5.1

5. REACTOR COOLANT SYSTEM

5.1 SUMMARY DESCRIPTION

The reactor coolant system (RCS) shown in figures 5.1-1 and 5.1-2 consists of 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, a pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control. All of 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 flowrate 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 provides a barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the plant.

RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Spring-loaded safety valves and power-operated relief valves are mounted on the pressurizer and discharge 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 reactor coolant side of the steam generators

C. Reactor coolant pumps

D. A pressurizer attached to one of the reactor coolant loops

E. The pressurizer relief tank

F. Safety and relief valves

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

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

5.1.1 REACTOR COOLANT SYSTEM COMPONENTS

5.1.1.1 Reactor Vessel

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

5.1.1.2 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.

5.1.1.3 Reactor Coolant Pumps

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

5.1.1.4 Piping

The reactor coolant loop piping is specified in sizes consistent with system requirements.

The hot leg inside diameter is 29 inches and the inside diameter of the cold leg return line to the reactor vessel is 27-1/2 inches. The piping between the steam generator and the pump suction is increased to 31 inches in inside diameter to reduce pressure drop and improve flow conditions to the pump suction.

5.1.1.5 Pressurizer

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

5.1.1.6 Pressurizer Relief Tank

The pressurizer relief tank is a horizontal, cylindrical vessel with elliptical dished heads. 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.

5.1.1.7 Safety and Relief Valves

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

5.1.2 REACTOR COOLANT SYSTEM PERFORMANCE CHARACTERISTICS

Tabulations of important design and performance characteristics of the RCS are provided in table 5.1-1.
System thermal and hydraulic data for different RCS average temperatures are shown in Table 5.1-1. 321

5.1.2.1 Reactor Coolant Flow

The reactor coolant flow, a major parameter in the design of the system and its components, is established with a detailed design procedure supported by operating plant performance data, by pump model tests and 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.

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

5.1.2.2 Best Estimate Flow

The best estimate flow is the most likely value for the actual plant operating condition. This flow is based on the best estimate of the reactor vessel, steam generator and piping flow resistance, 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 flowrates are applied in the thermal and mechanical designs.

5.1.2.3 Thermal Design Flow

Thermal design flow is the basis for the reactor core thermal performance, the steam generator thermal performance, and the nominal plant parameters used throughout the design. To provide the required margin, the thermal design flow accounts for the uncertainties in reactor vessel, steam generator and piping flow resistances, reactor coolant pump head, and the methods used to measure flowrate. The thermal design flow is approximately 4.9 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.

5.1.2.4 Mechanical Design Flow

Mechanical design flow is the conservatively high flow used in the mechanical design of the reactor vessel internals and fuel assemblies. To assure that a conservatively high flow is specified, the mechanical design flow is based on a reduced system resistance and on increased pump head capability. The mechanical design flow is approximately 4.2 percent greater than the best estimate flow.

Maximum pump overspeed results in a peak reactor coolant flow of 120 percent of the mechanical design flow. This overspeed condition, which is coincident with a turbine generator overspeed of 20 percent, is only applicable if, when a turbine trip would be actuated, the turbine governor fails and the turbine is tripped by the mechanical overspeed trip device.

5.1.2.5 Flows with One Pump Shut Down

The design procedure for calculation of flows with one pump shutdown 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 with a locked rotor pump impeller reverse flow resistance 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 flowrate for the thermal design. The mechanical design flow uncertainty is increased slightly to account for the slightly higher uncertainties at the higher pump flows. Thermal design, best estimate, and mechanical design flows for the three-loop plant are summarized in table 5.1-1. 382

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5.1.3 INTERRELATED PERFORMANCE AND SAFETY FUNCTIONS

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

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

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

C. The system heat removal capability under power operation and normal operational transients, including the transition from forced to natural circulation, shall assure 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 system maintains the homogeneity of soluble neutron poison concentration and rate of change of coolant temperature such that uncontrolled reactivity changes do not occur.

F. The reactor vessel is an integral part of the RCS pressure boundary and is capable of accommodating the temperatures and pressures associated with the operational transients. The reactor vessel functions to support the reactor core and control rod drive mechanisms.

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

H. The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.

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

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

K. Portions of the RCS are relied upon to function in conjunction with other equipment of the cold shutdown design during a safety grade cold shutdown. It is expected that the systems that are used for cold shutdown will be available any time the operator chooses to perform a reactor cooldown. Should only safety grade equipment be available, letdown of excessive coolant from the RCS and system depressurization can both be accomplished. Addition to the RCS inventory, as through boration, can be relieved via the reactor vessel head letdown line, and depressurization of the RCS is accomplished by venting through use of the pressurizer power operated relief valves. Details of the cold shutdown design are discussed in subsection 5.4.7.

5.1.4 SCHEMATIC FLOW DIAGRAM

The reactor coolant system is shown schematically in figure 5.1-2. Included in this figure is a tabulation of primary pressures, temperatures, and the flowrate of the system under normal steady-state full power operating conditions. These parameters are based on the best estimate flow at the pump discharge. Reactor coolant system volume under the above conditions is presented in table 5.1-1.

5.1.5 PIPING AND INSTRUMENTATION DIAGRAM

A piping and instrumentation diagram of the reactor coolant system is shown in figure 5.1-1. The diagram shows the extent of the systems located within the containment, the points of separation between the reactor coolant system, and the secondary (heat utilization) system.

5.1.6 ELEVATION DRAWINGS

Drawings showing the principal dimensions of the reactor coolant system in relation to the surrounding concrete structures are provided in figures 5.1-3 and 5.1-4.

