle.

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Operating experience, verified in numerous steam generator inspections, indicates that the tube degradation associated with phosphate water treatment is not occurring where only AVT has been utilized. Adherence to the AVT chemical specifications and close monitoring of the condenser integrity will assure the continued good performance of the steam generator tubing.

5.4.2.3.3 Control of Secondary-Side Impurities

Several provisions exist in the WCGS plants to limit the accumulations of impurities in the steam generator, either by limiting ingress or by facilitating removal. The materials of construction of the secondary system are such as to minimize the formation of corrosion products. The materials include stainless steel tubing in all feedwater heaters and Corten tubing in the moisture-separator-reheaters. A full-flow condensate demineralizer system is provided. A piping connection is provided from the feedwater heater, ahead of the steam generators, to the condenser hot well. During startup, this connection is used to circulate secondary system water through the condensate demineralizers. The flow circulation removes suspended corrosion products that may have accumulated during extended shutdowns.

For removal of impurities, the blowdown system has a capacity slightly in excess of 1 percent of full-load feedwater flow. As described in Section 5.4.2.2 and 5.4.2.3.2, the design of the Model F steam generator is expected to result in an increased efficiency of impurity removal by the blowdown system.

The feedwater system materials are discussed in Section 10.4.7, the steam generator blowdown system is discussed in Section 10.4.8, and the condensate demineralizer system is discussed in Section 10.4.6. Instrumentation to monitor secondary side water chemistry is described in Section 9.3.2.

During shutdowns, sludge lancing may be used to remove accumulated material. In sludge lancing, a hydraulic jet is inserted through an access opening (handhole) to loosen sludge deposits, which are removed by means of a suction pump.

5.4.2.4 Steam Generator Inservice Inspection

The steam generator and associated insulation is designed to permit inspection of Class 1 and 2 parts, including individual tubes. The design includes a number of openings to provide access to both the primary and secondary sides of the steam generator, and the inspection program followed complies with Section XI of the ASME Code, including addenda per 10 CFR 50.55a (g) with certain exceptions whenever specific written relief is granted by the

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NRC per 10 CFR 50.55a (g) (6). These openings include four manways, two for access to both chambers of the reactor coolant channel head inlet and outlet sides and two in the steam drum for inspection and maintenance of the moisture separators, and six 6-inch handholes, three located just above the tube sheet secondary surface and three located just above the flow distribution baffle. Access to the tube U-bend is provided through each of the three deck plates. For proper functioning of the steam generator, some of the deck plate openings are covered with welded, but removable, hatch plates. Inspection/access to the primary side is provided by two 16-inch manways located in the channel head.

In addition, a separate preservice and inservice inspection document which complies with the recommendations of Regulatory Guide 1.83 and "NRC Staff Guidance for complying with certain provisions of 10 CFR 50.55a (g) Inservice Inspection Requirements" was submitted to the NRC. This document provided the details to the areas subject to examination, method of examination, extent of examination, and frequency. WCGS now uses the guidance set forth in NEI 97-06 to monitor Steam Generator integrity.

The insulation in the area of longitudinal and circumferential welds, including tube-sheet-to-head or shell welds, primary nozzle-to-vessel head welds and nozzle-to-head inside radiused sections; primary nozzle-to-safe end welds; integrally welded vessel supports, circumferential butt welds, and nozzle-to-vessel welds on the secondary side is removable. The pressure-retaining bolting can be removed for examination. Manways in the primary head allow direct visual examination of the head cladding. The manways allow sufficient access for the installation of the remotely operated eddy current equipment capable of performing inservice inspections in accordance with the recommendations given in NEI 97-06.

5.4.2.4.1 Compliance with Section XI of the ASME Code

Eddy current examinations of steam generator tubing are performed in accordance with Section XI of the ASME Code per 10 CFR 50.55a(g), with certain exceptions whenever specific written relief is granted by the NRC per 10 CFR 50.55a, and the WCGS Technical Specifications.

Other Class 1 and Class 2 components of the steam generators are examined in accordance with the inservice inspection program. The inservice inspection program of Class 1 components of the steam generators is described in Section 5.2.4. The inservice inspection of Class 2 components of the steam generators is discussed in Section 6.6.

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5.4.2.4.2 Program for Inservice Inspection of Steam Generator Tubing

Steam generator tubing is inspected in accordance with the recommendations given in NEI 97-06, as discussed in Appendix 3A. This guide covers the inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, and required actions based on findings. Variations in the type of equipment and calibration material are approved for use through utilization of ASME Section XI Code Cases. The Cases utilized are included in the inservice inspection subtier program document addressing steam generator tubing inspection, as discussed in USAR Appendix 3A for Regulatory Guide 1.147. The design of the steam generators permits inservice inspection and/or plugging, if required, of each tube. Regulatory Guide 1.121 provides recommendations concerning tube plugging.

The eddy current examination equipment and procedures are capable of detecting and locating defects with a penetration of 20 percent or more of the wall thickness. The remotely operated equipment is capable of examining the entire length of the tubes.

All original examination data, results, and reports are stored in a fireproof facility and in an atmosphere controlled to minimize deterioration. The data is stored in a limited-access facility and retained for the operating life of the plant.

Standards consisting of similar as-manufactured steam generator tubing with known imperfections are used to establish sensitivity and to calibrate the equipment. Where practical, these standards include reference flaws that simulate the length, depth, and shape of actual imperfections that are characteristic of past experience.

Personnel engaged in taking or interpreting data are tested and qualified in accordance with American Society for Nondestructive Testing Standard SNT-TC-1A and supplements designated by the Edition and Addenda of Section XI used during the examination. Procedures governing the above examinations are qualified prior to examination in the plant.

All of the tubes in the steam generators are inspected by eddy current prior to service to establish a baseline condition of the tubing.

The sample selection and testing of tubes, the inspection intervals, and the actions to be taken if defects are identified follow the recommendations of NEI 97-06.

5.4.2.5 Design Evaluation

Seismic and LOCA loads are discussed in Section 3.9(N).

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5.4.2.5.1 Forced Convection of Reactor Coolant

The limiting case for heat transfer capability is the "nominal 100-percent design" case. The steam generator effective heat transfer coefficient is based on the coolant conditions of temperature and flow for this case. The best estimate for the heat transfer coefficient applied in steam generator design calculations and plant parameters selection is 1503 Btu/hr-ft²-F. The coefficient incorporates a specified fouling factor resistance of 0.00005 hr-ft²-F/Btu, which is the value selected to account for the differences in the measured and calculated heat transfer performance as well as provide the margin indicated above. Although margin for tube fouling is available, operating experience to date has not indicated that steam generator performance decreases over a long-time period. Adequate tube area is selected to ensure that the full design heat removal rate is achieved.

5.4.2.5.2 Natural Circulation of Reactor Coolant

The driving head created by the change in coolant density as it is heated in the core and rises to the outlet nozzle initiates convection circulation. This circulation is enhanced by the fact that the steam generators, which provide a heat sink, are at a higher elevation than the reactor core, which is the heat source. Natural circulation is sufficient for the removal of decay heat during hot shutdown and cooldown in the event of a loss of forced circulation.

5.4.2.5.3 Mechanical and Flow-Induced Vibration Under Normal Operation

The possibility of vibratory failure of tubes due to either mechanical or flow-induced excitation has been thoroughly evaluated. This evaluation includes detailed analysis of the tube support systems as well as an extensive research program with tube vibration model tests.

In evaluating possible failure due to vibration, consideration is given to such sources of excitation as those generated by the primary fluid flowing within the tubes. The effects of these as well as any other mechanically induced vibrations are considered to be negligible and should cause little concern.

Another source of possible vibratory failure in heat exchanger components is hydrodynamic excitation by the secondary fluid on the outside of the tubes.

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Consideration of secondary flow-induced vibration involves two types of flow, parallel and cross, and it is evaluated in three regions:

- a. At the entrance of the downcomer feed to the tube bundle (cross flow)
- b. Along the straight sections of the tube (parallel flow)
- c. In the curved tubed section of the U-bend (cross flow)

For the case of parallel flow, analysis is done to determine the vibratory deflections in order to verify that the flow velocities are sufficiently below those required for damaging fatigue or impacting vibratory amplitude. Thus, the support system is deemed adequate to preclude parallel flow excitation.

For the case of cross-flow excitation, several possible mechanisms of tube vibration exist. For the Model F steam generator design and conditions, only two of these mechanisms are deemed significant enough to merit extensive consideration: 1) Von Karman vortex shedding and 2) fluidelastic vibration. The steam generator is analyzed to ensure that the tube natural frequency is well above the anticipated vortex shedding frequency and that unstable fluidelastic vibration does not exist. In order to achieve this, adequate tube supports must be provided. An evaluation using the specific parameters for the Model F steam generator confirms the integrity of the support system.

To provide added strength as well as resistance to vibration, the quatrefoil tube support plate thickness has been increased. In addition, 12 peripheral supports also provide stability to the plates so that tube fretting or wear due to flow-induced plate vibrations at the tube support contact regions is abated.

Assurance against damaging flow induced tube vibration has been accomplished by a combination of analysis and testing. Cross and parallel flow velocities were calculated from thermal-hydraulic analysis of the secondary flow. Three possible vibrational mechanisms, vortex shedding, fluid-elastic excitation, and turbulence were studied.

For vortex shedding, resonance conditions were conservatively assumed, and amplitudes for different resonant modes were computed.

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For fluidelastic excitation, tubes that are unsupported by an anti-vibration bar (AVB) and contrary to design requirements, or tubes that are subject to significant flow peaking due to non-uniform insertion of the AVBs, were evaluated to determine if they are subject to possible fatigue failure during the lifetime of the steam generators. The analysis methodology is the same as the methodology used to satisfy the analysis requirements of NRC Bulletin 88-02. The analysis is described in WCAP-17990-P, "Wolf Creek U-Bend Vibration and Fatigue Assessment." Seventeen tubes were identified in the analysis that may be subject to fatigue failure based on the pinned and non-occluded case. These 17 tubes were all plugged (removed from service) during Refuel 20. All other tubes were shown to be acceptable for a 60 year operating lifetime of Wolf Creek (40 years, plus period of extended operation).

The amplitudes of turbulence-induced vibration are one order of magnitude less than those from vortex-shedding induced vibration. Therefore, vortex shedding is considered the predominant mechanism of flow-induced tube vibration. Combining both vortex shedding and turbulence effects in a conservative manner, the maximum predicted local tube wear depth over 40 years of operational life is less than 0.006 inches. This value is considerably below the limiting wall thickness reduction for a Model F steam generator tube.

5.4.2.5.4 Allowable Tube Wall Thinning Under Accident Conditions

An evaluation is performed to determine the extent of tube wall thinning that can be tolerated under accident conditions. The worst-case loading conditions are assumed to be imposed upon uniformly thinned tubes, at the most critical location in the steam generator. Under such a postulated design basis accident, vibration is of short enough duration that there is no endurance problem to be considered. The steam generator tubes, existing originally at their minimum wall thickness and reduced by a conservative general corrosion and erosion loss, can be shown to provide an adequate safety margin, that is, sufficient wall thickness, in addition to the minimum required for a maximum stress less than the allowable stress limit, as it is defined by the ASME Code.

The results of a study made on "D series" (0.75 inch nominal diameter, 0.043 inch nominal wall thickness) tubes under accident loadings are discussed in Reference 3. These results demonstrate that a minimum wall thickness of 0.026 inches would have a maximum faulted condition stress (i.e., due to combined LOCA and SSE loads) that is less than the allowable limit. This thickness is 0.010 inch less than the minimum "D series" tube wall thickness of 0.039 inch, which is reduced to 0.036 inch by the assumed general corrosion and erosion rate. Thus, an adequate safety margin is exhibited. The corrosion rate is based on a conservative weight loss rate for Inconel tubing in flowing 650 F primary side reactor coolant fluid. The weight loss, when equated to a thinning rate and projected over a 40-year plant life with appropriate reduction after initial hours, is equivalent to 0.083 mil thinning. The assumed corrosion rate of 3 mils leaves a conservative 2.917 mils for general corrosion thinning on the secondary side.

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The Model F steam generator is analyzed, using similar assumptions of general corrosion and erosion rates. The overall similarity between the tubes studied and the Model F tubes makes it reasonable to expect the same general results, that is, to conclude that the ability of the Model F steam generator tubes to withstand accident loading is not impaired by a lifetime of general corrosion losses. The results of the specific analysis are presented in WCAP 10043, "Steam Generator Tube Plugging Analysis for the Westinghouse Standardized Nuclear Unit Power Plant System (SNUPPS)." Wolf Creek uses the SNUPPS design.

5.4.2.6 Quality Assurance

The steam generator nondestructive examination 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.

Liquid penetrant inspection is performed on weld deposited tube sheet cladding, channel head cladding, divider plate to tube sheet and to channel head weldments, tube-to-tube sheet weldments, and weld deposit cladding. Liquid penetrant inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code.

Magnetic particle inspection is performed on the tube sheet forging, channel head casting, nozzle forgings, and the following weldments:

- a. Nozzle to shell
- b. Support brackets
- c. Instrument connection (secondary)
- d. Temporary attachments after removal
- e. All accessible pressure retaining welds after hydrostatic test

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

Ultrasonic tests are performed on the tube sheet forging, tube sheet cladding, secondary shell and head plate, and nozzle forgings.

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

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

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 Code, Section III, Class 1 requirements.

Stainless steel pipe conforms to ANSI B36.19 for sizes 1/2 inch through 12 inches and wall thickness Schedules 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 no less than those calculated using the ASME Code, Section 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, and ovality does not exceed 6 percent.

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

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.

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5.4.3.2 Design Description

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

- a. Charging line and alternate charging line from the system isolation valve up to the branch connections on the reactor coolant loop
- b. Letdown line and excess letdown line from the branch connections on the reactor coolant loop to the system isolation valve
- c. Pressurizer spray lines from the reactor coolant cold legs to the spray nozzle on the pressurizer vessel
- d. Residual heat removal lines to or from the reactor coolant loops up to the designated check valve or isolation valve
- e. Safety injection lines from the designated check valve to the reactor coolant loops
- f. Accumulator lines from the designated check valve to the reactor coolant loops
- g. Loop fill, loop drain, sample*, and instrument* lines to or from the designated isolation valve to or from the reactor coolant loops
- h. Pressurizer surge line from one reactor coolant loop hot leg to the pressurizer vessel inlet nozzle
- i. Resistance temperature detector scoop element, pressurizer spray scoop, sample connection* with scoop, reactor coolant temperature element installation boss, and the temperature element well itself

* Lines with a 3/8-inch (liquid service), 3/4-inch (steam service), or less flow restricting orifice qualify as Safety Class 2.

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- j. All branch connection nozzles attached to reactor coolant loops.
- k. Pressure relief lines* from nozzles on top of the pressurizer vessel up to and through the power operated pressurizer relief valves and pressurizer safety valves
- l. Seal injection water lines to the reactor coolant pump from the designated check valve (injection line)
- m. Auxiliary spray line from the isolation valve to the pressurizer spray line header
- n. Sample lines* from pressurizer to the isolation valve
- o. Reactor vessel head vent lines* to the isolation valves

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

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

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. Pipe and fittings are cast, seamless without longitudinal or electroslag welds, and comply with the requirements of the ASME Code, Section II (Parts A and C), Section III, and Section IX. All smaller piping which is part of the RCS, such as the pressurizer surge line, spray and relief line, loop drains and connecting lines to other systems, are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. All joints and connections are welded, except for the pressurizer code safety valves, where flanged joints are used. A thermal sleeve is installed on the pressurizer spray line nozzle.

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

- * Lines with a 3/8-inch (liquid service), 3/4-inch (steam service), or less flow restricting orifice qualify as Safety Class 2.

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- a. Residual heat removal 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, should this be required for maintenance.
- b. Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation as shown on Figure 5.1-1, Sheet 1.
- c. The differential pressure taps for flow measurement, which are downstream from the steam generators of the first 90-degree elbow as shown on Figure 5.1-1, Sheet 1.
- d. The pressurizer surge line, which is attached at the horizontal centerline is shown on Figure 5.1-1, Sheet 2.
- e. Two of the three scoops in each resistance temperature detector hot leg connection.
- f. The hot leg sample connections, the loop 3 thermowell, and the loop 4 boron injection tank injection connection, all located on the horizontal center-line.

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

- a. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.
- b. The reactor coolant sample system taps protrude into the main stream to obtain a representative sample of the reactor coolant.
- c. The hot leg connections to the resistance temperature detectors have scoops which extend into the reactor coolant to collect a representative temperature sample for the individual hot leg resistance temperature detector.
- d. The wide range temperature detectors are located in resistance temperature detector wells that extend into both the hot and cold legs of the reactor coolant pipes.

One hot leg and one cold leg temperature reading are provided from each coolant loop to use for protection. Narrow range, thermowell-mounted Resistance Temperature Detectors (RTDs) are provided for each coolant loop. In the hot legs, sampling scoops are used because the flow is stratified. That is, the fluid temperature is not uniform over a cross section of the hot leg.

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One dual element RTD is mounted in a thermowell in each of the three sampling scoops associated with each hot leg. The scoops extend into the flow stream at locations 120° apart in the cross sectional plane. Each scoop has five orifices which sample the hot leg flow along the leading edge of the scoop. Outlet ports are provided in the scoops to direct the sampled fluid past the sensing element of the RTDs. One of each of the RTD's dual elements is used while the other is an installed spare. Three readings from each hot leg are averaged to provide a hot leg reading for that loop.

One dual element RTD is mounted in a thermowell associated with each cold leg. One RTD element is used while the other is an installed spare.

The thermowells are pressure boundary parts which completely enclose the RTD. They have been shop hydrotested to 1.25 times the RCS design pressure. The external design pressure and temperature are the RCS design temperature and pressure. The RTD is not part of the pressure boundary. The scoop, thermowell, and thermowell/scoop assembly have been analyzed to the ASME Boiler and Pressure Vessel Code, Section III, Class 1. The effects of seismic and flow-induced loads were considered in the design.

Signals from the temperature detectors are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{HOT} minus the temperature of the cold leg, T_{COLD}) and an average reactor coolant temperature (T_{AVG}). The T_{AVG} for each loop is indicated on the main control board.

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(N).

5.4.3.3.1 Material Corrosion/Erosion Evaluation

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

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Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality listed in Table 5.2-5. Maintenance of the water quality to minimize corrosion is accomplished, using the chemical and volume control system and sampling system which are described in Chapter 9.0.

Components in the Reactor Coolant System were designed to provide access to permit inservice inspection in accordance with the ASME Code, Section XI. Pursuant to this, all pressure containing welds out to the second valve that delineates the RCS boundary are accessible for examination and are fitted with removable insulation.

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. Thread lubricants are approved in accordance with applicable procedures. Prior to application of thermal insulation, the austenitic stainless steel surfaces are cleaned and analyzed to halogen limits as defined by Westinghouse Process Specifications.

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 all pipe 27-1/2 inches 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.

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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 outlet nozzle of the steam generator is provided with a flow restrictor designed to limit steam flow in the unlikely event of a break in the main steam line. A large increase in steam flow will create a backpressure which limits further increase in flow. The flow restrictor performs the following functions: rapid rise in containment pressure is prevented, the rate of heat removal from the reactor coolant is such as to keep the cooldown rate within acceptable limits, thrust forces on the main steam line piping are reduced, and stresses on internal steam generator components, particularly the tube sheet and tubes, are limited. The restrictor is configured to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

The flow restrictor consists of seven Inconel (ASME SB-163) venturi inserts which are installed in holes in an integral 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 inserts are welded to the Inconel cladding on the inner surface of the forging.