Table 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

(Sheet 1 of 2)

Plant Design Life, years	40	
Nominal Operating Pressure, psig	2,235	
Total System Volume Including Pressurizer and Surge Line, ft^3	9,275	
System Liquid Volume, Including Pressurizer water at Maximum Guaranteed Power, ft^3	8,700	
Pressurizer Spray Rate, minimum required at full flow, gpm	700	
Pressurizer Heater Capacity, kwt	1,400	
Pressurizer Relief Tank Volume, ft^3	1,300	
System Thermal and Hydraulic Data		
RCS Tavg Cases (°F)	587	580
NSSS Power, MWt	2,912	2,912
Reactor Power, MWt	2,900	2,900
Thermal Design Flows, gpm		
Active Loop	**93,200**	**93,200**
Reactor	**279,900**	**279,900**
Total Reactor Flow, 10^6 lb/hr	**104.7**	**105.7**
Temperatures, °F		
Reactor Vessel Outlet	**621.3**	**614.7**
Reactor Vessel Inlet	**552.7**	**545.3**
Steam, generator Outlet	**552.4**	**545.0**
Steam Generator Steam	**533.3**	**527.5**
Feedwater	445.9	445.9

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Table 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

(Sheet 2 of 2)

System Thermal and Hydraulic Data			
Steam Pressure, psia	**811.0**	**867.0**	
Total Steam Flow, 10^6 lb/hr	**12.43**	**12.81**	
Best Estimate Flows, gpm			376
Active Loop	**100,000**	**102,400**	
Reactor	**300,000**	**307,200**	
Mechanical Design Flows, gpm			321
Active Loop	106,900	106,900	
Reactor	320,700	320,700	

System Pressure Drops		
Reactor Vessel ΔP, psi	41.0	
Steam Generator ΔP, psi	41.0	
Hot Leg Piping ΔP, psi	1.4	
Pump Suction Piping ΔP, psi	3.4	5
Cold Leg Piping ΔP, psi	3.4	
Pump Head, feet	280	

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관련 도면목록 :Table 1.7-2(Sh. 4 of 12)
관리도면 : P&ID , 3-M-BB-F001

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REACTOR COOLANT SYSTEM(RCS)
(Sheet 1 of 4)

Figure 5.1-1

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2008.09.17

Amendment 539
2015.11.19

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REACTOR COOLANT SYSTEM(RCS)
UNIT 3 ONLY
(Sheet 2 of 4)
Figure 5.1-1

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KOREA HYDRO & NUCLEAR POWER COMPANY
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REACTOR COOLANT SYSTEM(RCS)
UNIT 4 ONLY
(Sheet 3 of 4)
Figure 5.1-1

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REACTOR COOLANT SYSTEM (RCS)
(Sheet 4 of 4)
FIGURE 5.1-1

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KOREA ELECTRIC POWER CORPORATION

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REACTOR COOLANT SYSTEM
PROCESS FLOW DIAGRAM
(Sheet 1 of 3)

FIGURE 5.1-2

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KOREA ELECTRIC POWER CORPORATION
KRN 3 & 4 FSAR

REACTOR COOLANT SYSTEM
PROCESS FLOW DIAGRAM
(Sheet 2 of 3)

FIGURE 5.1-2

KOREA ELECTRIC POWER CORPORATION

KRN 3 & 4 FSAR

REACTOR COOLANT SYSTEM
PROCESS FLOW DIAGRAM
(Sheet 3 of 3)

FIGURE 5.1-2

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KOREA ELECTRIC POWER CORPORATION
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REACTOR COOLANT SYSTEM
LOOP PIPING

FIGURE 5.1-3

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

REACTOR COOLANT SYSTEM
SURROUNDING STRUCTURES

Figure 5.1-4

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. In this context, the RCPB is as defined in Section 50.2 of 10 CFR Part 50. In that definition, the RCPB extends to the outermost containment and is connected to the reactor coolant system (RCS). Since other sections of this FSAR describe the components of the auxiliary fluid systems, this section will be limited to the components of the RCS as defined in section 5.1, unless otherwise noted.

For additional information on the RCS and for components which are part of the RCPB but are not described in this section, refer to the following sections:

Section 6.3	--	For discussions of the RCPB components which are part of the emergency core cooling system
Subsection 9.3.4	--	For discussions of the RCPB components which are part of the chemical and volume control system
Subsection 3.9.1	--	For discussions of the design loadings, stress limits, and analyses applied to the RCS and ASME Code Class 1 components
Subsection 3.9.3	--	For discussions of the design loadings, stress limits, and analyses applied to ASME Code Class 2 and 3 Components

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10 CFR Section 50.55a

RCS components will be designed and fabricated in accordance with the rules of 10 CFR 50, Section 50.55a, Codes and Standards. The addenda of the ASME Code applied in the design of each component is listed in table 5.2-1.

5.2.1.2 Applicable Code Cases

Code cases used are listed in table 5.2-1a.

5.2.2 OVERPRESSURE PROTECTION

Reactor coolant system overpressure protection is accomplished at operating conditions by the utilization of 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, Paragraph NB-7300 and NC-7300, for the pressurized water reactor system.

5.2.2.1 Design Bases

Overpressure protection is provided for the RCS by the pressurizer safety valves which discharge through a common header to the pressurizer relief tank. The transient on which the design requirements are set for the primary system overpressure protection equipment is a complete loss of steam flow to the turbine with credit taken for steam generator safety valve operation and main feedwater flow maintained. For this transient, the peak RCS and peak steam system pressure must be limited to 110 percent of their respective design values. However, for the sizing of the pressurizer safety valves, no credit is taken for reactor trip nor for the operation of the following:

Pressurizer power-operated relief valves
Steam line power-operated relief valve
Steam dump system
Reactor control system
Pressurizer level control system
Pressurizer level spray valve

For the analysis, it is assumed that (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 (including instrumentation and control systems errors). These are the most limiting assumptions with respect to system overpressure.

Overpressure protection for the steam system is provided by the 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 must be done by 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 these pressure relieving devices are discussed in subsection 5.4.11.

Blowdown systems for the steam generator safety valves are discussed in subsection 10.3.2.

Postulated events and transients on which the design requirements of the overpressure protection system are based are discussed in reference 1.

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 and an analysis of the capability of the system to perform its function is presented in reference 3. The report 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 2.

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

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 common header.

The steam system safety valves are discussed in subsection 10.3.2.

5.2.2.4 Equipment and Component Description

The operation, significant design parameters, number and types of operating cycles, and environmental qualification of the pressurizer safety valves are discussed in subsection 5.4.13.

A discussion of the equipment and components of the steam system overpressure system is presented in subsection 10.3.2.

5.2.2.5 Mounting

Piping reaction loads on the safety valves are limited to acceptable values as discussed in paragraph 3.9.3.3. Mounting brackets are provided on the pressurizer and are used to support the pressurizer safety valves. Mounting of the components of the steam system overpressure protection system is discussed in section 10.3.

5.2.2.6 Applicable Codes and Classification

The requirements of ASME Boiler and Pressure Vessel Code Section III, 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 are shown on figure 5.1-1, sheets 1 and 2.

For further information, refer to section 3.9.

5.2.2.7 Material Specifications

Refer to subsection 5.2.3 for a description of this topic.

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. Steam leakage through each safety valve can also be detected by use of an acoustic leak monitor located on each valve discharge line. Indication of leakage and an adjustable alarm are provided on a console located in the control room. For a further discussion on process instrumentation associated with the system, refer to chapter 7.

5.2.2.9 System Reliability

The reliability of the pressure relieving devices is discussed in section 4 of reference 1. Operational data on pressurizer
3 safety valves and PORVs is also provided in Appendix I of reference 7.

5.2.2.10 Testing and Inspection

Testing and inspection of the overpressure protection components are discussed in paragraph 5.4.13.4 and chapter 14.

5.2.2.11 RCS Pressure Control During Low Temperature Operation

Administrative procedures are available to aid the operator in controlling RCS pressure during low temperature operation. However, to provide a backup to the operator and to minimize the frequency of RCS overpressurization, an automatic system is provided to control any inadvertent pressure excursion.

Protection against such overpressurization events is provided through use of two safety grade power-operated relief valves(PORVs) and two safety grade relief valves to mitigate any potential pressure transients. Analyses have shown that one PORV or one relief valve is sufficient to prevent violation of these limits due to anticipated mass and heat input transients. The mitigation system is required only during low temperature water solid operation. It is manually armed and automatically actuated. 447

5.2.2.11.1 System Operation

Two pressurizer power-operated relief valves are each supplied with actuation logic to ensure that an independent RCS pressure control back-up feature is available to the operator during low temperature operations. This system provides the capability for additional RCS inventory letdown, thereby maintaining RCS pressure within allowable limits. Refer to section 7.6 and subsections 5.4.7, 5.4.10, 5.4.13, and 9.3.4 for additional information on RCS pressure and inventory control during other modes of operation.

The basic function of the system logic is to continuously monitor RCS temperature and pressure conditions whenever plant operation is at low temperatures. An auctioneered system temperature will be continuously converted to an allowable pressure and then compared to the actual RCS pressure. The system logic will first annunciate a main control board alarm whenever the measured pressure approaches the setpoint. The operator arms the system and an actuation signal is transmitted to the actuation device which automatically opens the power-operated relief valves when required to prevent pressure temperature limits derived from Appendix G of 10 CFR 50 from being exceeded.

Also, overpressure protection of the RCS during low-temperature conditions is provided by the relief valves located in the RHR inlet lines. The RHR relief valves are shown on Figure 5.4-5 and describes in Subsection 5.4.7.
Alignment of the RHR relief valve to the RCS is provided via plant procedures to ensure RCS overpressure protection for all temperatures below the maximum low-temperature overpressure protection (LTOP) required temperature (T_{enable}). 447

5.2.2.11.2 Evaluation of Low Temperature Overpressure Transients

<u>Pressure Transient Analysis</u>

ASME Section III, Appendix G, establishes guidelines and limits for RCS pressure primarily for low temperature conditions (≤275°F). The relief system discussed in Subsection 5.4.13 addresses these guidelines. 621

Transient analyses were performed to determine the maximum pressure for the postulated worst case mass input and heat input events. Both heat input and mass input analyses took into account the single failure criteria and therefore, only one power-operated relief valve (PORV) <u>or one relief valve</u> was assumed to be available for pressure relief. 447
The evaluation of the transient results concludes that the guidelines of ASME III, Appendix G are met.

5.2.2.11.3 Operating Basis Earthquake Evaluation

The KNU 3 & 4 power-operated relief valves have been designed in accordance with the ASME Code to provide the integrity required for the reactor coolant pressure boundary and qualified in accordance with the Westinghouse valve operability program which is described in detail in paragraph 3.9.3.2.
The RHR suction line relief valves, isolation valves, associated interlocks and instrumentation are designed to seismic Category I requirements as discussed in Subsections 3.2.1 and 5.4.7.2.5 and Table 3.2-1. The interlocks and instrumentation associated with the RHR suction isolation valves satisfy the appropriate portions of IEEE 279 criteria as discussed in Subsections 5.4.7.2.4 and 7.6.2. 447

5.2.2.11.4 Administrative Procedures

Although the system described in subparagraph 5.2.2.11.1 is designed to maintain RCS pressure within allowable limits, administrative procedures have been provided for minimizing the potential for any transient that could actuate the overpressure relief system. The following discussion highlights these procedural controls, listed in hierarchy of their function, for preventing RCS cold overpressurization transients.

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

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

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

A. The residual heat removal inlet lines from the reactor coolant loop shall not be isolated unless the charging pumps are stopped. This precaution is to assure there 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 (less than 500 psi) and is water solid.

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

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

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

E. During plant cooldown, all steam generators shall be connected to the steam header to assure a uniform cooldown of the reactor coolant loops.