5.4.4.3 Design Evaluation

The flow restriction design has been analyzed to assure its structural adequacy. The equivalent throat diameter of the steam generator outlet is 16 inches, and the resultant pressure drop through the restrictor at 100-percent steam flow is approximately 3.4 psig. This was based on a design flow rate of 3.79E6 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

Since the restrictor is not a part of the steam system boundary, no tests and inspection beyond those during fabrication are anticipated.

5.4.5 MAIN STEAM LINE ISOLATION SYSTEM

The main steam line isolation system is discussed in Section 10.3.

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

This section is not applicable to WCGS.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The residual heat removal system (RHRS) functions to remove heat from the RCS when RCS pressure and temperature are below approximately 425 psig and 350°F, respectively. Heat is transferred from the RHRS to the component cooling water system.

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The design of the RHRS includes two motor-operated isolation valves that are closed during normal operations. They are provided with both a "prevent-open" interlock and "RHRS-Iso-Valve-Open" alarm which are designed to prevent possible exposure of the RHRS to normal RCS operating pressure.

The isolation valves are opened for residual heat removal during a plant cooldown after the RCS temperature is reduced to approximately 350°F and RCS pressure is less than approximately 360 psig in accordance with plant procedures. During a plant startup, the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above approximately 425 psig (alarm setpoint).

Portions of the RHRS also serve as portions of the ECCS during the injection and recirculation phases of a LOCA (see Section 6.3).

The RHRS also is used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of the refueling operations. The RHRS is designed to be isolated from the RCS whenever the RCS pressure exceeds the RHRS design pressure.

5.4.7.2 Design Description

5.4.7.2.1 Functional Design

RHRS design parameters are listed in Table 5.4-7. Nuclear plants employing the same RHRS design as the WCGS unit are given in Section 1.3.

During normal approaches to cold shutdown, the RHRS is placed in operation approximately 4 hours after reactor shutdown when the temperature and pressure of the RCS are approximately 350°F and 360 psig, respectively. Only one train of RHR is placed into operation initially to reduce the RCS temperature from 350°F to $\leq 225^\circ\text{F}$ when the other train of RHR is utilized. This sequence is necessary to safeguard a train of RHR for ECCS requirements when shutdown. This sequence and temperature restriction is due to limiting the temperature of RCS fluid allowed in the RHR pump suction piping. The temperature of RCS fluid allowed in at least one train of RHR suction piping is conservatively kept by plant procedures below the saturation temperature for the static head pressure of the RWST to avoid vaporization should the train be realigned to the RWST for shutdown LOCA mitigation. Assuming both trains of RHR operating in accordance with this sequence with a maximum service water temperature of 90°F, plant cooldown is completed in 17.9 hours following reactor shutdown (RCS temp $< 140^\circ\text{F}$). This cooldown rate is based on throttling RHR flow, as necessary, to maintain a maximum 120°F component cooling water to the shell side of the RHR heat exchangers and to limit the RCS cooldown rate to a maximum of 50°F/hr. The heat load handled by the RHRS during the cooldown transient includes residual and decay heat from the core and reactor coolant pump heat. The design heat load is based on the decay heat fraction that exists at 20 hours using the ANSI/ANS-5.1-1979 Decay heat standard, following reactor shutdown from an extended run at full power.

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Assuming that only one heat exchanger and pump are in service and that the heat exchanger is supplied with component cooling water at design flow and temperature, the RHRS is capable of reducing the temperature of the reactor coolant from 350°F to 200°F within 32 hours after shutdown.

The RHRS is isolated from the RCS on the suction side by two motor-operated valves in series on each suction line. Each motor-operated valve is interlocked to prevent its opening if RCS pressure is greater than approximately 360 psig. During plant startup, operator action is required to close the RHRS suction-isolation valves. An alarm will actuate on the Main Control Board if RHRS isolation valves are not fully closed in conjunction with RCS high pressure. The alarm setpoint pressure will be within the range of open permissive setpoint pressure, and RHR system design pressure minus RHR pump head pressure. P (open permissive setpoint) < P (alarm setpoint) < [P (RHR system design pressure) - P (pump discharge head)]. This interlock and alarm function is described in more detail in Sections 5.4.7.2.5 and 7.6.2. 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, motor-operated valve downstream of each RHRS heat exchanger. (These check valves and motor-operated valves are not considered part of the RHRS. They are shown as part of the ECCS, see Figures 5.1-1, 5.4-7, and 6.3-1.)

Each inlet line to the RHRS is equipped with a pressure relief valve designed to relieve the combined flow of all the charging pumps at the relief valve set pressure. These relief valves also protect the RHRS system from inadvertent overpressurization during plant cooldown or startup. Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve designed to relieve the maximum possible backleakage through the valves isolating the RHRS from the RCS.

The RHRS is provided for WCGS which is a single nuclear power unit.

The RHRS is designed to be fully operable from the control room for normal operation. Manual operations required of the operator are: opening the suction isolation valves, positioning the flow control valves downstream of the RHRS heat exchangers, and starting the residual heat removal pumps. By nature of its redundant two-train design, the RHRS is designed to accept major component single failures with the only effect being an extension in the required cooldown time. For two low probability electrical system single failures, i.e., failure in the suction isolation valve interlock circuitry or diesel generator failure in conjunction with loss of offsite power, operator action outside the control room is required to open the suction isolation valves. Manual actions are discussed in further detail in Sections 5.4.7.2.7 and 5.4.7.2.8. The motor-operated valves in the RHRS are not subject to flooding. Spurious operation of a single motor-operated valve can be accepted without loss of function, as a result of the redundant two-train design.

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

5.4.7.2.2 Piping and Instrumentation Diagrams

The RHRS, as shown in Figures 5.4-7 (piping and instrumentation diagram) and 5.4-8 (process flow diagram), consists of two residual heat exchangers, two residual heat removal 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 two reactor coolant loops, while the return lines are connected to the cold leg of each of the reactor coolant loops. These return lines are also the ECCS low head injection lines (see Figure 6.3-1).

The RHRS suction lines are isolated from the RCS by two motor-operated valves in series located inside the containment. Each discharge line is isolated from the RCS by two check valves in series located inside the containment and by a normally open motor-operated valve located outside the containment. (The check valves and the motor-operated valve on each discharge line are shown as part of the ECCS, see Figures 5.1-1, 5.4-7, and 6.3-1.)

During RHRS operation, reactor coolant flows from the RCS to the residual heat removal pumps, through the tube side of the residual heat exchangers, and back to the RCS. The heat is transferred to the component cooling water circulating through the shell side of the residual heat exchangers.

Coincident with operation of the RHRS, a portion of the reactor coolant flow may be diverted from downstream of the residual heat exchangers to the chemical and volume control system (CVCS) low pressure letdown line for cleanup and/or pressure control. By regulating the diverted flowrate and the charging flow, the RCS pressure may be controlled. Pressure regulation is necessary to maintain the pressure range dictated by the fracture prevention criteria requirement of the reactor vessel, by the number 1 seal differential pressure, and by net positive suction head requirements of the reactor coolant pumps.

The RCS cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the RHR heat exchangers. The flow control valve in the bypass line around each RHR heat exchanger automatically maintains a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature, and total flow.

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The RHRS may be used for filling the refueling cavity before refueling. After refueling operations, water is pumped back to the refueling water storage tank until the water level is brought down to two feet above the flange of the reactor vessel. The remainder of the water 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 nuclear sampling system to extract samples from the flow of reactor coolant downstream of the residual heat exchangers. A local sampling point is also provided on each residual heat removal train between the pump and heat exchanger.

The RHRS functions in conjunction with the high head portion of the ECCS to provide direct injection of borated water from the refueling water storage tank into the RCS cold legs during the injection phase following a LOCA. During normal operation, the RHRS is aligned to inject borated water upon receipt of a safety injection signal.

In its capacity as the low head portion of the ECCS, the RHRS also provides long-term recirculation capability for core cooling following the injection phase of a LOCA. This function is accomplished by aligning the RHRS to take fluid from the containment sump, cool it by circulation through the residual heat exchangers, and supply it to the core directly as well as via the centrifugal charging pumps and safety injection pumps.

The use of the RHRS as part of the ECCS is more completely described in Section 6.3.

The RHR pumps, in order to perform their ECCS function, are interlocked to start automatically on receipt of a safety injection signal (see Section 6.3).

The RHR suction isolation valves are also interlocked to prevent their being opened unless the isolation valves in the following lines are closed:

- a. Recirculation lines from the residual heat exchanger outlets to the suctions of the safety injection pumps and centrifugal charging pumps
- b. RHR pump suction lines from the refueling water storage tank
- c. RHR pump suction lines from the containment sump

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The motor-operated valves in the RHR miniflow bypass lines are interlocked to open when the RHR pump discharge flow is less than approximately 816 gpm at 300°F (783 gpm at 68°F) and close when the flow exceeds approximately 1650 gpm at 300°F (1582 gpm at 68°F).

5.4.7.2.3 Equipment and Component Descriptions

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

Residual Heat Removal Pumps

Two pumps are installed in the RHRS. The pumps are sized to deliver reactor coolant flow through the RHR heat exchangers to meet the plant cooldown requirements. The availability of two separate RHR trains assures that cooling capacity is only partially lost should one pump become inoperative.

The RHR pumps are protected from overheating and loss of discharge flow by miniflow bypass lines. A valve located in each miniflow line is regulated by a signal from the flow transmitters located in each pump discharge header. The control valves open when the residual pump discharge flow is less than approximately 816 gpm at 300°F (783 gpm at 68°F) and close when the flow exceeds approximately 1650 gpm at 300°F (1582 gpm at 68°F).

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.

The two 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.

The RHR pumps also function as the low head safety injection pumps in the ECCS (see Section 6.3 for further information and for the residual heat removal pump performance curves).

Residual Heat Exchangers

Two residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and temperature differences between reactor coolant and component cooling water

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existing 20 hours after reactor shutdown when the temperature difference between the two systems is small.

The availability of two heat exchangers in separate and independent residual heat removal trains assures that the heat removal capacity of the system is only partially lost if one train becomes inoperative.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

The residual heat exchangers also function as part of the ECCS (see Section 6.3).

Residual Heat Removal System Valves

Valves that perform a modulating function are equipped with graphite packing.

Manual and motor-operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions.

Encapsulation

The RHR suction lines from the containment recirculation sumps are each provided with a single motor-operated gate valve outside the containment. This valve, including its operator, is encapsulated in a pressure vessel which is leaktight at containment design pressure. The piping from the sump to the valve is also encapsulated in a concentric guard pipe which is leaktight. A leaktight seal is provided such that the ambient inside the pressure vessel and outside the process line and enclosed within the guard pipe is not directly connected with the containment sump or containment atmosphere. Component parameters for the encapsulation tank are given in Table 5.4-8.

The valve provides a barrier outside the containment to prevent loss of sump water should a leak develop in the recirculation loop. Should a leak develop in the valve body or in the pipe between the valve and the sump, the sump fluid is contained by the leaktight seal and/or by the guard pipe.

With this system, no single failure of either an active or a passive component will prevent the recirculation phase or adversely affect the integrity of the containment.

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5.4.7.2.4 System Operation

Reactor Startup

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

At initiation of the plant startup, the RCS is completely filled, and the pressurizer heaters are energized. The RHRS is operating and is connected to the CVCS via the low pressure letdown line for purification and/or to control reactor coolant pressure. During this time, the RHRS acts as an alternate letdown path. The manual valves downstream of the residual heat exchangers leading to the letdown line of the CVCS are opened. The control valve in the line from the RHRS to the letdown line of the CVCS is then manually adjusted in the control room to permit letdown flow.

After the reactor coolant pumps are started, pressure control via the RHRS and the low pressure letdown line is continued until the pressurizer steam bubble is formed. Indication of steam bubble formation is provided in the control room by the damping out of the RCS pressure fluctuations and by pressurizer level indication. The RHRS is then isolated from the RCS, the residual heat removal pumps are stopped, and the system pressure is controlled by normal letdown and the pressurizer spray and pressurizer heaters.

Power Generation and Hot Standby Operation

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

Normal Reactor Cooldown

Reactor cooldown is defined as the operation which brings the reactor from no-load temperature and pressure to cold conditions.

The initial phase of reactor cooldown is accomplished by transferring heat from the RCS to the steam generators then to the steam and power conversion system. The heat is removed by dumping steam to the condenser (turbine bypass system), or to the atmosphere (atmospheric relief valves).

When the reactor coolant temperature and pressure are reduced to approximately 350°F and 360 psig, approximately 4 hours after reactor shutdown, the second phase of cooldown starts and the RHRS may be placed in operation. The steam and power conversion system may continue to be used to cool the steam generators and establish refueling or maintenance conditions in a more expedient time frame.

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Startup of the RHRS includes a warmup period of one train of RHR at 350°F followed by the other train of RHR at $\leq 225^\circ\text{F}$. During the warmup time reactor coolant flow through the heat exchanger 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 residual heat exchangers. By adjusting the control valves downstream of the residual heat exchangers, the mixed mean temperature of the return flows is controlled. Coincident with the manual adjustment, each heat exchanger 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 component cooling water system and steam dump cooldown/atmospheric relief valve position. To maintain reactor cooldown rates as the reactor coolant temperature decreases, the reactor coolant flow through the residual heat exchangers is increased by adjusting the control valve in each heat exchanger's tube side outlet line and/or opening the steam dump cooldown/atmospheric relief valves further.

As cooldown 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 the reactor coolant pressure is reduced and the temperature is 140°F or lower, the RCS may be opened for refueling or maintenance.

Refueling

One of the two residual heat removal pumps may be utilized during refueling to pump borated water from the refueling water storage tank to the refueling cavity. During this operation, the RHRS isolation valve in the suction line from the RCS is closed, and the suction isolation valve from the refueling water storage tank is opened.

After the water level reaches the normal refueling level, the RHRS suction isolation valve for the RCS is opened, the refueling water storage tank supply valve is closed, and residual heat removal is resumed if needed for RCS cooling.

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

Following refueling, the RHR pumps are used to drain the refueling cavity down to two feet above the top of the reactor vessel flange by pumping water from the RCS to the refueling water storage tank. The vessel head is then replaced and the normal RHRS flowpath re-established. The remainder of the water is removed from the refueling canal via a drain connection in the bottom of the canal.

5.4.7.2.5 Control

Each inlet line to the RHRS is equipped with a pressure relief valve conservatively sized to relieve the combined flow of all the charging pumps at the relief valve set pressure; however, maximum flow through the valves is expected to be the flow of one centrifugal charging pump at its maximum delivery rate. These relief valves also protect the system from inadvertent overpressurization during plant cooldown or startup. Each valve has a relief flow capacity of 986 gpm at a set pressure of 450 psig.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve any backleakage through the valves separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gpm at a set pressure of 600 psig. These relief valves are located in the RHRS (see Figure 5.4-7).

The fluid discharged by the suction side relief valves is collected in the pressurizer relief tank. The fluid discharged by the discharge side relief valves is collected in the recycle holdup tank of the boron recycle system.

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 operations, and are provided with both a "prevent-open" interlock and "RHRS-Iso-Valve-Open" alarm which are designed to prevent possible exposure of the RHRS to normal RCS operating pressure.

The isolation valves on one train of RHR are opened for residual heat removal during a plant cooldown after the RCS temperature is reduced to below 350°F and at ≤225°F the other train valves are opened. The isolation valves are separately and independently interlocked with pressure signals to prevent their being opened whenever the RCS pressure is greater than approximately 360 psig.

During a plant startup, the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above approximately 425 psig (alarm setpoint). Each inlet isolation valve will provide alarm indication on the main control board if the valve remains open above the alarm setpoint.

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The use of two independently powered, motor-operated valves in each of the two inlet lines, along with two independent pressure interlock signals for each function, assures a design which meets applicable single failure criteria. Not only more than one single failure but also different failure mechanisms must be postulated to defeat the function of preventing possible exposure of the RHRS to normal RCS operating pressure. These protective interlock designs and alarms, in combination with plant operating procedures and alarms, 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-green position indicator lights on the main control board.

Isolation of the low pressure RHRS from the high pressure RCS is provided on the discharge side by two check valves in series. These check valves are located in the ECCS and RCS, and their testing is described in Section 6.3.4.2. |

5.4.7.2.6 Applicable Codes and Classifications

The entire RHRS is designed as Safety Class 2, with the exception of the suction isolation valves, which are Safety Class 1. Class 1 discharge valves are discussed in Section 6.3. Component codes and classifications are given in Section 3.2.

5.4.7.2.7 System Reliability Considerations

General Design Criterion 34 requires that a system to remove residual heat be provided. The safety function of this required 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, both nuclear steam supply system (NSSS) scope and balance-of-plant (BOP) scope, to perform this function. The NSSS scope safety grade systems which perform this function for all plant conditions except a LOCA are: the RCS and steam generators, which operate in conjunction with the auxiliary feedwater system and the steam generator safety and Atmospheric Relief Valves; and the RHRS, which operates in conjunction with the component cooling water and service water systems. The BOP scope safety grade systems which perform this function for all plant conditions, except a LOCA, are: the auxiliary feedwater system; the steam generator safety and Atmospheric Relief Valves, which operate in conjunction with the RCS and the steam generators; and the component cooling water and service water systems, which operate in conjunction with the RHRS. For LOCA conditions, the safety grade system which performs

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the function of removing residual heat from the reactor core is the ECCS, which operates in conjunction with the component cooling water system and the essential service water system.

The auxiliary feedwater system, along with the steam generator safety and Atmospheric Relief Valves, provides a completely separate, independent, and diverse means of performing the safety function of removing residual heat, which is normally performed by the RHRS when RCS temperature is less than 350°F.

The auxiliary feedwater system is capable of performing this function for an extended period of time following plant shutdown.

The RHRS is provided with two residual heat removal pumps and heat exchangers arranged in two separate, independent flow paths. To assure reliability, each residual heat removal pump is connected to a different vital bus. Each train is isolated from the RCS on the suction side by two motor-operated valves in series with each valve receiving power via a separate motor control center and from a different vital bus. Each suction isolation valve is also provided with "open-prevent" interlock and "RHRS-Iso-Valve-Open" alarm to prevent exposure of the RHRS to the normal operating pressure of the RCS (see Section 5.4.7.2.5).

RHRS operation for normal conditions and for major failures is accomplished completely from the control room. The redundancy in the RHRS design provides the system with the capability to maintain its cooling function even with major single failure, such as failure of a residual heat removal pump, valve, or heat exchanger without impact on the redundant train's continued heat removal.

Although such major system failures are within the system design basis, there are other less significant failures which can prevent opening of the residual heat removal suction isolation valves from the control room. Since these failures are of a minor nature, improbable to occur, and easily corrected outside the control room, with ample time to do so, they have been realistically excluded from the engineering design basis. Such failures are not likely to occur during the limited time period in which they can have any effect (i.e., when opening the suction isolation valves to initiate residual heat removal operation). However, even if they should occur, they have no adverse safety impact and can be readily corrected. In such a situation, the auxiliary feedwater system and the steam generator Atmospheric Relief Valves can be used to perform the safety function of removing residual heat and, in fact, can be used to continue the plant cooldown below 350°F, until the RHRS is made available.

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One example of this type of a failure is the interlock circuitry which is designed to prevent exposure of the RHRS to the normal operating pressure of the RCS (see Section 5.4.7.2.5). In the event of such a failure, RHRS operation can be initiated by defeating the failure interlock through corrective action at the solid state protection system cabinet or at the individual affected motor control centers.

The other type of failure which can prevent opening the residual heat removal suction isolation valves from the control room is a failure of an electrical power train. Such a failure is extremely unlikely to occur during the few minutes out of a year's operating time during which it can have any consequence. If such an unlikely event should occur, several alternatives are available. The most realistic approach would be to obtain restoration of offsite power, which can be expected to occur in less than 1/2 hour. Other alternatives are to restore the emergency diesel generator to operation or to bring in an alternative power source.

The only impact of either of the above types of failures is some delay in initiating residual heat removal operation, while action is taken to open the residual heat removal suction isolation valves. This delay has no adverse safety impact because of the capability of the auxiliary feedwater system and steam generator atmospheric relief valves to continue to remove residual heat, and, in fact, to continue plant cooldown.

A failure mode and effects analysis of the RHRS for normal plant cooldown is provided as Table 5.4-9.

5.4.7.2.8 Manual Actions

The RHRS is designed to be fully operable from the control room for normal operation. Manual operations required of the operator are: opening the suction isolation valves, positioning the flow control valves downstream of the RHRS heat exchangers, and starting the residual heat removal pumps.

Manual actions required outside the control room, under conditions of single failure, are discussed in Section 5.4.7.2.7.

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 component cooling water system at stepwise intervals following the initiation of RHR operation. Heat removal through the RHR and component cooling water heat exchangers is calculated at each interval by use of standard water-to-water heat

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exchanger performance correlations. The resultant fluid temperatures for the RHRS and component cooling water system are calculated and used as input to the next interval's heat balance calculation.

Assumptions utilized in the series of the heat balance calculations describing plant RHR cooldown are as follows:

- a. RHR operation is initiated 4 hours after reactor shutdown.
- b. RHR operation begins at a reactor coolant temperature of 350°F.
- c. Thermal equilibrium is maintained throughout the RCS during the cooldown.
- d. Component cooling water heat exchanger outlet temperature during cooldown is limited to a maximum of 120°F.
- e. Expected cooldown rates of 50°F per hour are not exceeded.
- f. Service water temperature is 90°F.
- g. RCS heat input from one reactor coolant pump is maintained until RCS temperature reaches 160°F.
- h. Auxiliary CCW heat loads are ($\times 10^6$ Btu/hr)

	1 (350 to 225°F)	
	2 (225 to 140°F)	
<u>Auxiliary CCW heat loads</u>	<u>Train RHR</u>	<u>1-Train RHR</u>
≥4 hrs. after shutdown	15.5	15.5
≥20 hrs. after shutdown	15.5	15.5

Cooldown curves calculated using this method are provided for the case when using both trains of residual heat removal cooldown (Figure 5.4-9) and for the case of a single train residual heat removal cooldown (Figure 5.4-10).

5.4.7.4 Preoperational Testing

Preoperational testing of the RHRS is addressed in Chapter 14.0.

5.4.8 REACTOR WATER CLEANUP SYSTEM

This section is not applicable to WCGS.

5.4.9 MAIN STEAM LINE AND FEED WATER PIPING

Discussion pertaining to the main steam line and feedwater piping are contained in the following sections:

- a. Main Steam Line Piping - Section 10.3.
- b. Main Feedwater Piping - Section 10.4.7.
- c. Auxiliary Feedwater Piping - Section 10.4.9.
- d. Inservice Inspection of a, b, and c - Section 6.6.

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5.4.10 PRESSURIZER

5.4.10.1 Design Bases

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

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

- a. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.
- b. The water volume is sufficient to prevent the heaters from being uncovered during a step load increase of 10 percent at full power.
- c. The steam volume is large enough to accommodate the surge resulting from a 50-percent reduction of full load with automatic reactor control and a 40-percent steam dump without the water level reaching the high level reactor trip point.
- d. 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 reactor control or steam dump.
- e. The pressurizer does not empty following reactor trip and turbine trip.
- f. The emergency core cooling does not activate because of a reactor trip and turbine trip.

The surge line is sized to minimize, to an acceptable value, the pressure drop between the RCS and the safety valves with maximum discharge flow from the safety valves.

The surge line and the thermal sleeves are designed to withstand the thermal stresses resulting from volume surges of water of different temperatures, which occur during operation.

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5.4.10.2 Design Description

5.4.10.2.1 Pressurizer and Surge Line

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

The general configuration of the pressurizer is shown in Figure 5.4-11. The design data of the pressurizer are given in Table 5.4-10. Codes and material requirements are provided in Section 5.2.

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

The surge line nozzle and removable electric heaters are located in the bottom of the pressurizer. The heaters are removable for maintenance or replacement.

The pressurizer surge line nozzle diameter is given in Table 5.4-10, and the pressurizer surge line diameter is shown in Figure 5.1-1, Sheet 2.

A thermal sleeve is provided in the surge line nozzle to minimize thermal stresses. A retaining screen is located above the nozzle to prevent 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 in mixing.

Spray line nozzles, relief and safety valve connections are located in the top head of the pressurizer vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves also can be operated manually 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 boron concentration in the pressurizer is not dissimilar from that in the reactor coolant and to prevent excessive cooling of the spray piping.

During an outsurge of water from the pressurizer, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit.

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

Material specifications are provided in Table 5.2-2 for the pressurizer, pressurizer relief tank, and the surge line. Design transients for the components of the RCS are discussed in Section 3.9(N).1. Additional details on the pressurizer design cycle analysis are given in Section 3.9(N).1.

5.4.10.2.2 Pressurizer Instrumentation

Refer to Chapter 7.0 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

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

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

5.4.10.3 Design Evaluation

5.4.10.3.1 System Pressure

Whenever a steam volume is present within the pressurizer, the RCS pressure is governed by conditions in the pressurizer.

A design basis safety limit is that RCS pressure does not exceed the maximum transient value allowed under the ASME Code, Section III.

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

During startup and shutdown, the rate of temperature change in the RCS is controlled by the operator. Heatup rate is controlled by energy input from the reactor coolant pumps and by the pressurizer electrical heating capacity. This heatup rate takes into account the continuous spray flow provided to the pressurizer. When the

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reactor core is in cold shutdown, the pressurizer heaters are de-energized except when establishing or maintaining a pressure bubble.

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 normal operating water volume at full load conditions is given in Table 5.4-10.

5.4.10.3.3 Pressure Setpoints

The RCS design and operating pressure, together with the safety, power relief, and pressurizer spray valves setpoints and the protection system pressure setpoints, are listed in Table 5.4-11. 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 valves is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping routed to the pressurizer forms a water seal which prevents the steam buildup back to the control valves. The spray rate is selected to prevent the pressurizer pressure from reaching the operating setpoint of the power relief valves during a step reduction in power level of 10 percent of full load.

The pressurizer spray lines and valves are large enough to provide the required spray flow rate under the driving force of the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop in order to utilize the velocity head of the reactor coolant loop flow to add to the spray driving force. The spray valves and

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spray line connections are arranged so that the spray will operate when one reactor coolant pump is not operating. The line may also be used to assist in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

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

5.4.10.4 Tests and Inspections

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

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

- a. Support skirt to the pressurizer lower head
- b. Surge nozzle to the lower head
- c. Nozzles safe ends to the surge, safety, relief, and spray lines *
- d. Nozzle to safe end attachment welds *
- e. All girth and longitudinal full penetration welds
- f. 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 5.4-12.

* In order to mitigate primary water stress corrosion cracking concerns with the originally installed Alloy 600 (82/182) dissimilar metal welds, full structural weld overlays made of ERNiCrFe-7A (Alloy 52M/UNS N06054) have been installed to cover portions of the Pressurizer nozzles (Surge, Safety, Relief, and Spray), nozzle weld butter layers, dissimilar metal welds between the butter and the safe end, safe ends, safe end to stainless steel pipe welds, and connecting stainless steel piping.

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5.4.11 PRESSURIZER RELIEF DISCHARGE SYSTEM

5.4.11.1 Design Bases

The pressurizer relief discharge system collects, cools, and directs for processing the steam and water discharged from safety and relief valves in the containment. The system consists of the pressurizer relief tank, the safety and relief valve discharge piping, the relief tank internal spray header and associated piping, the tank nitrogen supply, the vent to containment, and the drain to the waste processing system.

The system design is based on the requirement to absorb a discharge of steam 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 reactor trip. A delayed reactor trip is considered in the design of the system.

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

A safety-related flowpath downstream of the excess letdown heat exchanger is provided to direct a cooled flow to the PRT. This flow path may be used if the normal and excess letdown paths are unavailable or if it is desired to contain the reactor coolant inside the containment. Another flowpath is provided for the controlled release of fluid from the PRT to the containment normal sump.

The vessel saddle supports and anchor bolt arrangement 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 for the pressurizer relief discharge system is given in Figure 5.1-1, Sheet 2.

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Codes and materials of the pressurizer relief tank and associated piping are given in Section 5.2. Design data for the tank are given in Table 5.4-13.

The steam and water discharged from the various safety and relief valves inside the containment is routed to the pressurizer relief tank if the discharged fluid is of reactor grade quality. Table 5.4-14 provides an itemized list of valves discharging to the tank, together with references to the corresponding piping and instrumentation diagrams.

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 with the steam discharged through a sparger pipe located near the tank bottom and under the water level. The sparger holes are designed to ensure a resultant steam velocity close to sonic. The water in the tank may be discharged to allow increased capacity for RC letdown via the excess letdown path. In this mode, the water is cooled before it enters the tank.

The tank is also equipped with an internal spray and a drain which are used to cool the water following a discharge. Cold water is drawn from the reactor makeup water system, or the contents of the tank are circulated through the reactor coolant drain tank heat exchanger of the waste processing system and back into the spray header.

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 pressure based on an arbitrary design pressure of 100 psig. The design discharge raises the worst case initial conditions to 50 psig, a pressure low enough to prevent fatigue of the rupture discs. Provision is made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen.

The contents of the tank can be drained to the waste holdup tank in the waste processing system or the recycle holdup tank in the boron recycle system via the reactor coolant drain tank pumps in the waste processing system. Under emergency conditions, the tank contents can be drained to the containment normal sump.

5.4.11.2.1 Pressurizer Relief Tank

The general configuration of the pressurizer relief tank is shown in Figure 5.4-12. The tank is a horizontal, cylindrical vessel with elliptical dished heads. The vessel is constructed of

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austenitic stainless steel, and is overpressure protected in accordance with the ASME Code, Section VIII, Division 1, by means of two safety heads with stainless steel rupture discs. The PRT saddle supports are designed to withstand the loadings resulting from a combination of nozzle loadings acting simultaneously with the vessel seismic and static loadings.

A flange nozzle is provided on the tank for the pressurizer discharge line connection to the sparger pipe. The tank is also equipped with an internal spray connected to a cold water inlet and with a bottom drain, which are used to cool the tank following a discharge.

5.4.11.3 Design Evaluation

The pressurizer relief discharge system does not constitute part of the reactor coolant pressure boundary per 10 CFR 50, Section 50.2, since all of its components are downstream of the RCS safety and relief valves. Thus, General Design Criteria 14 and 15 are not applicable. Furthermore, complete failure of the auxiliary systems serving the pressurizer relief tank will not impair the capability for safe plant shutdown.

The design of the system piping layout and piping restraints is consistent with the hazards protection requirements indicated in Appendix 3.B. The safety and relief valve discharge piping is restrained so that the integrity and operability of the valves are maintained in the event of a rupture. Regulatory Guide 1.67 is not applicable, since the system is not an open discharge system.

The pressurizer relief discharge system is capable of handling the design discharge of steam without exceeding the design pressure and temperature of the pressurizer relief tank.

The volume of 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 relief device on the tank would pass 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 discs holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

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The discharge piping from the pressurizer 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 pressurizer relief tank pressure transmitter provides an indication of pressure relief tank pressure. An alarm is provided to indicate high tank pressure.

The pressurizer relief tank level transmitter supplies a signal for an indicator with high and low level alarms. The temperature of the water in the pressurizer relief tank is indicated, and an alarm actuated by high temperature informs the operator that cooling of the tank contents is required.

5.4.11.5 Tests and Inspections

The system components and piping are subject to nondestructive and hydrostatic testing during construction, in accordance with Section VIII, Division 1 of the ASME Code and ANSI B31.1, respectively.

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 inch, are ANS Safety Class 1, and ASME III, 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 5.4-15.

For a check valve to qualify as part of the RCS, it must be located inside the containment system. 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 back-flow leakage of the first valve when closed, if required.

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 design.

5.4.12.2 Design Description

All manual and motor-operated valves of the RCS which are larger than 2 inches are provided with graphite packing. Throttling control valves are provided with graphite packing. Leakage to the atmosphere is essentially zero for these valves.

Gate valves at the engineered safety features interface are wedge design and are essentially straight through. The wedges are flex-wedge or solid. Gate valves have backseats. Globe valves are "T" and "Y" styles. Check valves are swing type for sizes 2-1/2 inches 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. All operating parts are contained within the check valve body. The disc has limited rotation to provide a change of seating surface and alignment after each check valve opening.

5.4.12.3 Design Evaluation

The design requirements for Class 1 valves, as discussed in Section 5.2, limit stresses to levels which ensure the structural integrity of the valves. In addition, the testing programs described in Section 3.9(N) 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 is analyzed periodically.

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

5.4.12.4 Tests and Inspections

The tests and inspections discussed in Section 3.9(B).6 are performed to ensure the operability of the active valves.

There are no full-penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual

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inspection, to the extent practical. Plant layout configurations determine the degree of inspectability. The valve nondestructive examination program is given in Table 5.4-16. Inservice 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 can accommodate the maximum pressurizer surge resulting from complete loss of load, without reactor trip or any operator action and by the opening of the steam generator safety valves when steam pressure reaches the steam side safety setting.

The pressurizer power-operated relief valves 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.

5.4.13.2 Design Description

The pressurizer safety valves are of the pop type. The valves are spring loaded, open by direct fluid pressure action, and have backpressure compensation features.

The pipe connecting each pressurizer nozzle to its safety valve is 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 valve 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 actuation period.

The pressurizer power-operated relief valves are solenoid actuated valves which respond to a signal from a pressure sensing system or to manual control. Motor-operated valves are provided to isolate the power-operated relief valves if excessive leakage develops or if the PORV fails to close.

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

Liquid flow rates assumed in the analysis are based on the homogeneous equilibrium saturated flow model which gives the most conservative relief rate. Accident analysis demonstrates that water relief through the pressurizer valves occurs only during the

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feedline rupture event. The results of the WCGS feedline rupture analysis show that there is no water relief through the pressurizer valves at any time during the event.

The power-operated relief valves provide the safety-related means for reactor coolant system depressurization to achieve cold shutdown.

Design parameters for the pressurizer safety and power relief valves are given in Table 5.4-17.

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

5.4.13.4 Tests and Inspections

Safety and relief valves are subjected to hydrostatic tests, seat leakage tests, operational tests, and inspections, as required. For safety valves that are required to function during a faulted condition, additional tests are performed. These tests are described in Section 3.9(N). There are no full penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection.

Each pressurizer power-operated relief valve is demonstrated operable every 18 months by performing a channel calibration of the actuation instrumentation.

5.4.14 COMPONENT SUPPORTS

5.4.14.1 Design Bases

Component supports allow essentially unrestrained lateral thermal movement of the loop during plant operation except for a minor thermal restriction at the steam generator upper lateral supports as the system approaches operating temperature, and provide restraint to the loops and components during accident and seismic conditions. The loading combinations and design stress limits are discussed in Section 3.9(N).1.4. Support design is in accordance with the ASME Code, Section III, Subsection NF. The design maintains the integrity of the RCS boundary for normal, seismic, and accident conditions and satisfies the requirements of the piping code. Results of piping and supports stress evaluation are presented in Section 3.9(N).

5.4.14.2 Description

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

The supports permit essentially 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, and tie-rods for lateral support.

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

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

5.4.14.2.1 Reactor Pressure Vessel

Supports for the reactor vessel (Figure 5.4-13) 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 which transfers the loads to the primary shield wall concrete, and connecting vertical plates which bear against an embedded support. The supports are air cooled to maintain the supporting concrete temperature within acceptable levels.

5.4.14.2.2 Steam Generator

As shown in Figure 5.4-14, the steam generator supports consist of the following elements:

a. Vertical support

Four individual columns provide vertical support for each steam generator. These are bolted at the top to the steam generator and at the bottom to the concrete structure. Spherical ball bushings at the top and

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bottom of each column allow unrestrained lateral movement of the steam generator during heatup and cooldown. The column base design permits both horizontal and vertical adjustment of the steam generator for erection and adjustment of the system.

b. Lower lateral support

Lateral support is provided at the generator tube sheet by fabricated steel girders and struts. These are bolted to the compartment walls and include bumpers that bear against the steam generator but permit unrestrained movement of the steam generator during changes in system temperature.

Stresses in the beams caused by wall displacement during compartment pressurization are considered in the design.

c. Upper lateral support

The upper lateral support of the steam generator is provided by a ring band at the operating deck. One-way acting compression struts restrain sudden seismic or blowdown induced motion, but permit essentially unrestrained thermal movement of the steam generator. Movement perpendicular to the thermal growth direction of the steam generator is prevented by struts.

5.4.14.2.3 Reactor Coolant Pump

Three individual columns, similar to those used for the steam generator, provide the vertical support for each pump. Lateral support for seismic and blowdown loading is provided by three lateral tension tie bars. The pump supports are shown in Figure 5.4-15.

5.4.14.2.4 Pressurizer

The supports for the pressurizer, as shown in Figures 5.4-16 and 5.4-17, consist of:

- a. A steel ring 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.

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- b. The upper lateral support consists of struts cantilevered off the compartment walls that bear against the "seismic lugs" provided on the pressurizer.

5.4.14.2.5 Pipe Restraints

- a. Crossover leg

Restraint at each elbow of the reactor coolant pipe between the pump and the steam generator (crossover leg) was provided in the original design to prevent excessive stresses on the system resulting from postulated breaks in this pipe. The support includes pipe bumpers with straps and steel thrust blocks, as shown in Figure 5.4-18, and concrete. Also, a whip restraint strut, as shown in Figure 5.4-19, was originally provided to prevent whipping of the crossover leg pipe following a postulated break at the steam generator outlet nozzle. This restraint was attached to the secondary shield wall and extended horizontally to the vertical run of the crossover leg pipe.

Using leak-before-break technology, as allowed by revised GDC-4 (see USAR Section 3.6), the crossover leg whip restraints have been deactivated. The shims have been removed from between the saddle blocks and backup structures at the elbow restraints, and for the vertical run restraints, the tie rods and pipe clamp assemblies have been removed.

- b. Hot leg

A restraint located near the 50-degree elbow in the hot leg was provided in the original design to prevent excessive displacement of the hot leg following a postulated guillotine break at the steam generator inlet nozzle. This restraint consists of structural steel members which transmit loads to the concrete structure. This restraint is shown in Figure 5.4-20. Using leak-before-break technology as allowed by revised GDC-4, the hot leg elbow whip restraint has been deactivated. The shims between the pipe saddle and the backup structure have been removed.