F. At least one reactor coolant pump shall be maintained in service until the reactor coolant temperature is reduced to 160°F. | 433

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

The specific plant configurations of emergency core cooling system testing and alignment will also highlight procedures required to prevent developing cold overpressurization transients. During these limited periods of plant operation, the following recommendations will be made:

A. To preclude inadvertent emergency core cooling system actuation during heatup and cooldown, blocking of the low pressurizer pressure and low steam line pressure safety injection signal actuation logic at approximately 1900 psig is required.

B. During further cooldown, closure and power lockout of the accumulator isolation valves and power lockout to the nonoperating charging pumps is required at approximately 1000 psig, and at a RCS temperature of approximately 425°F providing additional backup to step A above. 433

C. Periodic emergency core cooling system pump performance testing requires the testing of the pumps during normal power operation or at hot shutdown conditions. This precludes any potential for developing a cold overpressurization transient.

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

   If cold shutdown testing with the vessel closed is necessary, the procedures will require emergency core cooling system (ECCS) discharge valve closure and residual heat removal system (RHRS) alignment to both isolate potential emergency core cooling system pump input and to provide backup benefit of the RHRS relief valves.

D. "S" signal circuitry testing, if done during cold shutdown, will also require RHRS alignment and safety injection (SI) and nonoperating charging pump power lockout to preclude developing cold overpressurization transients.

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

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 reactor coolant pressure boundary (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 unstabilized austenitic stainless steel material specifications used for components in systems required for reactor shutdown and for emergency core cooling.

The unstabilized austenitic stainless steel materials for the reactor vessel internals, which are required for core support for any mode of normal operation or under postulated accident conditions and for core structural load-bearing members, are listed in table 5.2-4.

The materials utilized conform to the applicable ASME code rules. It should be noted that these material specifications are typical for the listed applications. In the course of material and component procurement, substitutions may be necessitated by material unavailability, construction schedules, or other considerations. If substitutions are made, materials of equivalent specifications are chosen; these materials are also procured in accordance with the requirements of the ASME code.

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 tested and qualified to the requirements of ASME Section III.

In addition, 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:

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

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

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

5.2.3.2 Compatibility With Reactor Coolant

5.2.3.2.1 Chemistry of Reactor Coolant

The reactor coolant system (RCS) chemistry specifications are given in table 5.2-5.

The RCS water chemistry is selected to minimize corrosion. A routinely scheduled analysis of the coolant chemical composition is 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 control the pH of the coolant during pre-startup testing and subsequent operation, scavenge oxygen from the coolant during heatup, and control radiolysis reactions involving hydrogen, oxygen, and nitrogen during all power operations subsequent to startup. The limits specified for chemical additives and reactor coolant impurities for power operation are shown in table 5.2-5.

The pH control chemical specified is lithium hydroxide monohydrate, enriched in Li^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, Li^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. If the concentration exceeds this range, the cation bed demineralizer is employed in the letdown line in series operation with the mixed bed demineralizer.

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

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 such 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 to accomplish long term reactivity control of the core. The mechanism for the process involves the absorption of neutrons by the B^{10} isotope of naturally occurring boron.

Zinc (as depleted zinc), in the chemical form of zinc acetate, is added to lower radiation dose rates during normal steady state operation. The target zinc concentration in the RCS is 5 ppb and no greater than 20 ppb. 516

Suspended solids (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the CVCS mixed bed demineralizer. Use of the reactor coolant purity control system provides additional capability for cleanup of reactor makeup water and boric acid solution (see subsection 9.3.4).

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 are provided with corrosion resistant cladding on all surfaces that are exposed to the reactor coolant. The corrosion resistance of this 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 on ferritic type base materials receives a post weld heat treatment, as required by the ASME Code.

Ferritic low alloy and carbon steel nozzles are safe ended with either stainless steel wrought materials, stainless steel weld metal analysis A-7 (designated A-8 in the 1974 Edition of the ASME Code), or nickel-chromium-iron alloy weld metal F-Number 43. The latter buttering material requires further safe ending with austenitic stainless steel base material after completion of the post weld heat treatment when the nozzle is larger than a 4-inch nominal inside diameter and/or the wall thickness is greater than 0.531 inches.

All of the austenitic stainless steel and nickel-chromium-iron alloy base materials, except for the steam generator tubes, with primary pressure retaining applications are used in the solution anneal heat treat condition (the steam generator tubes are in the thermally treated condition). These heat treatments are as required by the material specifications.

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

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

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

In general, all of the materials listed in 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 reactor coolant pressure boundary is either reflective stainless steel type or made of compounded materials which yield low leachable chloride and/or fluoride concentrations. The compounded materials in the form of blocks, boards, cloths, tapes, adhesives, cements, etc. are silicated to provide protection of austenitic stainless steels against stress corrosion which may result from accidental wetting of the insulation by spillage, minor leakage or other contamination from the environmental atmosphere. 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 in-service visual and/or nondestructive inspection program to assure the integrity of the component for subsequent service.

5.2.3.3 <u>Fabrication and Processing of Ferritic Materials</u>

5.2.3.3.1 Fracture Toughness

Assurance of adequate fracture toughness of ferritic materials in the reactor coolant pressure boundary is provided by compliance with the requirements of ASME Code, Section III and 10 CFR 50, Appendix G, as appropriate.

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

Limiting steam generator and pressurizer RT_{NDT} temperatures are guaranteed at 60°F for the base materials and the weldments. These materials will meet the 50 ft-lbs 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 are retained by KHNP. 433 554

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 reactor coolant pressure boundary. The results of the program are documented in Reference 7, which has been submitted to the NRC for review (via letter NS-CE-1730 dated March 17, 1978, to Mr. J.F. Stolz, NRC Office of Nuclear Reactor Regulation, from Mr. C. Eicheldinger, Westinghouse PWRSD Nuclear Safety).