- c. Hot leg and cold leg lateral restraints

A restraint on each reactor coolant system hot leg and cold leg is located near the reactor vessel safe-end to reactor coolant system piping weld with the reactor vessel primary shield wall to prevent excessive displacement of either the hot leg or the cold leg following a postulated guillotine break at the reactor vessel safe-end to piping weld. These restraints are shown in Figure 5.4-21.

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5.4.14.3 Design Evaluation

Detailed evaluation ensures the design adequacy and structural integrity of the reactor coolant loop and the primary equipment supports system. This detailed evaluation is made by comparing the analytical results with established criteria for acceptability. Structural analyses are performed to demonstrate design adequacy for safety and reliability of the plant in case of a large or small seismic disturbance and/or LOCA conditions. Loads which the system is expected to encounter often during its lifetime (thermal, weight, and pressure) are applied, and stresses are compared to allowable values as described in Section 3.9(N).1.4.

The safe shutdown earthquake and design basis LOCA, resulting in a rapid depressurization of the the system, are required design conditions for public health and safety. The methods used for the analysis of the safe shutdown earthquake and LOCA conditions are given in Sections 3.9(N).1.4.

5.4.14.4 Tests and Inspections

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

5.4.15 REFERENCES

1. "Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September, 1973.
2. Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactor, Program Summaries - Winter 1977 - Summer 1978," WCAP-8768, Revision 2, October, 1978.
3. DeRosa, P., et al., "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions," WCAP-7832, January, 1974.
4. "Structural Analysis of the Reactor Coolant Loop for the Standard Nuclear Unit Power Plant System, Volume 2, Analysis of the Primary Equipment Supports," WCAP 9728 Rev. 3, January, 1993.
5. Letter 07-00401, dated July 19, 2007, from USNRC to WCNO, Authorization of Relief Request 13R-05, alternatives to Structural Weld Overlay Requirements.

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TABLE 5.4-1

REACTOR COOLANT PUMP DESIGN PARAMETERS

Unit design pressure, psig	2,485
Unit design temperature, F	650 (a)
Unit overall height, ft	26.93
Seal water injection, gpm	8
Seal water return, gpm	3
Cooling water flow, gpm	366
Maximum continuous cooling water inlet temperature	105
Pump	
Capacity, gpm	100,600
Developed head, ft	288
NPSH required, ft	Figure 5.4-2
Suction temperature, F	558.2
Pump discharge nozzle, inside diameter, in.	27-1/2
Pump suction nozzle, inside diameter, in.	31
Speed, rpm	1,185
Water volume (casing), ft ³	78.6
*Weight total (including pump casing, motor, and motor supports), dry, lb	204,035 (with bolts) 205, 696 (with studs)
Motor	
Type	Drip proof, squirrel cage induction, water/air cooled
Power, hp	7,000
Voltage, Volts	13,200
Phase	3
Frequency, Hz	60
Insulation class	Class F, thermalastic epoxy insulation

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

*Total pump weights between values shown are bounded by existing analyses.

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TABLE 5.4-1 (Sheet 2)

Starting

Current	1,750 amp @ 13,200 Volts
Input, hot reactor coolant	253 + 5 amp
Input, cold reactor coolant	336 + 7 amp
Pump moment of inertia, maximum (lb-ft ²)	
Flywheel	64,000
Shaft	745
Impeller	1,980
Rotor core	27,700
Runner	675
Coupling	190

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TABLE 5.4-2

REACTOR COOLANT PUMP QUALITY ASSURANCE PROGRAM

	<u>RT*</u>	<u>UT*</u>	<u>PT*</u>	<u>MT*</u>
Castings	Yes		Yes	
Forgings				
Main shaft		Yes		Yes
Main studs		Yes		Yes
Flywheel (rolled plate)		Yes		
Weldments				
Circumferential	Yes		Yes	
Instrument connections			Yes	

*RT - Radiographic
 UT - Ultrasonic
 PT - Dye penetrant
 MT - Magnetic particle

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TABLE 5.4-3

STEAM GENERATOR DESIGN DATA

Design pressure, reactor coolant side, psig	2,485
Design pressure, steam side, psig	1,185
Design pressure, primary to secondary, psi	1,600
Design temperature, reactor coolant side, F	650
Design temperature, steam side, F	600
Design temperature, primary to secondary, F	650
Total heat transfer surface area, ft ²	55,000
Maximum moisture carryover, wt percent	0.25
Overall height, ft-in.	67-8
Number of U-tubes	5,626
U-tube nominal diameter, in.	0.688
Tube wall nominal thickness, in.	0.040
Number of manways	4
Inside diameter of manways, in.	16
Number of handholes	6
Design fouling factor, ft ² -hr-F/Btu	0.00005
Steam flow (per unit), lb/hr	3.785 x 10 ⁶
Nominal primary side water volume, ft ³	
No load	962
Full load	962
Nominal secondary side water volume, ft ³	
No load	3,559.6
Full load	2,212.3

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TABLE 5.4-4

STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	(a) <u>RT</u>	(a) <u>UT</u>	(a) <u>PT</u>	(a) <u>MT</u>	(a) <u>ET</u>
Tube Sheet					
Forging		Yes		Yes	
Cladding		(b) Yes	Yes		
Channel Head (if fabricated)					
Fabrication	(c) Yes	(d) Yes		Yes	
Cladding			Yes		
Secondary Shell and Head					
Plates		Yes			
Tubes		Yes			Yes
Nozzles (Forgings)		Yes		Yes	
Weldments					
Shell, longitudinal	Yes			Yes	
Shell, circumferential	Yes			Yes	
Cladding (channel head-tube sheet joint cladding restoration)			Yes		
Primary nozzles to fab head	Yes			Yes	
Manways to fab head	Yes			Yes	
Steam and feedwater nozzle to shell	Yes			Yes	
Support brackets				Yes	
Tube to tube sheet			Yes		

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TABLE 5.4-4 (Sheet 2)

	(a) <u>RT</u>	(a) <u>UT</u>	(a) <u>PT</u>	(a) <u>MT</u>	(a) <u>ET</u>
Instrument connections (primary and secondary)				Yes	
Temporary attachments after removal				Yes	
After hydrostatic test (all major pressure boundary welds and complete cast channel head - where accessible)				Yes	
Nozzle safe ends (if weld deposit)	Yes		Yes		

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

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TABLE 5.4-5

REACTOR COOLANT PIPING DESIGN PARAMETERS

Reactor inlet piping, inside diameter, in.	27-1/2
Reactor inlet piping, nominal wall thickness, in.	2.32
Reactor outlet piping, inside diameter, in.	29
Reactor 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.	14
Pressurizer surge line piping, nominal wall thickness, in.	1.406
Nominal water volume, all four loops including surge line, ft ³	1,225
Reactor Coolant Loop Piping	
Design/operating pressure, psig	2,485/2,235
Design temperature, F	680
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

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TABLE 5.4-6

REACTOR COOLANT PIPING QUALITY ASSURANCE PROGRAM

	<u>RT*</u>	<u>UT*</u>	<u>PT*</u>
Fittings and Pipe (Castings)	Yes		Yes
Fittings and Pipe (Forgings)		Yes	Yes
Weldments			
Circumferential	Yes		Yes
Nozzle to runpipe (except no RT for nozzles less than 6 inches)	Yes		Yes
Instrument connections			Yes
Castings	Yes		Yes (after finishing)
Forgings		Yes	Yes (after finishing)

* RT - Radiographic
 UT - Ultrasonic
 PT - Dye Penetrant

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TABLE 5.4-7

DESIGN PARAMETERS FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION |

Residual heat removal system startup, hours after reactor shutdown	~4
Reactor coolant system initial pressure, psig	~360
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	≤20
Reactor coolant system temperature at end of cooldown, F	140
Decay heat generation at 20 hours after reactor shutdown, Btu/hr	75.2 x 10 ⁶

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TABLE 5.4-8

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pumps

Number	2
Design pressure, psig	600
Design temperature, F	400
Design flow, gpm	3,800
Design head, ft	350
NPSH required at 3,800 gpm, ft	17
Power, hp	500

Residual Heat Exchangers

Number	2
Design heat removal capacity, Btu/hr	39.0 x 10 ⁶
Estimated UA, Btu/hr F _{LMTD}	2.3 x 10 ⁶

	<u>Tube Side</u>	<u>Shell Side</u>
Design pressure, psig	600	150
Design temperature, F	400	200
Design flow, lb/hr	1.9 x 10 ⁶	3.8 x 10 ⁶
Inlet temperature, F	140	105
Outlet temperature, F	119.4	115.2
Material	Austenitic stainless steel	Carbon steel
Fluid	Reactor coolant	Component cooling water

RHR Isolation Valve Encapsulation Tank (TEJ01A & B)

Manufacturer	Richmond Eng.
Quantity	2
Height ft-in.	12-6
Diameter ft-in.	5-6
Design Pressure, psig	75
Design Temperature, F	400
Material	Austenitic stainless steel
Codes and Standards	ASME Section III, Class 2
Seismic Category	I

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TABLE 5.4-9 (Sheet 1 of 5)

FAILURE MODES AND EFFECTS ANALYSIS - RESIDUAL HEAT REMOVAL SYSTEM
ACTIVE COMPONENTS - PLANT COOLDOWN OPERATION

<u>Component</u>	<u>Failure Mode</u>	<u>Effect on System Operation*</u>	<u>Failure Detection Method**</u>	<u>Remarks</u>
1. Motor-operated gate valve 8701A (8701B analogous)	a. Fails to open on demand (open manual mode CB switch selection)	Failure blocks reactor coolant flow from hot leg of RC loop 1 through train "A" of RHRS. Fault reduces redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop 4 flowing through train "B" of RHRS. However, time required to reduce RCS temperature will be extended.	Valve position indication (closed to open position change) at CB; RC loop 1 hot leg pressure indication (PI-405) at CB; RHR train "A" discharge flow indication (FI-618) and low flow alarm at CB; and RHR pump discharge pressure indication (PI-614) at CB.	<p>1. Valve is electrically interlocked with the containment sump isolation valves 8811A and 8812A, with RHR to charging pump suction line isolation valve 8804A and with a "prevent-open" pressure interlock (PB-405A) of RC loop 1 hot leg. The valve cannot be opened remotely from the CB if one of the indicated isolation valves is open or if RC loop pressure exceeds 360 psig.</p> <p>2. If both trains of RHRS are unavailable for plant cooldown due to multiple component failures, the auxiliary feedwater system and SG atmospheric relief</p>

*See list at end of table for definition of acronyms and abbreviations used.

**As part of plant operation; periodic tests, surveillance inspections, and instrument calibrations are made to monitor equipment and performance. Failures may be detected during such monitoring of equipment, in addition to detection methods noted.

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TABLE 5.4-9 (Sheet 2 of 5)

<u>Component</u>	<u>Failure Mode</u>	<u>Effect on System Operation*</u>	<u>Failure Detection Method**</u>	<u>Remarks</u>
				valves can be used to perform the safety function of removing residual heat.
2. Motor-operated gate valve 8702A (8702B analogous)	Same failure modes as those stated for item 1.	Same effect on system operation as that stated for item 1.	Same methods of detection as those stated for item 1.	Same remarks as those stated for item 1, except for pressure interlock (PB-403A) control.
3. RHR pump 1, APRH (RHR pump 2 analogous)	Fails to deliver working fluid.	Failure results in loss of reactor coolant flow from hot leg of RC loop 1 through train "A" of RHRS. Fault reduces redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg or RC loop 4 flowing through train "B" of RHRS. However, time required to reduce RCS temperature will be extended.	Open pump switchgear circuit breaker indication at CB; circuit breaker close position monitor light for group monitoring of components at CB; common breaker trip alarm at CB; RC loop 1 hot leg pressure indication (PI-405) at CB; RHR train "A" discharge flow indication (FI-618) and low flow alarm at CB; and pump discharge pressure indication (PI-614) at CB.	The RHRS shares components with the ECCS. Pumps are tested as part of the ECCS testing program (see Section 6.3.4). Pump failure may also be detected during ECCS testing.
4. Motor-operated gate valve FCV-610 (FCV-611 analogous)	a. Fails to open on demand (open manual mode CB switch selection).	Failure blocks miniflow line to suction of RHR pump "A" during cooldown operation of checking boron concentration level of coolant in train "A" of RHRS. Circulation through miniflow line is not available. If the operator does not secure RHR pump "A" before cavitation occurs, failure will reduce the redundancy of RHR coolant trains. No effect on safety for system operation.	Valve position indication (closed to open position change) at CB.	Valve is automatically controlled to open when pump discharge is less than ~816 gpm and close when the discharge exceeds ~1650 gpm. The valve protects the pump from dead-heading during ECCS operation. CB switch set to "Auto"

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TABLE 5.4-9 (Sheet 3 of 5)

<u>Component</u>	<u>Failure Mode</u>	<u>Effect on System Operation*</u>	<u>Failure Detection Method**</u>	<u>Remarks</u>
				position for automatic control of valve positioning.
	b. Fails to close on demand ("Auto" mode CB switch selection).	Failure allows for a portion of RHR heat exchanger "A" discharge flow to be bypassed to suction of RHR pump "A." RHRs train "A" is degraded for the regulation of coolant temperature by RHR heat exchanger "A." No effect on safety for system operation. Cooldown of RCS within established specification cooldown rate may be accomplished through operator action of throttling flow control valve HCV-606 and controlling cooldown with redundant RHRs train "B".	Valve position indication (open to closed position change) and RHRs train "A" discharge flow indication (FI-618) at CB.	
5. Air diaphragm-operated butterfly valve FCV-618 (FCV-619 analogous)	a. Fails to open on demand ("Auto" mode CB switch selection)	Failure prevents coolant discharged from RHR pump "A" from bypassing RHR heat exchanger "A" resulting in mixed mean temperature of coolant flow to RCS being low. RHRs train "A" is degraded for the regulation of controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished through operator action of throttling flow control valve HCV-606 and controlling cooldown with redundant RHRs train "B".	RHR pump "A" discharge flow temperature and RHRs train "A" discharge to RCS cold leg flow temperature recording (TR-612) at CB; and RHRs train "A" discharge to RCS cold leg flow indication (FI-618) at CB.	Valve is designed to fail "closed" and is electrically wired so that electrical solenoid of the air diaphragm operator is energized to open the valve. Valve is normally "closed" to align RHRs for ECCS operation during plant power operation and load follow.
	b. Fails to close on demand ("Auto" mode CB switch selection).	Failure allows coolant discharged from RHR pump "A" to bypass RHR heat exchanger "A", resulting in mixed mean temperature of coolant flow to RCS being high. RHRs train "A" is degraded for the regulation of	Same methods of detection as those stated for item 5.a.	

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TABLE 5.4-9 (Sheet 4 of 5)

<u>Component</u>	<u>Failure Mode</u>	<u>Effect on System Operation*</u>	<u>Failure Detection Method**</u>	<u>Remarks</u>
		controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished through operator action of throttling flow control valve HCV-606 and controlling cooldown with redundant RHRS train "B." However, cooldown time will be extended.		
6. Air diaphragm-operated butterfly valve HCV-606 (HCV-607 analogous)	a. Fails to close on demand for flow reduction.	Failure prevents control of coolant discharge flow from RHR heat exchanger "A," resulting in loss of mixed mean temperature coolant flow adjustment to RCS. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished by operator action of controlling cooldown with redundant RHRS train "B."	Same methods of detection as those stated for item 5. In addition, monitor light and alarm (valve closed) for group monitoring of components at CB.	Valve is designed to fail "open." Valve is normally "open" to align RHRS for ECCS operation during plant power operation and load follow.
	b. Fails to open on demand for increased flow.	Same effect on system operation as that stated for item 6.a.	Same methods of detection as those stated for item 6.a.	
7. Manual globe valve V001 (V002 analogous)	Fails closed.	Failure blocks flow from train "A" of RHRS to CVCS letdown heat exchanger. Fault prevents (during the initial phase of plant cooldown) the adjustment of boron concentration level of coolant in lines of RHRS train "A" so that it equals the concentration level in the RCS, using the RHR cleanup line to CVCS. No effect on safety for system operation. Operator can balance boron concentration levels by cracking open flow control valve HCV-606 to permit flow to cold leg of loop 1 of RCS in order to balance levels using normal CVCS letdown flow.	CVCS letdown flow indication (FI-132) at CB.	Valve is normally "closed" to align RHRS for ECCS operation during plant power operation and load follow.

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TABLE 5.4-9 (Sheet 5 of 5)

<u>Component</u>	<u>Failure Mode</u>	<u>Effect on System Operation*</u>	<u>Failure Detection Method**</u>	<u>Remarks</u>
8. Air diaphragm operated globe valve HCV-128	Fails to open on demand	Failure blocks flow from trains "A" and "B" of RHRS to CVCS letdown heat exchanger. Fault prevents use of RHR cleanup line to CVCS for balancing boron concentration levels of RHR trains "A" and "B" with RCS during initial cooldown operation and later in plant cooldown for letdown flow. No effect on safety for system operation. Operator can balance boron concentration levels with similar actions, using pertinent flow control valve HCV-606 (HCV-607), as stated for item 7. Normal CVCS letdown flow can be used for purification if RHRS cleanup line is not available.	Valve position indication (degree of openings) at CB and CVCS letdown flow indication (FI-132) at CB.	<p>1. Same remark as that stated above for item 7.</p> <p>2. Valve is a component of the CVCS that performs an RHR function during plant cooldown operation.</p>
9. Motor-operated gate valve 8812A (8812B analogous)	Fails to close on demand.	Failure reduces the redundancy of isolation valves provided to isolate RHRS train "A" from RWST. No effect on safety for system operation. Check valve 8958A in series with motor operated valve provides the primary isolation against the bypass of RCS coolant flow from the suction of RHR pump "A" to RWST.	Valve position indication (open to closed position change) at CB and valve (closed) monitor light and alarm at CB.	Valve is a component of the ECCS that performs a RHR function during plant cooldown. Valve is normally "open" to align RHRS for ECCS operation during-plant power operation and load follow.

List of acronyms and abbreviations

Auto - Automatic
 CB - Control board
 CVCS - Chemical and volume control system
 ECCS - Emergency core cooling system
 RC - Reactor coolant
 RCS - Reactor coolant system
 RHR - Residual heat removal
 RHRS - Residual heat removal system
 RWST - Refueling water storage tank
 SG - Steam generator

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TABLE 5.4-10

PRESSURIZER DESIGN DATA

Design pressure, psig	2,485
Design temperature, F	680
Surge line nozzle diameter, in.	14
Heatup rate of pressurizer using heaters only, F/hr	55
Internal volume, ft ³	1,800
Normal conditions at 100% rated load	
Steam volume, ft ³	720
Water volume, ft ³	1,080

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TABLE 5.4-11

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	Psig
Hydrostatic test pressure	3,107
Design pressure	2,485
Safety valves (begin to open)	2,460
High pressure reactor trip	2,385
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
Operating pressure	2,235
Proportional heater (full operation)	2,220
Backup heaters on	2,210
Low pressure alarm	2,210
Pressurizer power-operated relief and iso- lation valve interlock - auto closure	2,185
Low pressure reactor trip	1,940

*At 2,335 psig, a pressure signal initiates actuation (opening) of these valves. Remote manual control is also provided.