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 reactor coolant pressure boundary with Regulatory Guides 1.34, Control of Electroslag Properties, 1.43, Control of Stainless Steel Weld Cladding of Low-Alloy Steel, 1.50 Control of Preheat Temperature for Welding of Low Alloy Steel, 1.66, Nondestructive Examination of Tubular Products, and 1.71, Welder Qualification for Areas of Limited Accessibility.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

Subparagraphs 5.2.3.4.1 to 5.2.3.4.5 address Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," present the methods and controls utilized by Westinghouse, 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

It is required that all austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems be handled, protected, stored, and cleaned according to recognized and accepted methods which are designed to minimize contamination which could lead to stress corrosion cracking. The rules covering these controls are stipulated in the Westinghouse Electric Corporation process specifications. As applicable, these process specifications supplement the design specifications and purchase order requirements for every individual austenitic stainless steel component or system which Westinghouse procures for the KNU 5 & 6 nuclear steam supply system (NSSS), 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 include the following:

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.
84351NL	Determination of Surface Chloride and Fluoride on Austenitic Stainless Steel Materials.

Process Specification Number	
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 Systems.

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

5.2.3.4.2 Solution Heat Treatment Requirements

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

5.2.3.4.3 Material Inspection Program

The Westinghouse practice is that austenitic stainless steel materials of product forms with simple shapes need not be corrosion tested provided 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 262-70, Practice A or E, or as amended by Westinghouse Process Specification 84201 MW.

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

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, Westinghouse relies on the elimination of conditions A and B to prevent intergranular attack on wrought stainless steel components.

The water chemistry in the reactor coolant system of a Westinghouse pressurized water reactor (PWR) is rigorously controlled to prevent the intrusion of aggressive species. In particular, the maximum permissible oxygen and chloride concentrations are 0.005 ppm and 0.15 ppm, respectively. Reference 3 describes the precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage. The use of hydrogen over pressure precludes the presence of oxygen during operation. The effectiveness of these controls has been demonstrated by both laboratory tests and operating experience. The long-time exposure of severely sensitized stainless in early plants to PWR coolant environments has not resulted in any sign of intergranular attack. Reference 3 describes the laboratory experimental findings and the Westinghouse operating experience. The additional years of operations since the issuing of reference 3 have provided further confirmation of the earlier conclusions. Severely sensitized stainless steels do not undergo any intergranular attack in Westinghouse PWR coolant environments.

In spite of the fact there never has been any evidence that PWR coolant water attacks sensitized stainless steels, Westinghouse considers it good metallurgical practice to avoid the use of sensitized stainless steels in the NSSS components. Accordingly, measures are taken to prohibit the purchase of sensitized stainless steels and to prevent sensitization during component fabrication. Wrought austenitic stainless steel stock used for components that are part of 1) the reactor coolant pressure boundary, 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 five minutes.

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

Westinghouse recognizes that the heat-affected zones of welded components must, of necessity, be heated into the sensitization temperature range, 800 to 1500°F. However, severe sensitization, i.e., continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion, can still be avoided by control of welding parameters and welding processes. The heat input[a] and associated cooling rate through the carbide precipitation range are of primary importance. Westinghouse has demonstrated this by corrosion testing a number of weldments. 433

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. However, in only the socket weld did the sensitized condition exist at the surface, where the material is exposed to the environment; a material change has been made to eliminate this condition.

Westinghouse controls the heat input in all austenitic boundary weldments by:

A. Prohibiting the use of block welding

B. Limiting the maximum interpass temperature to 350°F 433

C. Exercising approval rights on all welding procedures.

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a. Heat input is calculated according to the formula:

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

where: H = Joules/in; E = Volts; I = Amperes; and
S = Travel Speed in inches/minutes.

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

Reactor Vessel Safe Ends
Pressurizer Safe Ends
Surge Line and Reactor Coolant Pump Nozzles
Control Rod Drive Mechanisms Head Adaptors
Control Rod Drive Mechanisms Seal Welds
Lower Instrumentation Penetration Tubes

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

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 to 1500°F temperature range. 433

C. Control of welding processes and procedures to avoid HAZ 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.

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

5.2.3.4.5 Retreating Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures

It is not normal Westinghouse practice to expose unstabilized austenitic stainless steels to the sensitization range of 800 to 1500°F during fabrication into components. If, during the course of fabrication, the steel is inadvertently exposed to the sensitization temperature range, 800 to 1500°F, the material may be tested in accordance with ASTM A262, Practice A or E, or 433 as amended by Westinghouse Process Specification 84201 MW 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 five percent or more,

B. Material with a carbon content of 0.03 percent or less that is subjected to temperatures in the range of 800 to 1500°F for less than one hour, 433

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

If it is not verified that such material is not susceptible to intergranular attack, the material will be re-solution annealed and water quenched or rejected.

5.2.3.4.6 Control of Welding

The following paragraphs address Regulatory Guide 1.31, Control of Stainless Steel Welding, 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 the controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated or stamped in accordance with ASME Boiler and Pressure Vessel 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[a] as determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III. When new welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, they are performed in accordance with the requirements of Section III and Section IX.

The results of all the destructive and non-destructive 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-8, 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 SFA-5.9 and are procured in a wire-flux combination to be capable of providing not less than 5 percent delta ferrite in the deposit according to Section III. Welding materials are tested using the welding energy inputs to be employed in production welding.

Combinations of approved heats 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 gauges and instruments, identification of "starting" and

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

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 non-destructive 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 4, 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 results, which do support the hypothesis presented in reference 4, are summarized in reference 5.

Appendix 3A includes discussions which indicate the degree of conformance of the austenitic stainless steel components of the reactor coolant pressure boundary with Regulatory Guides 1.34, Control of Electroslag Properties, 1.66, Nondestructive Examination of Tubular Products, and 1.71, Welder Qualification of Areas of Limited Accessibility.

5.2.4 INSERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

Inservice inspection and testing of pressure-retaining components, such as vessels, piping, pumps, valves, and bolting and supports associated with the reactor coolant pressure boundary, will comply with Section XI of the ASME B&PV Code, including addenda, in accordance with 10 CFR 50.55a(b)(2) and 10 CFR 50.55a(g), with certain exceptions when specific written relief is granted by the ROK-AEB per 10 CFR 50.55a(g)(6). The inservice testing of pumps and valves per the requirements of Articles IWP and IWV of the ASME Code, Section XI, respectively, are discussed in subsection 3.9.6.

In addition, KHNP will prepare a separate preservice and inservice inspection program document, including pump and valve testing, which complies with "NRC Staff Guidance for Complying with Certain Provisions of 10 CFR 50.55 a(g)--Inservice Inspection Requirements." This document will provide the details to the areas subject to examination, method of examination, extent of examination, and frequency. 554

Since the plant will be required to meet the requirements of future editions of Section XI, insofar as practical, provisions were made during design to allow access for inspections and coverages anticipated to be required by later editions of the Code. The result of this effort has increased the areas on the reactor pressure vessel (RPV) available to mechanized inservice inspection.

5.2.4.1 System Boundary Subject to Inspection

The system boundary subject to inspection includes all piping and components in Quality Group A (ASME B&PV Code, Section III, Class 1). The reactor vessel (RV), pressurizer, Class 1 portion of the steam generators, and all Class 1 piping, pumps, and valves will be examined in accordance with the requirements of the ASME B&PV Code, Section XI, 1977 Edition through the Summer 1978 Addendum.

A detailed plan for preservice and inservice inspections of Class 1 components and piping will be prepared as a separate document.

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 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 have been 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 ROK-AEB 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.

Access is provided for volumetric examination of the pressure-retaining 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 will minimize the time required for these inspections to be performed. Therefore, the amount of radiation exposure is reduced 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 will be provided with temporary platforms and/or scaffolding, as required.

5.2.4.2.2 Access to Reactor Vessel

Access for inspection of the RV will be as follows:

A. Performance of preservice and inservice examinations of the safe end and pipe-to-elbow welds of the vessel are done by access through an annular cavity located within the primary shield wall. Manholes at the 122-foot elevation are provided for entry into the annular cavity.

B. The vessel flange seal surface will be accessible during refueling outages when the closure head is removed. The vessel-to-flange weld and flange ligaments can be examined manually from the flange seal surface, using ultrasonic techniques.

C. Access to the inner surface of the RV will be available during refueling outages when the vessel core structure is removed. A remotely operated examination device designed to perform ultrasonic examinations from the inner surface of the vessel can be 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 will be 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 will be inserted into the RV.

D. The closure head will be stored dry during refueling, which will facilitate direct manual examination. Removable insulation will allow examination of the head welds from the outside surface. All reactor nuts, bolts, washers, and studs will be inspected during refueling.

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 hydrostatic and pressure testing.

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.

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 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 provide access required in these areas. Working platforms are provided in areas required to facilitate the servicing of pumps and valves and perform required testing in accordance with Articles IWP and IWV.

Temporary or permanent platforms and ladders will be provided, as necessary, to gain access to piping welds. Welds requiring inspection have been located to permit ultrasonic examinations from at least one side of the weld, and where component geometries permit, access to both sides 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 hydrostatic and pressure testing.

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.

5.2.4.2.7 Valve Pressure Boundaries

Class 1 valves over 1-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 techniques and procedures for visual, surface, and volumetric examinations, will be written in accordance with the requirements of Subarticle IWA-2200 and IWB-2500 of Section XI of the ASME Code. The liquid penetrant or magnetic particle methods will be utilized for surface examinations; ultrasonic 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 will be qualified prior to examinations in the plant.

5.2.4.3.1.1 Ultrasonic Equipment. The equipment for inservice inspection of the reactor pressure vessel circumferential and vertical welds below the 122-foot vessel elevation consists of a polar manipulator with various attachments. This equipment provides movement of ultrasonic transducers over all required lengths of the shell welds, nozzle-to-vessel welds, and safe end welds. The nozzle-to-vessel weld examination equipment will provide radial and circumferential motion to the ultrasonic transducer while rotating inside the nozzle. The pipe weld examination device will provide longitudinal and circumferential motion to the ultrasonic transducer while rotating inside the pipe.

The remotely operated device for examination of the vessel from the inner surface will be attached to the RPV at the flange surface. The device is capable of moving the transducers over the inner surface of the RV in any direction.

An electronic system with a receiver or data channel for each ultrasonic transducer will be 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 will be 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, width, and length of reflector. However, similar compatible systems of data acquisition may be utilized.

5.2.4.3.1.2 Surface Examination Equipment. Mechanized surface examination techniques will provide results which are at least equivalent to those obtainable by manual surface techniques.

5.2.4.3.1.3 Visual Examination Equipment. Remote visual examination techniques will provide a resolution capability which is at least equivalent to that obtainable by direct visual observation.

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, all reportable indications will be mapped and records made of maximum signal amplitude, depth below the scanning surface, and length of the reflector. The data compilation format will provide for comparison of data from subsequent examinations. In areas where manual surface or direct visual examinations are performed, all reportable indications will be 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, Inspection Program B, 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 shall be in accordance with IWB-2400. Inservice examinations will usually be performed during normal plant outages, such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval.

5.2.4.5 Examination Categories and Requirements

The extent of the examinations performed is in accordance with Subarticle IWB-2500.

In addition, preservice inspections comply with IWB-2200.

5.2.4.6 Evaluation of Examination Results

Evaluation of examination results for Class 1 components is conducted in accordance with Articles IWA-3000 and IWB-3000. Unacceptable indications will be repaired in accordance with the requirements of Article IWB-4000 of the ASME B&PV Code, Section XI.