WOLF CREEK

TABLE 5.4-12

PRESSURIZER QUALITY ASSURANCE PROGRAM

	(a) <u>RT</u>	(a) <u>UT</u>	(a) <u>PT</u>	(a) <u>MT</u>
Heads				
Plates		Yes		
Cladding			Yes	
Shell				
Plates		Yes		
Cladding			Yes	
Heaters				
(b) Tubing		Yes	Yes	
Centering of element	Yes			
Nozzle (Forgings)		Yes	Yes (c)	Yes (c)
Weldments				
Shell, longitudinal	Yes			Yes
Shell, circumferential	Yes			Yes
Cladding			Yes	
Nozzle safe end	Yes		Yes	
Instrument connection			Yes	
Support skirt, longitudinal seam	Yes			Yes
Support skirt to lower head		Yes		Yes
Temporary attachments (after removal)				Yes
All external pressure boundary welds after shop hydrostatic test				Yes

- (a) RT - Radiographic
- UT - Ultrasonic
- PT - Dye Penetrant
- MT - Magnetic Particle
- (b) Or a UT and ET (Eddy Current)
- (c) MT or PT

WOLF CREEK

TABLE 5.4-13

PRESSURIZER RELIEF TANK DESIGN DATA

Design pressure, psig	100
Normal operating pressure, psig	3
Final operating pressure, psig	50
Rupture disc release pressure, psig	
Nominal	91
Range	86 to 100
Normal water volume, ft ³	1,350
Normal gas volume, ft ³	450
Design temperature, F	340
Initial operating water temperature, F	120
Final operating water temperature, F	200
Total rupture disc relief capacity at 100 psig, lb/hr	1.6 x 10 ⁶
Cooling time required following maximum discharge approximately, hr	
Spray feed and bleed	1
Utilizing RCDT heat exchanger	8

WOLF CREEK

TABLE 5.4-14

RELIEF VALVE DISCHARGE TO THE PRESSURIZER RELIEF TANK

Reactor Coolant System	
3 Pressurizer safety valves	Figure 5.1-1, Sheet 2
2 Pressurizer power-operated relief valves	Figure 5.1-1, Sheet 2
Residual Heat Removal System	
2 Residual heat removal pump suction lines from the reactor coolant system hot legs	Figure 5.4-7
Chemical and Volume Control System	
1 Seal water return line	Figure 9.3-8, Sheet 1
1 Letdown line	Figure 9.3-8, Sheet 1

WOLF CREEK

TABLE 5.4-15

REACTOR COOLANT SYSTEM VALVE DESIGN PARAMETERS

Design/normal operating pressure, psig	2,485/2,235
Preoperational plant hydrotest, psig	3,107
Design temperature, F	650

WOLF CREEK

TABLE 5.4-16

REACTOR COOLANT SYSTEM VALVES NONDESTRUCTIVE
EXAMINATION PROGRAM

	(a) <u>RT</u>	(a) <u>UT</u>	(a) <u>PT</u>
Boundary Valves, Pressurizer			
Relief and Safety Valves			
Castings (larger than 4 inches)	Yes		Yes
(2 inches to 4 inches)	Yes	(b)	Yes
Forgings (larger than 4 inches)	(c)	(c)	Yes
(2 inches to 4 inches)			Yes

- (a) RT - Radiographic
- UT - Ultrasonic
- PT - Dye Penetrant
- (b) Weld ends only
- (c) Either RT or UT

WOLF CREEK

TABLE 5.4-17

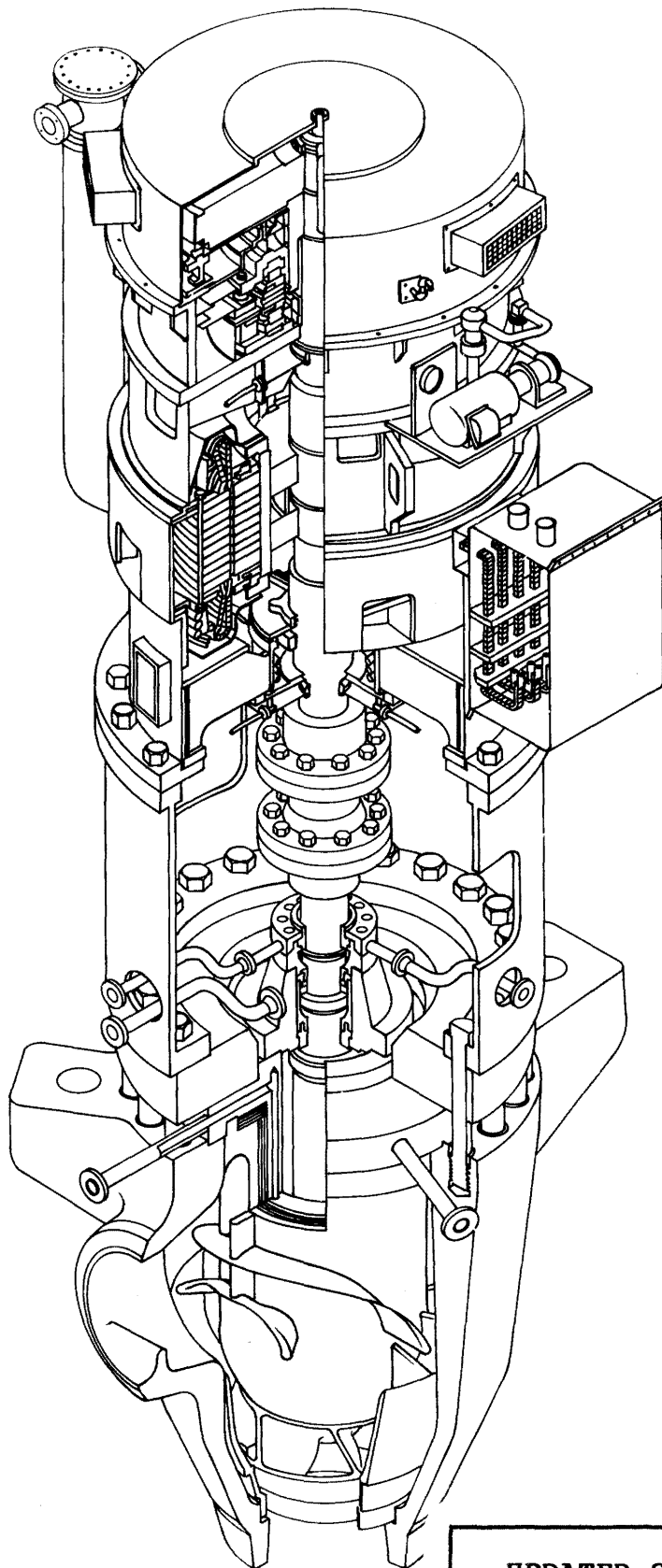
PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Safety Valves

Number	3
Maximum relieving capacity, ASME rate flow, lb/hr	415,764
Set pressure, psig	2,460
Design temperature, F	650
Fluid	Saturated steam
Transient condition, F	(Superheated steam) 680
Backpressure	
Normal, psig	3 to 5
Expected during discharge, psig	500
Environmental conditions	
Ambient temperature (F)	50 to 120
Relative humidity (%)	0 to 100

Pressurizer Power-Operated Relief Valves

Number	2
Design pressure, psig	2,485
Design temperature, F	650
Relieving capacity at 2,335 psig, per valve, lb/hr	210,000
Fluid	Saturated steam
Transient condition, F	(Superheated steam) 680

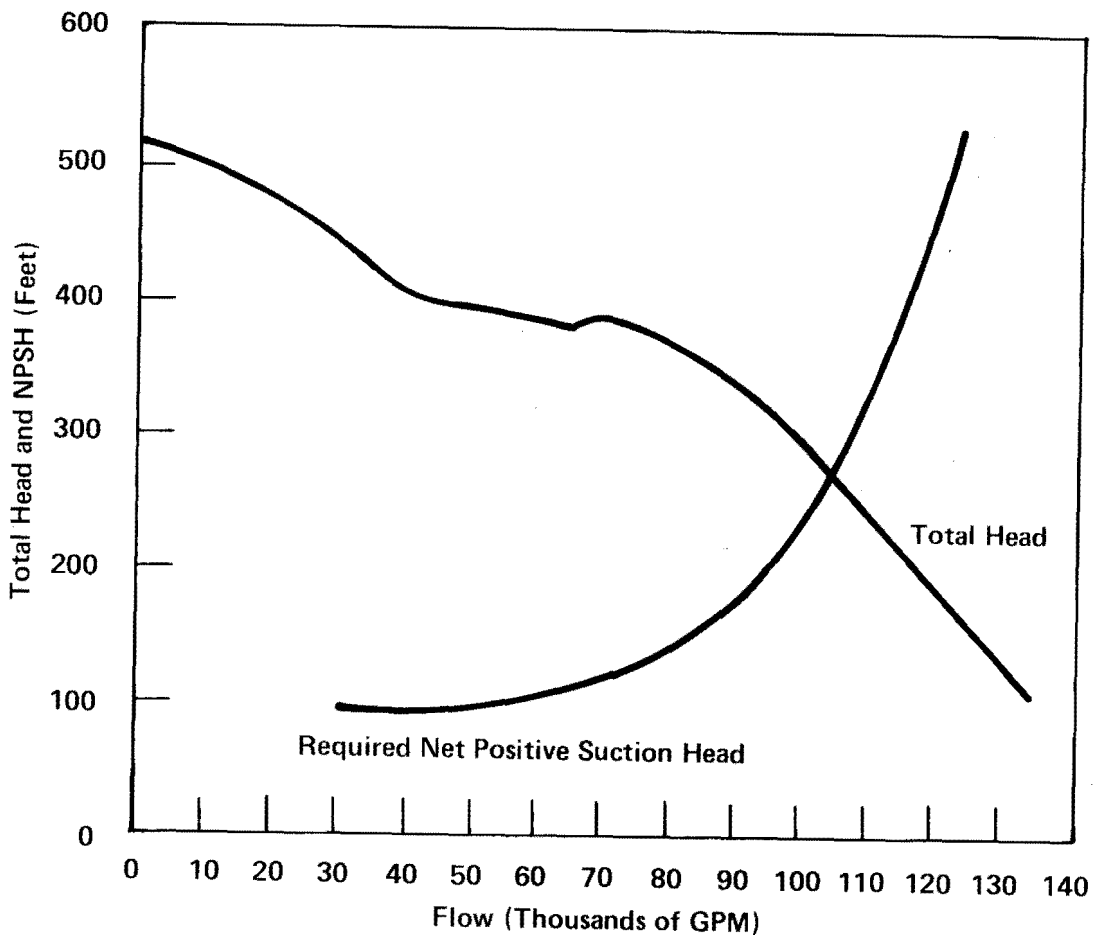


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FIGURE 5.4-1

REACTOR COOLANT CONTROLLED
LEAKAGE PUMP

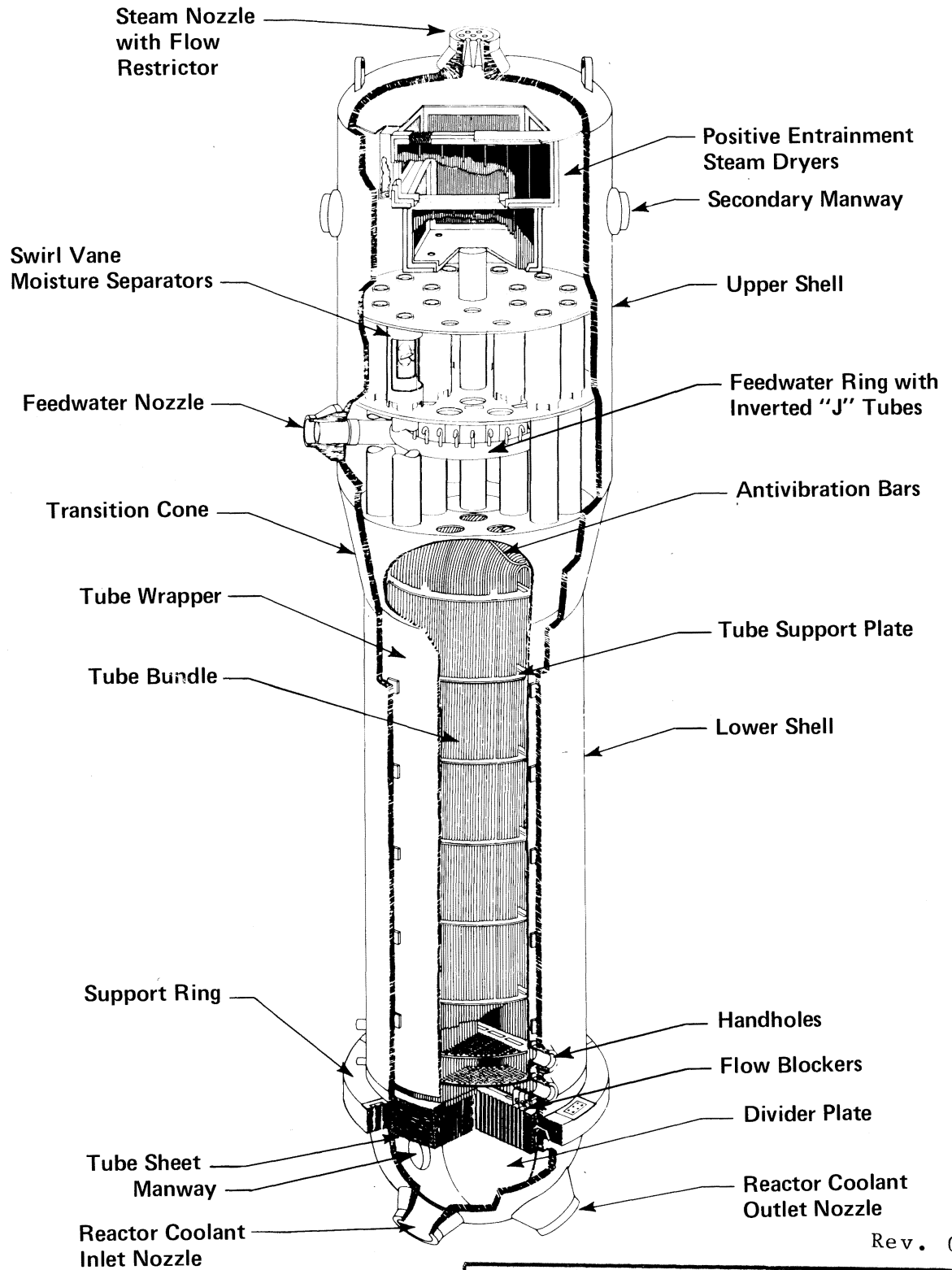


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FIGURE 5.4-2

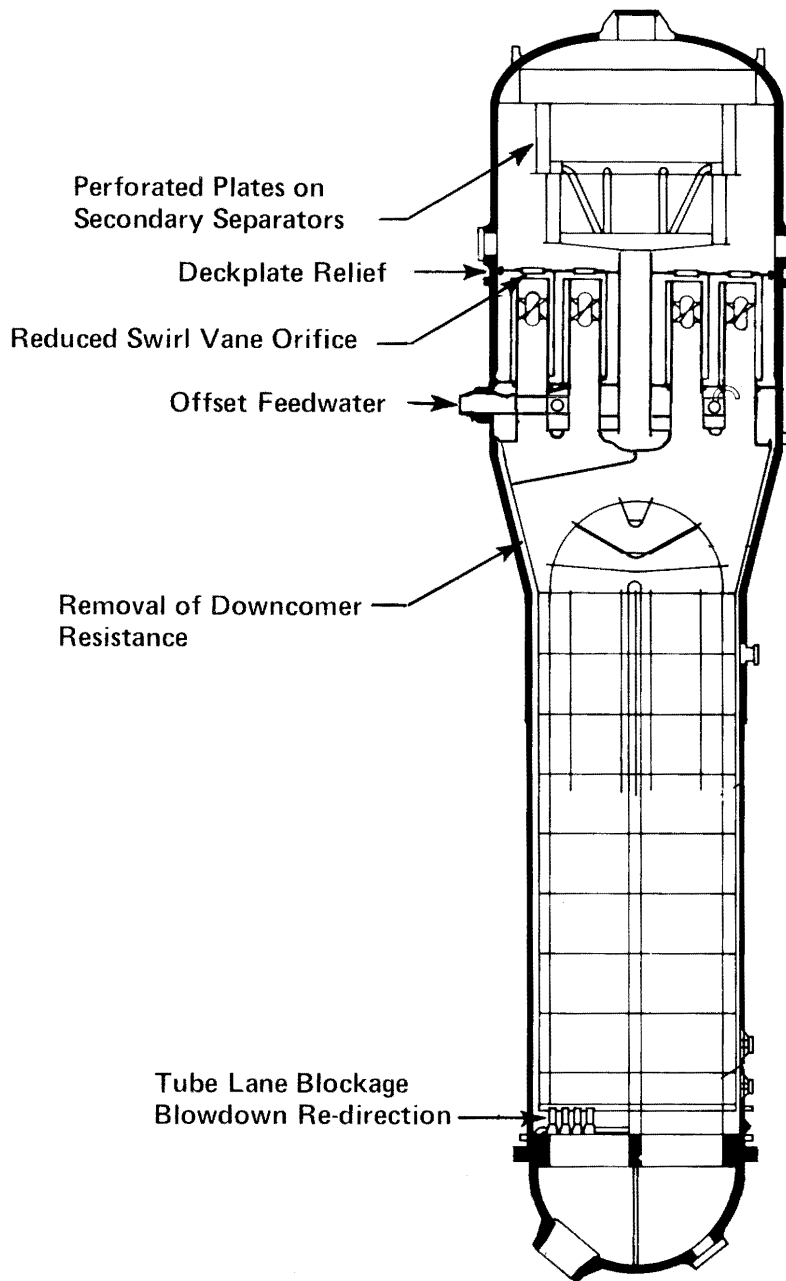
REACTOR COOLANT PUMP ESTIMATED
PERFORMANCE CHARACTERISTIC



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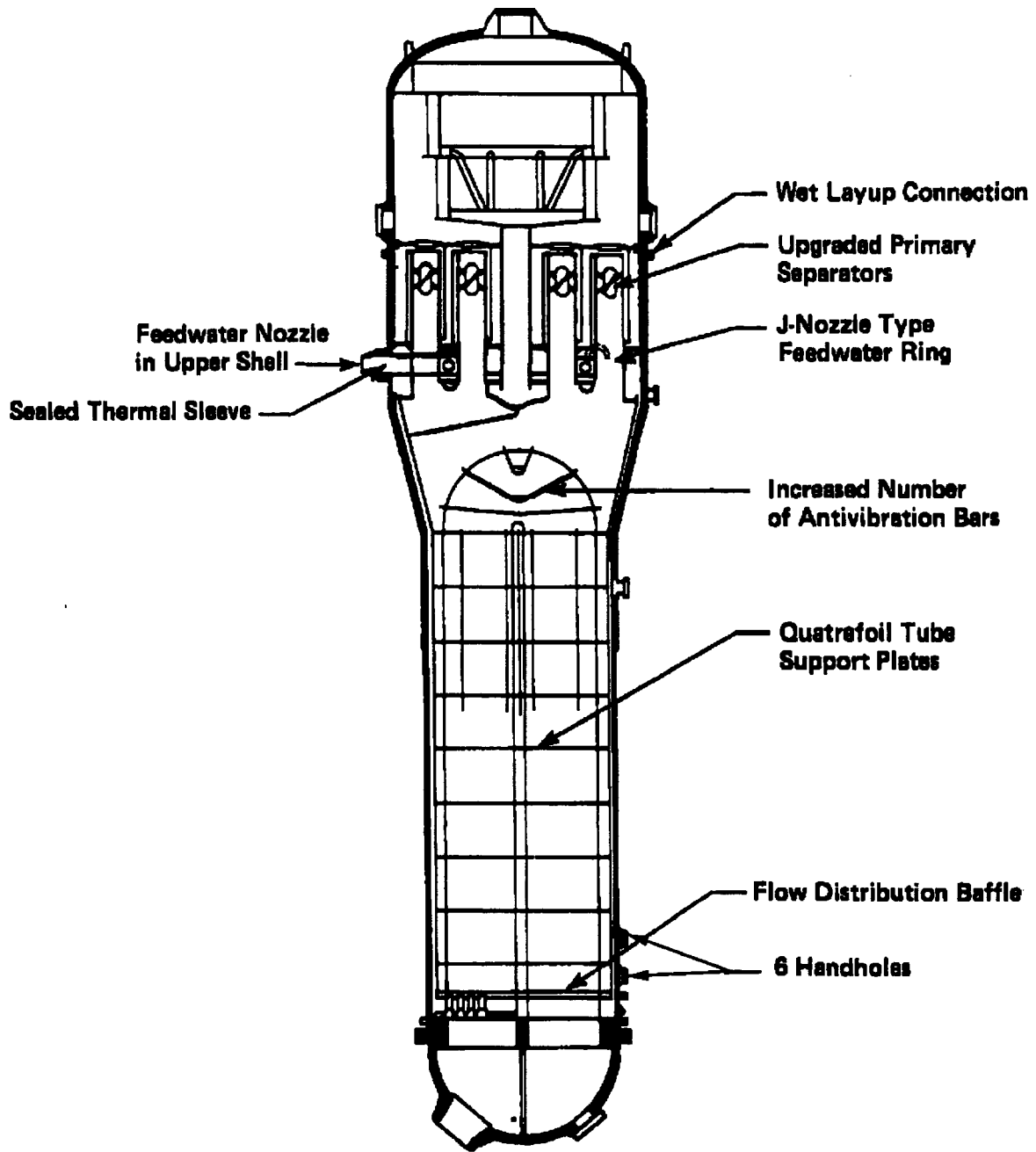
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FIGURE 5.4-3
WESTINGHOUSE MODEL F
STEAM GENERATOR



Rev. 0

<p>WOLF CREEK UPDATED SAFETY ANALYSIS REPORT</p>
<p>FIGURE 5.4-4</p>
<p>WESTINGHOUSE MODEL F STEAM GENERATOR MECHANICAL MODIFICATION IMPROVEMENTS</p>

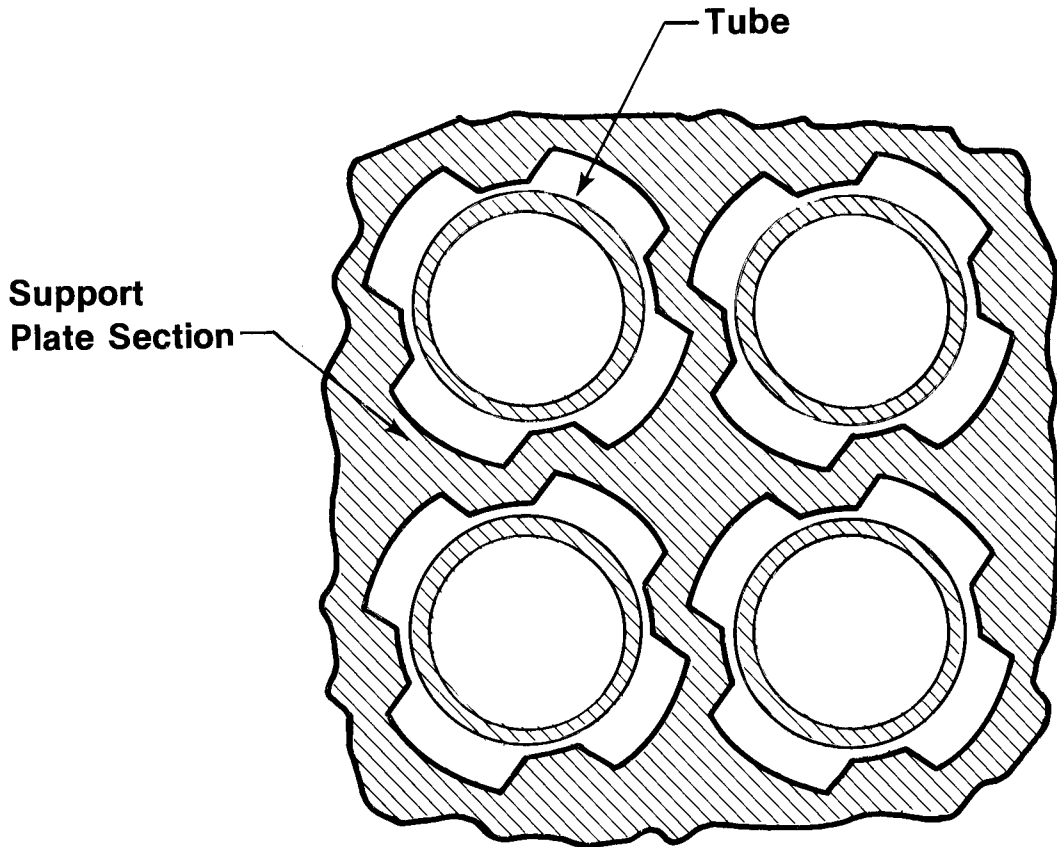


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Figure 5.4-5

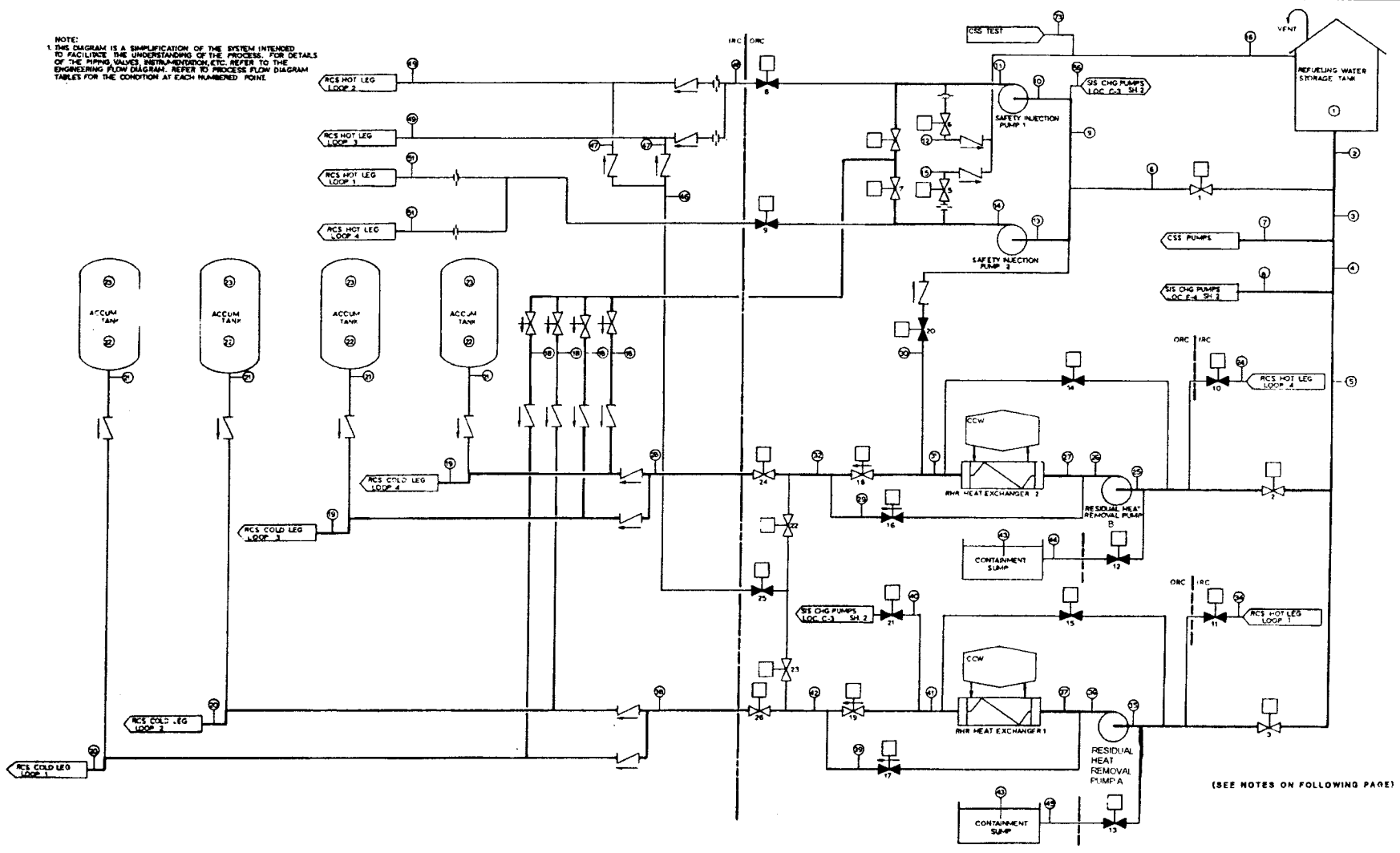
WESTINGHOUSE MODEL F STEAM
GENERATOR DESIGN IMPROVEMENTS



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<p>WOLF CREEK UPDATED SAFETY ANALYSIS REPORT</p>
<p>FIGURE 5.4-6 QUATREFOIL BROACHED HOLES</p>

NOTE:
 1. THIS DIAGRAM IS A SIMPLIFICATION OF THE SYSTEM INTENDED TO FACILITATE 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 CONDITION AT EACH NUMBERED POINT.



(SEE NOTES ON FOLLOWING PAGE)

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**FIGURE 5.4-8
 RESIDUAL HEAT REMOVAL SYSTEM
 PROCESS FLOW DIAGRAM**

WOLF CREEK

NOTES TO FIGURE 5.4-8

MODES OF OPERATION

MODE A - INITIATION OF RHR OPERATION

When the reactor coolant temperature and pressure are reduced to 350 F and 360 psig, approximately 4 hours after reactor shutdown, the second phase of plant cooldown starts with one train of RHR being placed in operation. Before starting the pump, the inlet isolation valves are opened, the heat exchanger flow control valve is set at minimum flow, and the outlet valve is verified open. The automatic miniflow valve is open and remains so until the pump flow exceeds the close setpoint at which time it closes. Should the pump flow drop below the open setpoint, the miniflow valves open automatically.

The other train of RHR is in the ECCS standby mode of operation from 350°F to 225°F. At ≤225°F this train is then allowed to operate in the shutdown cooling mode.

Startup of the RHRS includes a warmup 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 residual heat exchangers. The total flow is regulated automatically by control valves in the heat exchanger bypass line to maintain a constant total flow. The cooldown rate is limited to 50 F/hr, based on equipment stress limits and a 120 F maximum component cooling water temperature.

MODE B - END CONDITIONS OF NORMAL COOLDOWN

This situation characterizes the RHRS operation at lower RCS temperatures. As the reactor coolant temperature decreases, the flow through the residual heat exchanger is increased until all of the flow is directed through the heat exchanger to obtain maximum cooling.

Note:

For the safeguards functions performed by the RHRS, refer to Section 6.3, ECCS.

WOLF CREEK

NOTES TO FIGURE 5.4-8 (Sheet 2)

VALVE ALIGNMENT CHART

<u>Valve No.</u>	<u>Operational Mode</u>	
	<u>A</u>	<u>B</u>
2	C	C
3	O*	C
10	O	O
11	C*	O
12	C	C
13	C	C
14	C	C
15	O*	C
16	P	C
17	C*	C
18	P	O
19	O*	O
20	C	C
21	C	C
22	C*	O
23	C*	O
24	O	O
26	O	O

O = Open
 C = Closed
 P = Partially Open

*Valve position for RHR train in ECCS standby mode 350°F to 225°F.

WOLF CREEK

NOTES TO FIGURE 5.4-8 (Sheet 3)

MODE A - INITIATION OF RHR OPERATION

Location	Fluid	Pressure (psig)	Temperature (F)	Flow (gpm) ^(a)	Flow (lb/hr)
24	Reactor coolant	360	350	3,800	1.60 x 10 ⁶
25	Reactor coolant	367	350	3,800	1.60 x 10 ⁶
26	Reactor coolant	502	350	3,800	1.60 x 10 ⁶
27	Reactor coolant	501	350	1,259	0.56 x 10 ⁶
31	Reactor coolant	499	140	1,259	0.56 x 10 ⁶
29	Reactor coolant	456	350	2,541	1.13 x 10 ⁶
32	Reactor coolant	456	280	3,800	1.69 x 10 ⁶
28	Reactor coolant	440	280	3,690	1.64 x 10 ⁶
19 Loop 4	Reactor coolant	364	280	1,992	0.885 x 10 ⁶
19 Loop 3	Reactor coolant	379	280	1,698	0.755 x 10 ⁶
34*	RHR	Static Head	Ambient	0	0
35*	RHR	Static Head	Ambient	0	0
36*	RHR	Static Head	Ambient	0	0
37*	RHR	Static Head	Ambient	0	0
41*	RHR	Static Head	Ambient	0	0
39*	RHR	Static Head	Ambient	0	0
42*	RHR	Static Head	Ambient	0	0
38*	RHR	Static Head	Ambient	0	0
20 Loop 1*	RHR	Static Head	Ambient	0	0
20 Loop 2*	RHR	Static Head	Ambient	0	0

^(a)At reference conditions 350 °F and 360 psig

*RHR train in ECCS standby mode 350°F to 225°F

WOLF CREEK

NOTES TO FIGURE 5.4-8 (Sheet 4)

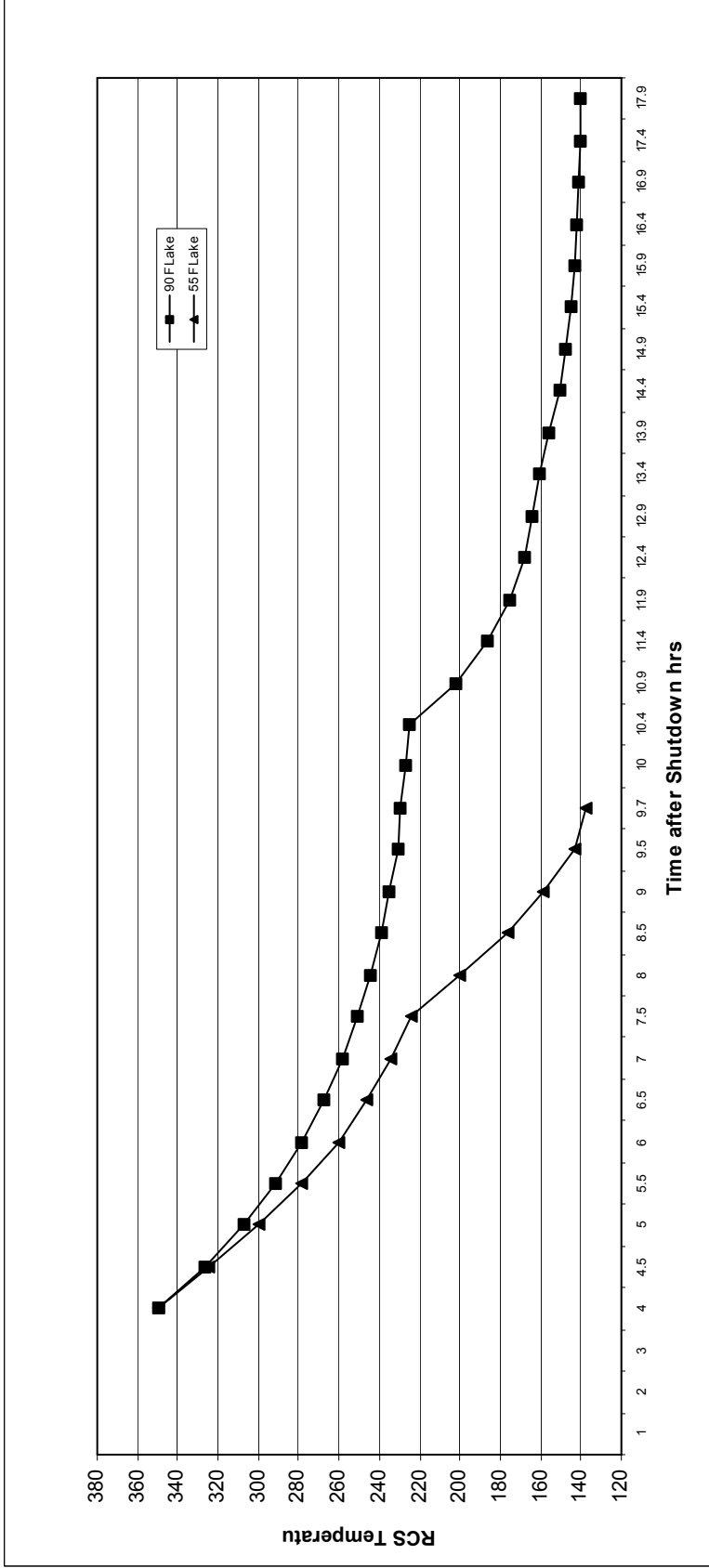
MODE B - END CONDITIONS OF A NORMAL COOLDOWN

Location	Fluid	Pressure (psig)	Temperature (F)	Flow	
				(gpm) ^(a)	(lb/hr)
24	Reactor coolant	0	140	3,800	1.87 x 10 ⁶
25	"	7	140	3,800	1.87 x 10 ⁶
26	"	156	140	3,800	1.87 x 10 ⁶
27	"	149	140	3,800	1.87 x 10 ⁶
31	"	129	120	3,800	1.87 x 10 ⁶
20	"	93	120	0	0
32	"	93	120	3,800	1.87 x 10 ⁶
28	"	75	120	3,800	1.87 x 10 ⁶
19	"	2	120	1,900	0.935 x 10 ⁶
34	"	0	140	3,800	1.87 x 10 ⁶
35	"	7	140	3,800	1.87 x 10 ⁶
36	"	156	140	3,800	1.87 x 10 ⁶
37	"	149	140	3,800	1.87 x 10 ⁶
41	"	129	120	3,800	1.87 x 10 ⁶
39	"	93	120	0	0
42	"	93	120	3,800	1.87 x 10 ⁶
38	"	75	120	3,800	1.87 x 10 ⁶
20	"	2	120	1,900	0.935 x 10 ⁶