5.2.4.7 System Leakage and Hydrostatic Tests

The hydrostatic test for the reactor pressure vessel and remainder of the reactor coolant pressure boundary will be conducted in accordance with the requirements of Articles IWA-5000 and IWB-5000. System leakage tests will be conducted prior to startup following each reactor refueling outage, in accordance with Paragraph IWB-5221, as required by Article IWB-5000. A system hydrostatic test will be conducted at or near the end of each inspection interval. Examinations performed during these tests will be conducted without the removal of insulation.

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant leakage detection systems are provided to detect leakage from the reactor coolant system into auxiliary systems and the containment, and to provide the means to locate such leakage. The safety significance of leaks from the reactor coolant pressure boundary (RCPB) can vary widely depending on the source of the leak as well as leakage rate and duration. Therefore, detection and monitoring of reactor coolant leakage is required.

The leakage detection systems provide information which indicates the need for corrective action should excessive leakage be identified.

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

5.2.5.1.2.1 Power Generation Design Basis One. For leaks of 1 gal/min or greater, other than identified leakage sources, the reactor coolant pressure 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 Regulatory Guide 1.45 is provided in table 5.2-6.

5.2.5.1.2.2 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 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

5.2.5.2.1.1 Leakage Classification. Reactor coolant pressure boundary leakage is classified as identified or unidentified and methods for physically separating the leakage into these classifications are provided to supply prompt and quantitative information about the leakage to the plant operators.

Identified leakage comprises:

A. Leakage (except controlled leakage) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

B. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage, or

C. Reactor coolant system leakage through a steam generator to the secondary system.

Unidentified leakage is all leakage which is not identified or controlled leakage.

5.2.5.2.1.2 Identified Leakage Detection. Certain components of the RCPB may have small amounts of leakage and cannot practically 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 sensor and then to the reactor coolant drain tank. The reactor coolant leakage gives a high temperature indication and alarm. Valves larger than 2 inches which are connected to the reactor coolant system have dual packing with a leak-off connection between the two packings piped to the containment normal sump. However, the controlled leakage shaft seal system for the reactor coolant pumps is piped to the reactor coolant drain tank. Safety and relief valve discharge lines are piped to the pressurizer relief tank. Temperature sensors and acoustic leak monitors are installed on the discharge lines to detect leakage past the valve seats. Leakage detected is alarmed in the control room. 427

5.2.5.2.1.3 <u>Detection Methods For Unidentified Leakages</u>. Leakage to the containment atmosphere from the RCPB would cause a change in the containment airborne radioactivity which would be detected by the containment air particulate and gas 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 the gross noble gas activities would reach an equilibrium level. Under these conditions any increase in monitored activity would be the result 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 variation and variations in letdown rate. However, significant increases in leakage can be detected. 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 and gas 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 any of the monitor channels indicate the reactor coolant pressure boundary as the source of leakage. Conversely, if the humidity detector detects increased containment moisture without a corresponding increase in airborne activity level, the indicated source of leakage would be judged to be a nonradioactive 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 later methods are expected to be noted first by the more sensitive radiation and moisture detection equipment.

Normally, unidentified leakage from the RCPB is essentially zero. The RCS is an all welded system, with the exception of the connections on the pressurizer safety valves, reactor vessel head, and the pressurizer and steam generator manways which are flanged. All connections to the RCS are welded. All isolation or check valves between the RCS and other systems have been designed for low-seat leakage, and RCPB check valve back-leakage is checked periodically. In general, valves in the RCS 2 inches and under are of the packless type. All valves larger than
6 2 inches have dual packing with a leak-off connection to the containment sump.

The RCPB leakage detection system consists of the sump level and flow (level versus time) monitoring system, the containment air particulate monitoring system, the containment radioactive gas monitoring system and the containment humidity monitoring system. Condensate from containment coolers drains into containment sump along with any other unidentified leakages from RCPB. The sump level and flow monitoring system indicates leakages by monitoring increases in sump level. The containment humidity measuring system detects leakage from the release of steam or water to the containment atmosphere. The air particulate and radioactive gas monitoring systems detect leakages from the release of the radioactive materials to the containment atmosphere. Primary-to-secondary reactor coolant leakage, if it occurs, is detected by the main condenser air ejector exhaust and the steam generator blowdown radiation monitors. The containment purge exhaust radiation monitors will identify any increase in radioactivity leakage inside the containment.

RCPB leakage is also indicated by increasing charging pump flow rate compared with RCS inventory changes and by unscheduled increases in reactor makeup water usage.

5.2.5.2.1.3.1 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 include isolation valves designed for low-seat leakage, periodic testing of RCPB check valves (see subsection 6.3.4) and inservice inspection (see subsection 5.2.4). Leakage will be 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 residual heat removal (RHR) system is isolated from the RCS on the suction side by motor-operated valves 8701A, 8702A and 8701B, 8702B. Leakage past these valves will be detected by lifting of the relief valves 8708A or 8708B accompanied by increasing pressurizer relief tank level, pressure, and temperature indications and alarms on the main control board (see figure 5.2-1).

B. Safety Injection System/Accumulators

The accumulators are isolated from the RCS by check valves 8956A, B, and C and 8948A, B, and C. Leakage past these valves and into the accumulator subsystem will be detected by redundant control room accumulator pressure and level indicators and alarms (see figure 5.2-2).

C. Safety Injection System/RHR Discharge Subsystem

The RHR pump portion of the safety injection system is isolated from the cold legs by check valves 8973A, B, and C, 8974A, B, and 8998A, B, and C; and from the hot legs by check valves 8988A, B and 8993A, B and by normally closed motor-operated valve 8889. Leakage past these valves will eventually pressurize the RHR discharge header and result in lifting of the relief valves 8864A, B and 8865. Relief valve lifting will be accompanied by control room indication and alarms of increasing equipment drain tank level, pressure and temperature (see figures 5.2-1 and 5.2-3).