^(a)At reference conditions 140 F and 0 psig

Wolf Creek NSSS Power 3579 MWt

Normal Plant Cooldown – One Train from 350 – 225 °F then both Trains

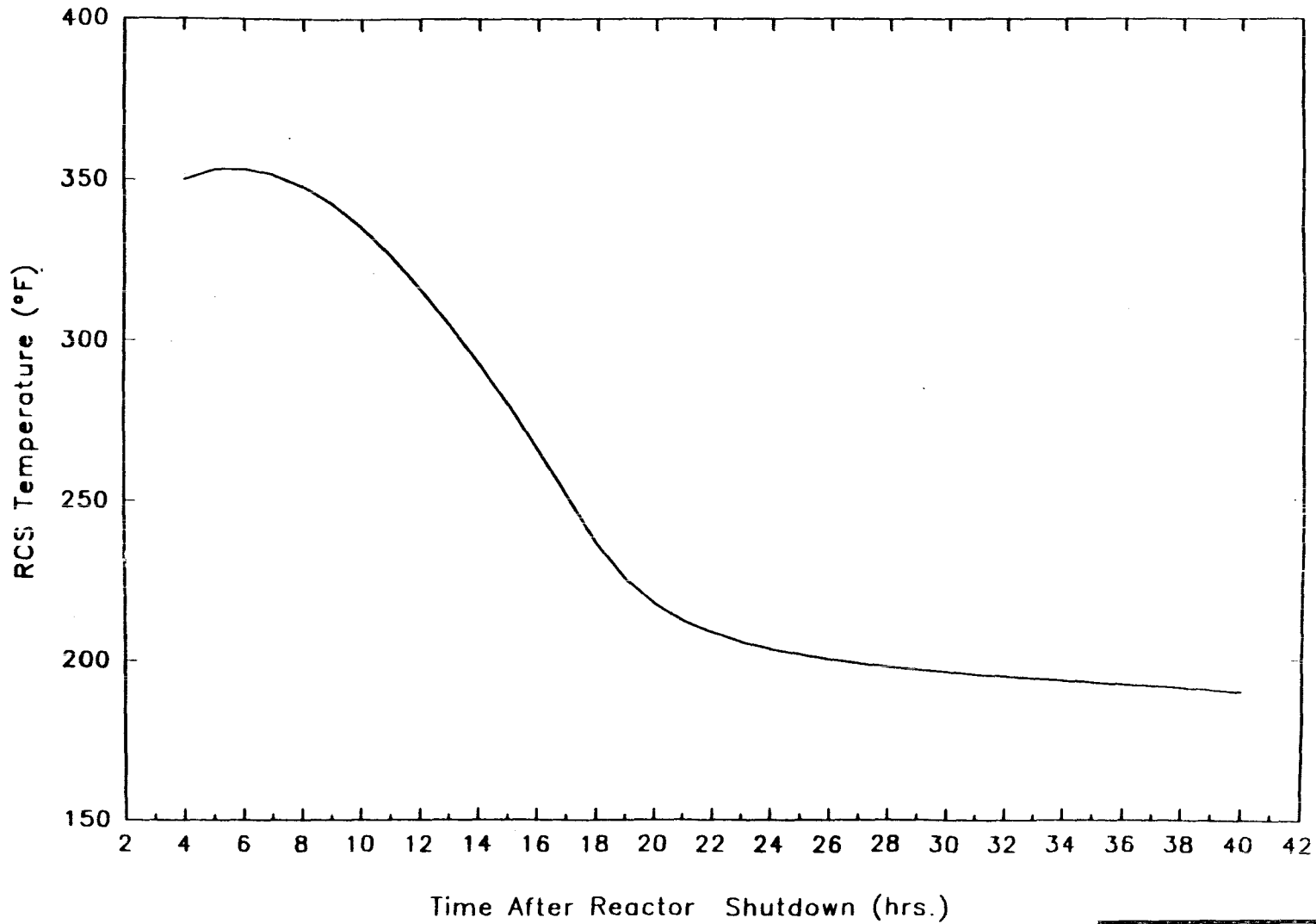


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FIGURE 5.4-9
NORMAL RESIDUAL HEAT REMOVAL
COOLDOWN

Wolf Creek NSSS Power 3579 MWt
Plant Cooldown - Single Train



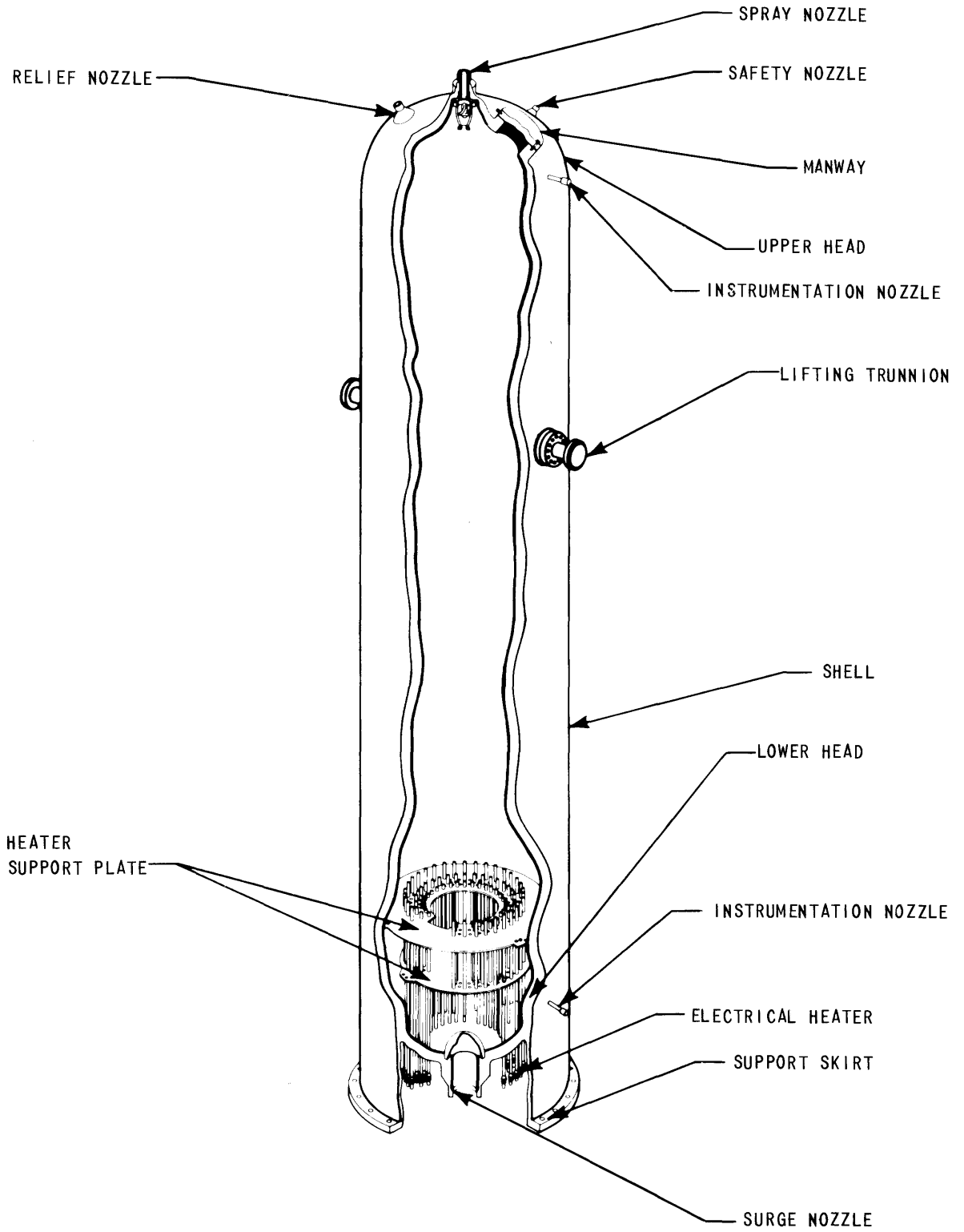
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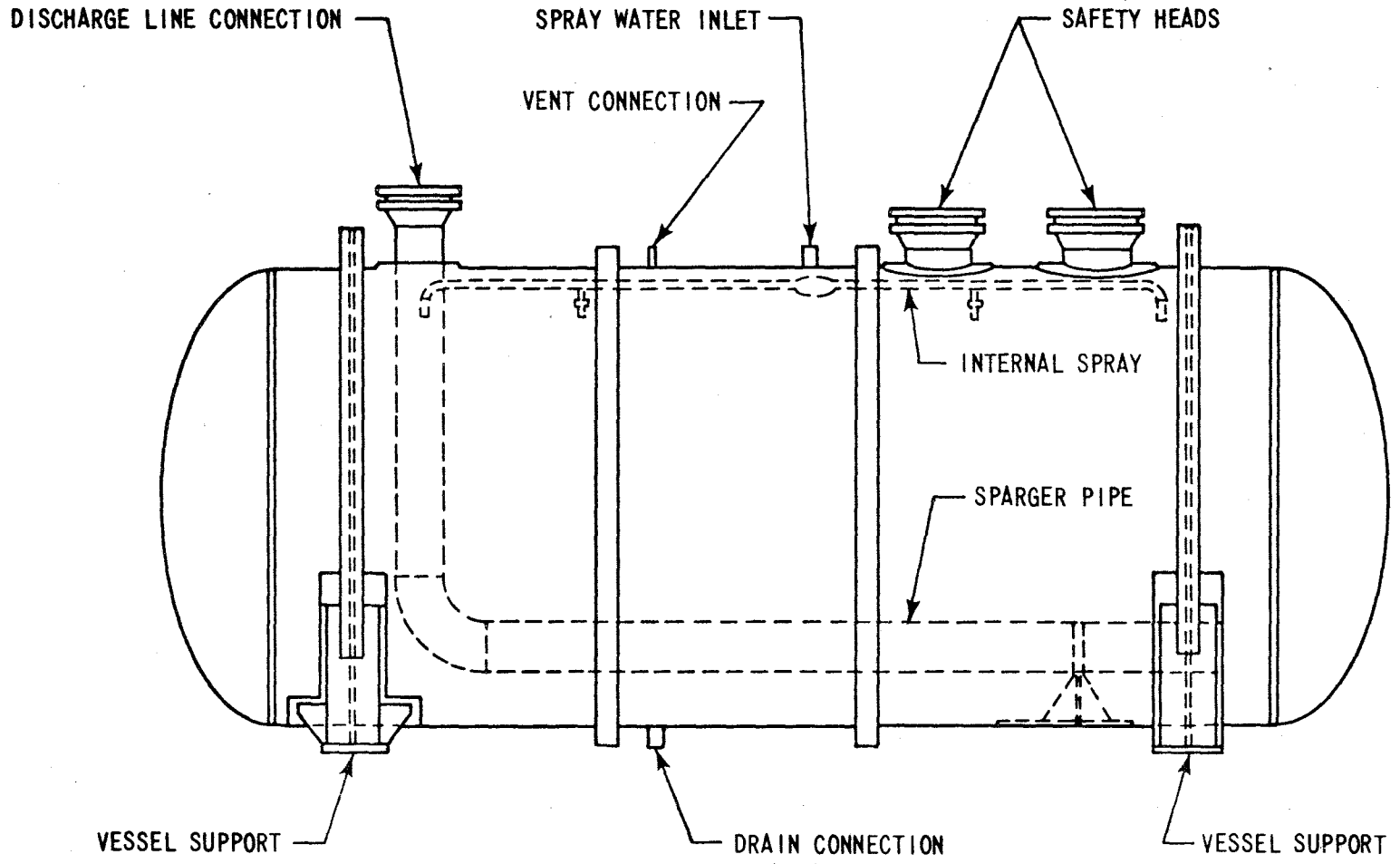
FIGURE 5.4-10

SINGLE RESIDUAL HEAT REMOVAL
TRAIN COOLDOWN

WOLF CREEK

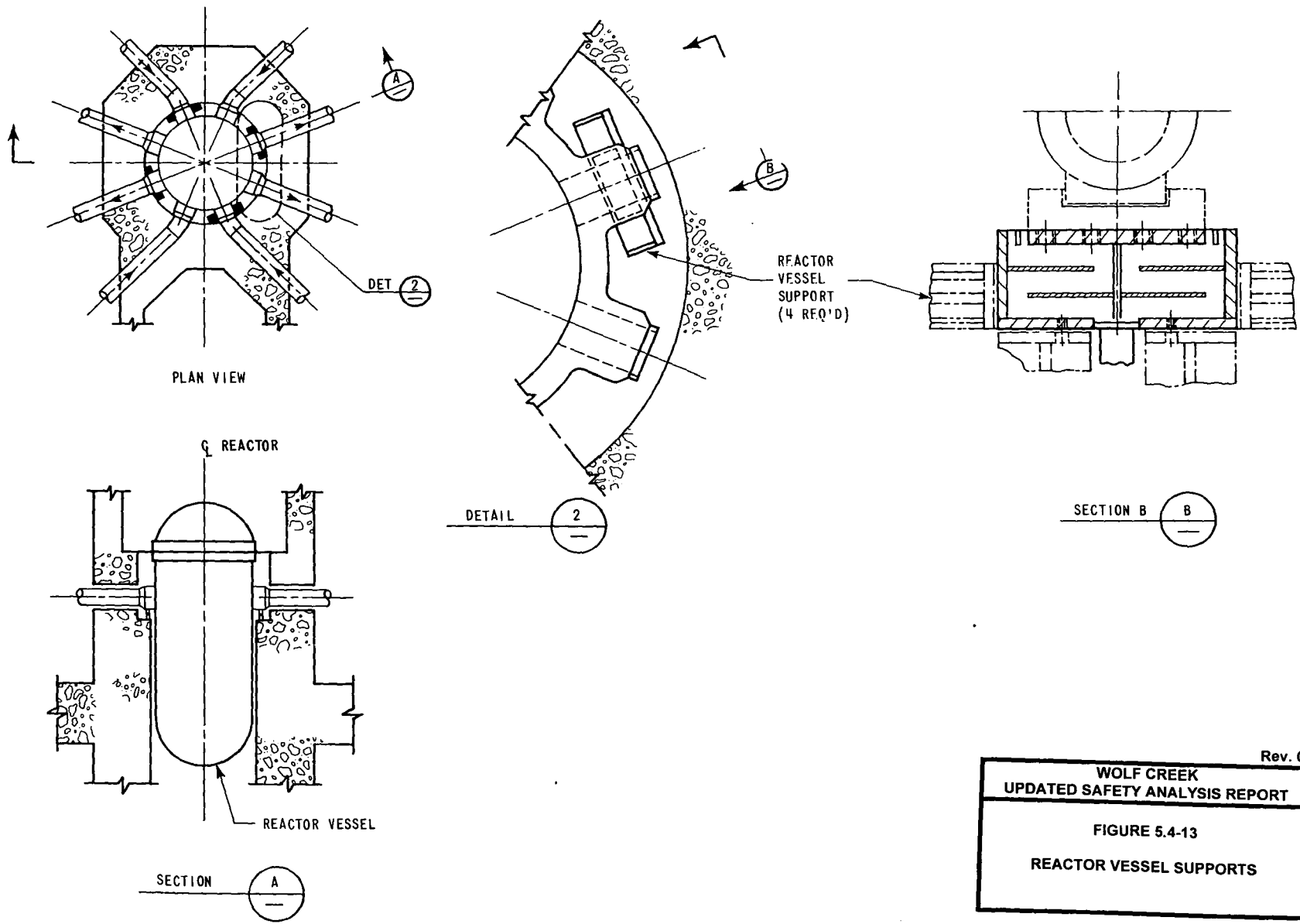


<p>WOLF CREEK UPDATED SAFETY ANALYSIS REPORT</p>
<p>FIGURE 5.4-11</p> <p>PRESSURIZER</p>
<p>Rev. 0</p>



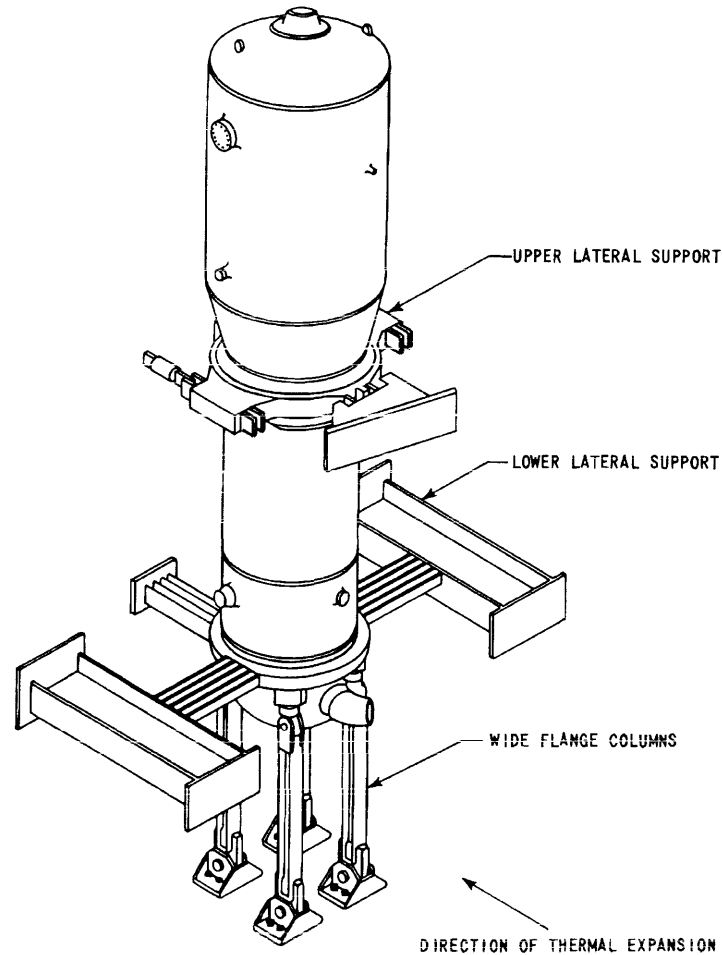
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<p>WOLF CREEK UPDATED SAFETY ANALYSIS REPORT</p>
<p>FIGURE 5.4-12 PRESSURIZER RELIEF TANK</p>



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FIGURE 5.4-13
REACTOR VESSEL SUPPORTS

13,557-34



USAR FIG. 5.4-14 REV. 9

WOLF CREEK
NUCLEAR OPERATING CORPORATION

STEAM GENERATOR SUPPORTS

SCALE

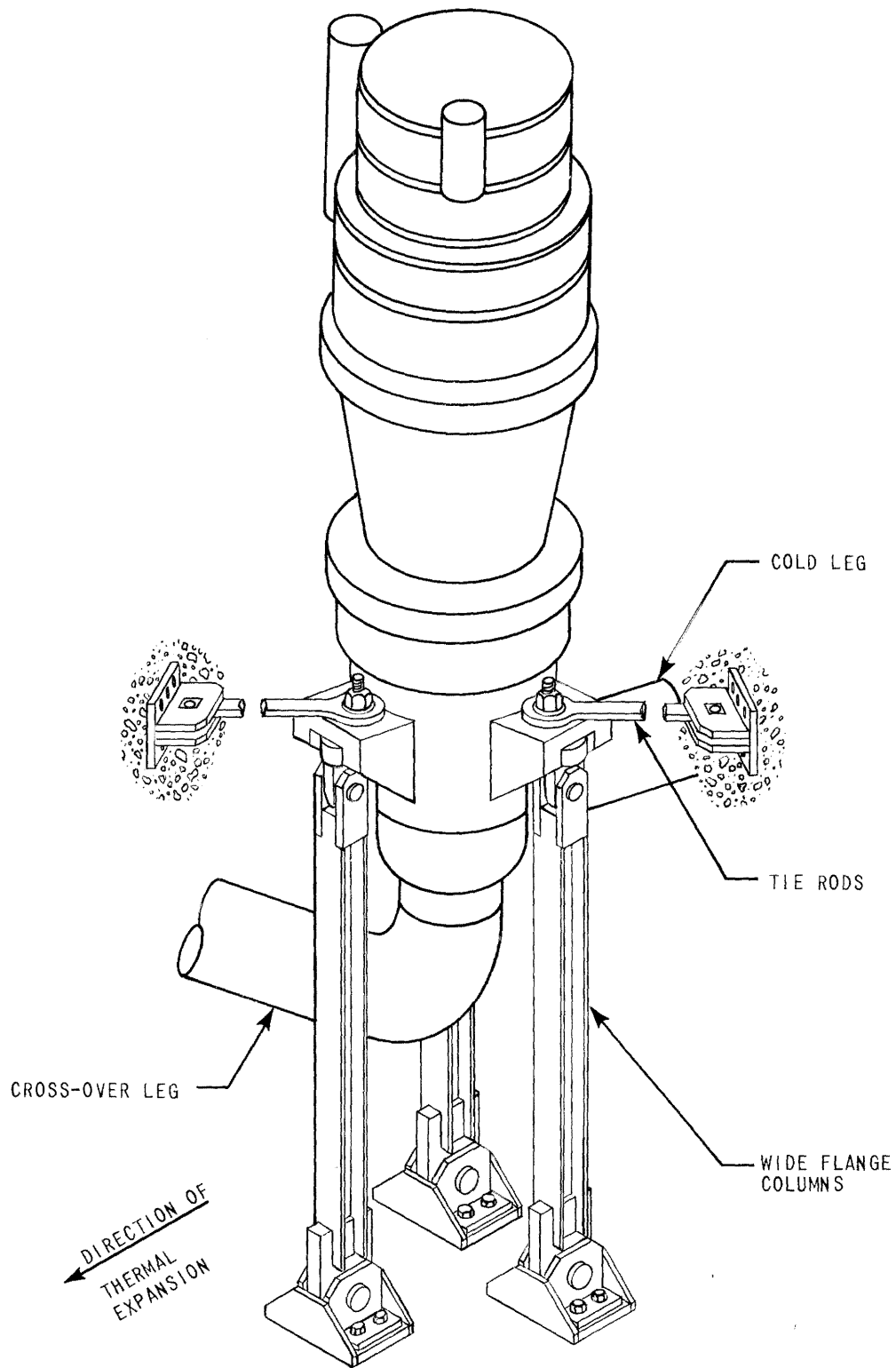
DRAWING NUMBER

SHEET

REV

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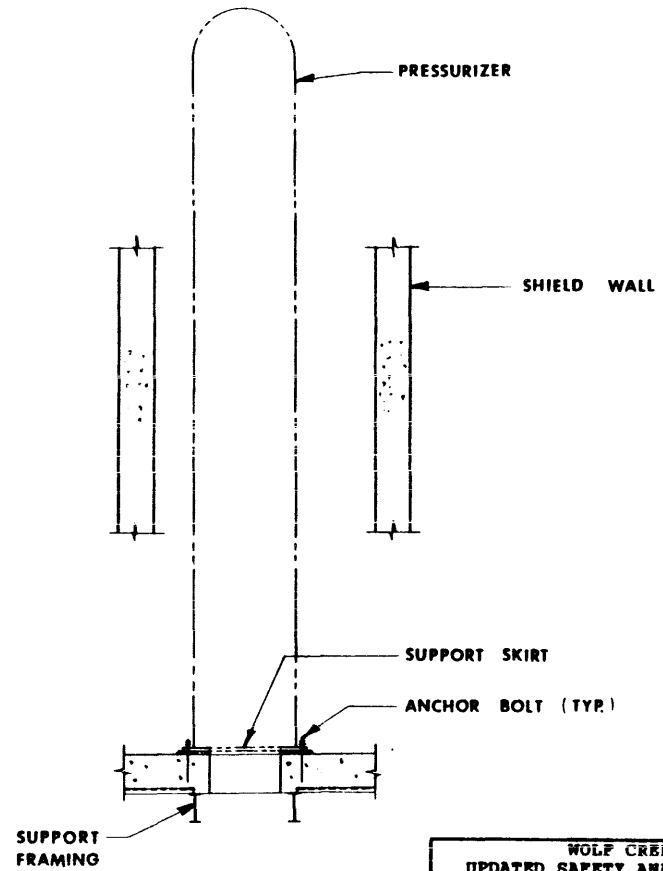
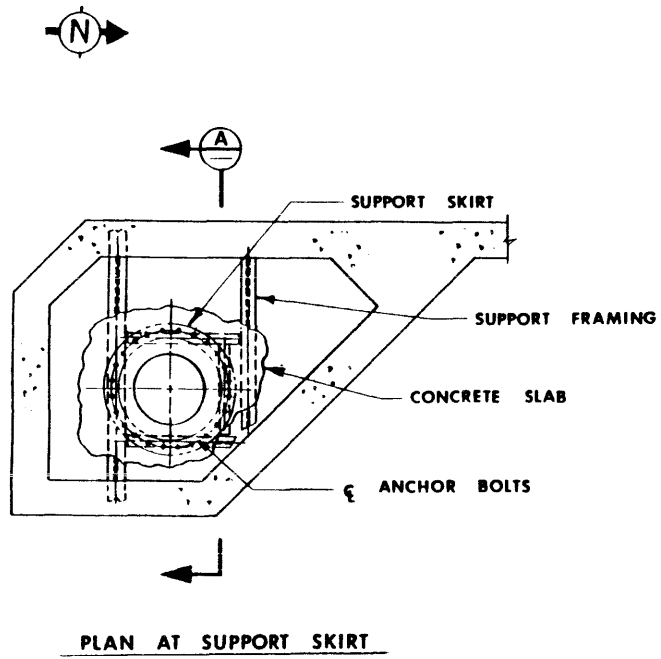
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<p>WOLF CREEK UPDATED SAFETY ANALYSIS REPORT</p>
<p>FIGURE 5.4-15 REACTOR COOLANT PUMP SUPPORTS</p>

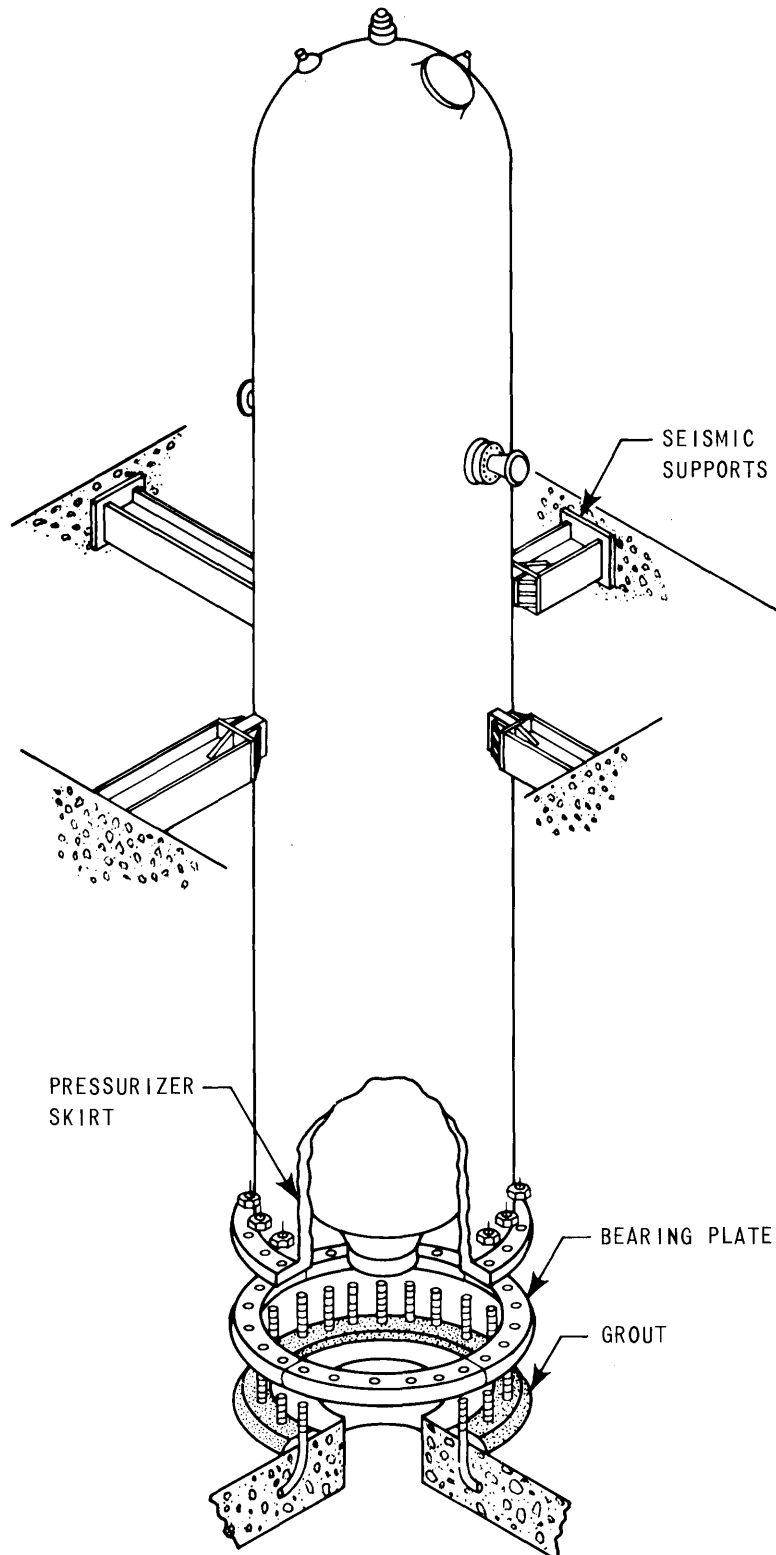
WOLF CREEK



SECTION A

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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 5.4-16 REACTOR BUILDING INTERNALS PRESSURIZER SUPPORTS

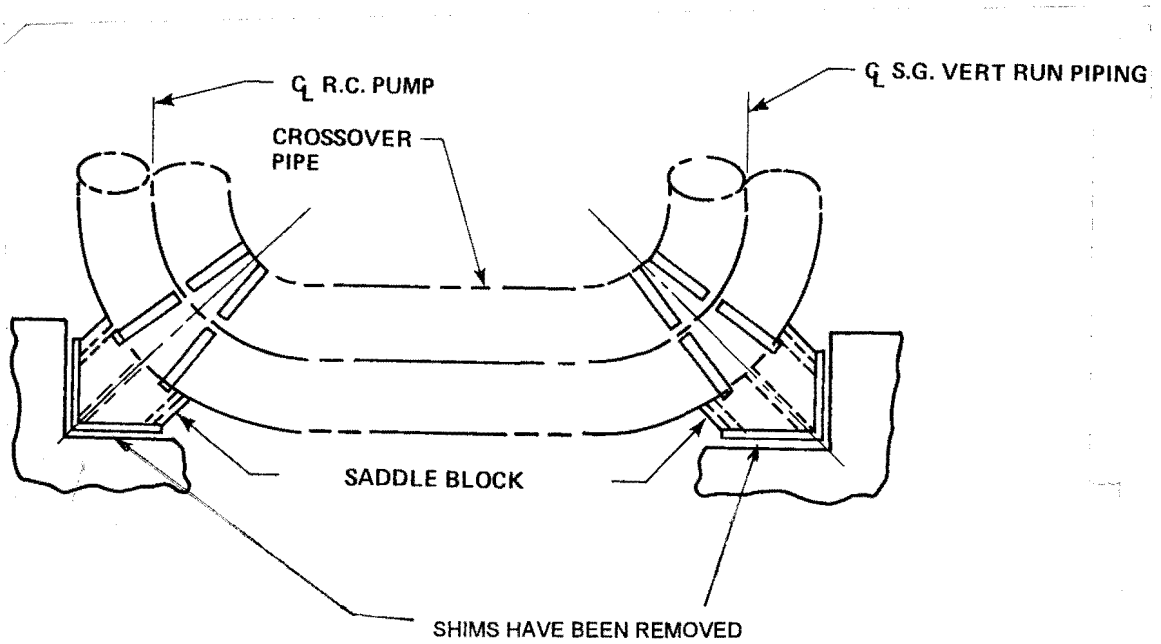


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FIGURE 5.4-17

PRESSURIZER SUPPORTS



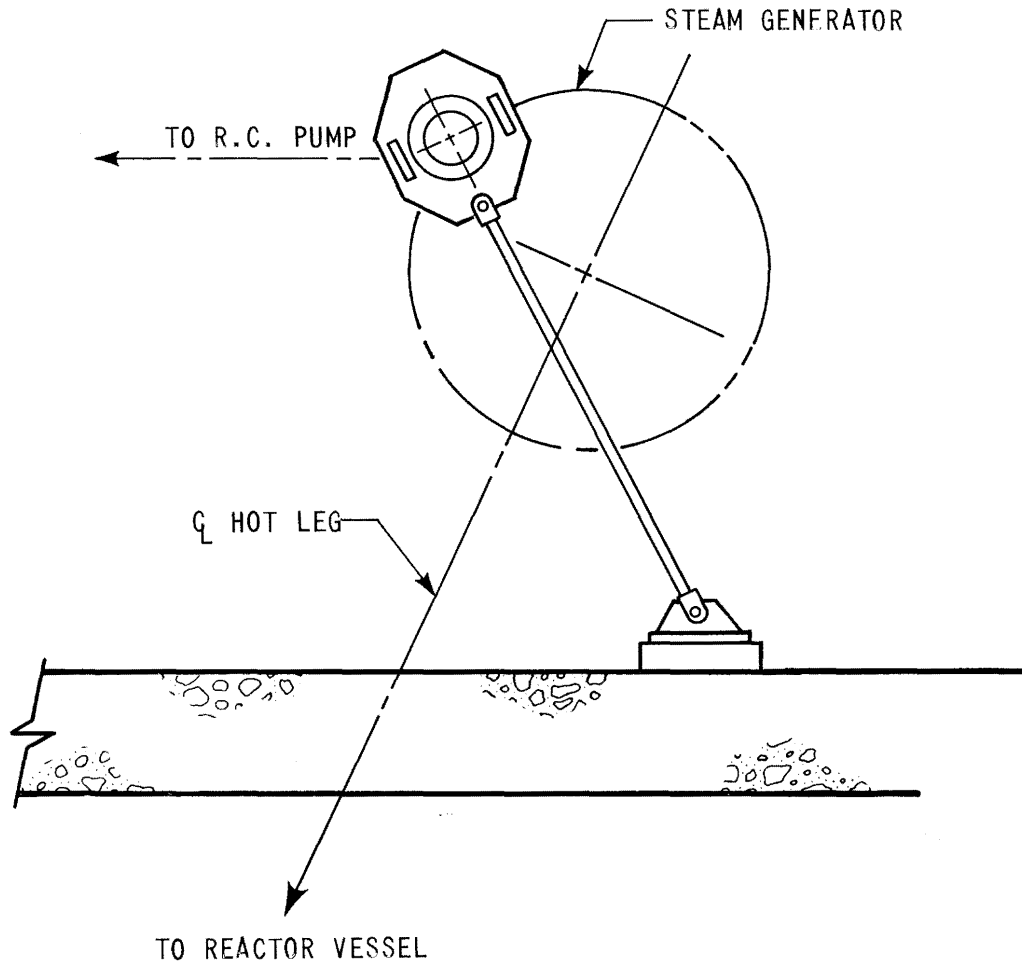
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FIGURE 5.4-18

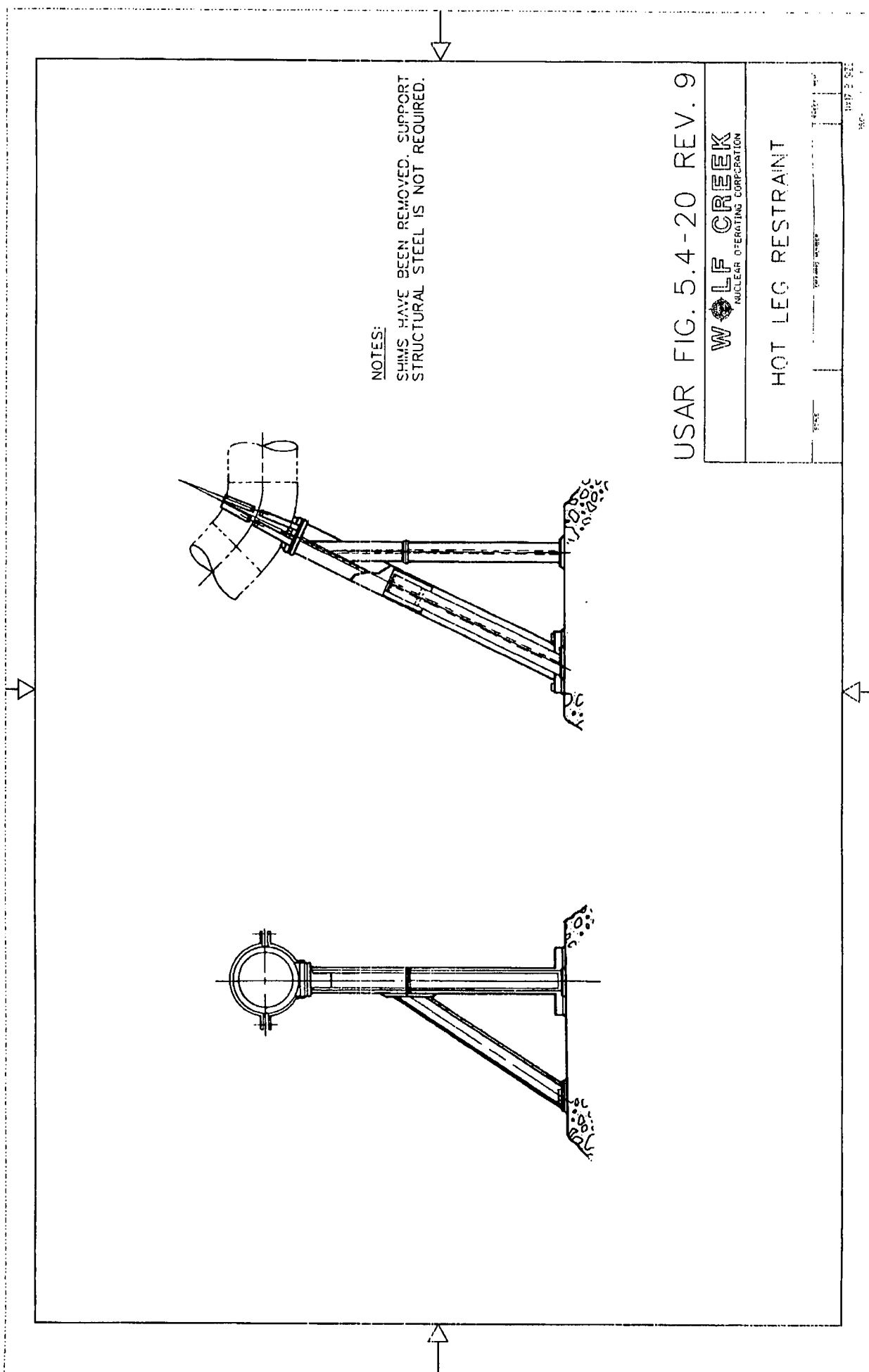
CROSSOVER LEG SUPPORTS

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FIGURE 5.4-19 CROSSOVER LEG VERTICAL RUN RESTRAINT (DELETED IN 5TH REFUELING OUTAGE)



NOTES:

SHIMS HAVE BEEN REMOVED. SUPPORT STRUCTURAL STEEL IS NOT REQUIRED.

USAR FIG. 5.4-20 REV. 9

WOLF CREEK
NUCLEAR OPERATING CORPORATION

HOT LEG RESTRAINT

DATE: _____
DRAWN BY: _____
CHECKED BY: _____
SCALE: _____

SHEET 2 OF 2