D. Safety Injection System/Charging Pump Subsystem

The charging pump subsystem of the safety injection system is isolated from the reactor coolant system by check valves 8990A, B, and C, 8992A, B, and C, 8993A, B, and C, 8995A, B, and C, 8997A, B, and C, 8998A, B, and C and normally closed motor-operated valves 8801A and B, 8884, 8885 and 8886. Leakage past valve 8801A or B will result in redundant local indication of increasing high-level alarms. Leakage past valve 8884, 8885 or 8886 is not possible, since the valve inlet will be pressurized by the operating charging pump (see figures 5.2-3 and 5.2-4).

E. Waste Processing System

The waste processing systems are isolated from the RCS by manual valves 8057A, B, and C. Leakage past these valves will result in an increasing control room indication of the reactor coolant drain tank level and reactor coolant drain tank pump flow (see figure 5.2-5.

F. Head Gasket Monitoring Connection

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

G. Component Cooling Water.

Leakage from the RCS 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 system radiation monitors (see section 11.5).

5.2.5.2.1.3.2 Maximum Allowable Total Leakage. The limits for the RCPB boundary leakage are: identified, 10 gal/min and unidentified, 1 gal/min. When leakage has been identified, it will be evaluated by the operating staff to determine if the operation can safely continue. Under these conditions, an allowable total leakage from known sources of 10 gal/min has been established. Continued operation of the reactor with identified or unidentified leakage shall be in accordance with ITS Chapter 1 3.4.13. 357

For normal chemical and volume control system operation, the charging pump flow rate is 84 gal/min, which consists of 60 gal/min being charged through the normal charging line and 24 gal/min being supplied to the reactor coolant pump seals. Total flow leaving the reactor coolant system via the normal letdown path is 75 gal/min, with the remaining 9 gal/min returning via the seal water return line to the chemical and volume control system.

Maximum letdown flow is 120 gal/min with an additional 9 gal/min leaving via the seal water return line. A centrifugal charging pump with 150 gal/min rated capacity provides an adequate reserve capacity to make up for the leakage. Thus, under normal or maximum letdown flow conditions, a 10 gal/min maximum limit on RCPB leakage can be accommodated by operation of one charging pump.

The RCPB leakage detection system provides ample protection to assure that, in the unlikely event of a failure of the RCPB, small cracks will be 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 gal/min. This is well within the detection capability of the RCPB leakage detection system.

B. The time span for a crack to go from detectable size to critical size varies from five to more than 40 years(8). This assures adequate safety from a major loss-of-coolant accident (LOCA). 3

The above methods are supplemented by visual and ultrasonic inspections of the RCPB during plant shutdown periods, in accordance with the inservice inspection program (subsection 5.2.4).

5.2.5.2.2 Component Description

5.2.5.2.2.1 Containment Air Particulate Monitor. This monitor is a part of the containment airborne radioactivity monitoring system which takes continuously flowing samples from the containment atmosphere. The air samples are drawn outside the containment in a closed system and are passed through a moving paper filter which removes the particulate material.

The filter is located in a shielded counting chamber where a beta scintillation crystal with a photomultiplier tube measures the activity. Signal processing is provided by a digital signal processor. The monitor has a maximum sensitivity of 10-9 mCi/cm3 and a range of 5 decades. Filter paper is used once. The filter paper mechanism is provided as an integral part of the air particle detector assembly and its advance speed is adjustable in continuous movement and can be set by the operator. Filter paper can also move in step advancement mode. The filter collects more than 99 percent of the particulate matter larger than 0.3 micron.

The containment airborne radioactivity monitoring system is described further in the discussion of the containment radioactive gas monitor below and also in subparagraph 11.5.2.9.9.

5.2.5.2.2.2 Containment Radioactive Gas Monitor. The containment radioactive gas monitor determines gaseous activity in the containment by monitoring continuous air samples from the containment atmosphere as a part of the containment airborne radioactivity monitoring system. The samples flow continuously through a fixed shielded volume where its activity is monitored by a beta sensitive detector. The minimum detectable concentration level with the gas detector is 10-6 mci/cc for Kr-85 isotope. The sensor range is 5 decades.

After passing through the gas monitor, the sample is returned via a closed system to the containment atmosphere. The containment airborne radioactivity monitoring system is capable of performing its radioactivity monitoring functions following a safe shutdown earthquake. The sample transport system includes:

A. A pump to obtain the sample

B. An indicating flow control valve to provide the flow adjustment

C. A flow alarm assembly to provide a low flow alarm signal.

The containment airborne radioactivity monitoring system is also discussed in section 11.5.

5.2.5.2.2.3 Containment Humidity Measuring System. The containment humidity monitoring system, utilizing temperature-compensated humidity detectors, is provided to determine the water vapor content of the containment atmosphere. Containment humidity is recorded and alarmed high in the main control room.

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 0 to 100 percent relative humidity with a temperature range of 40 to 120°F. 433

5.2.5.2.2.4 Sump Level Monitoring System. Since a leak in the primary system will result in the reactor coolant flowing into the containment sump, leakage will be indicated by an increase in sump level. Indication of increasing sump level is transmitted from the sump to the control room level indicator and recorder by means of the level transmitter. The system provides measurement of low leakages by monitoring level increases versus time and alarming on a high rate of change.

5.2.5.2.2.5 Containment Purge Monitors. The containment purge system radioactivity monitors serve as a backup to the containment air particulate and gaseous airborne radioactivity monitoring system while the purge is in operation.

5.2.5.2.2.6 Gross Leakage Detection Methods. The following methods provide backup means of detecting RCPB leakage which are less sensitive than the systems and equipments identified above.

A. Charging Pump Operation

During normal operation, only one charging pump is operating. In the event of leakage of the reactor coolant, the flow rate of charging pump, correlated with RCS inventory changes as indicated by letdown flow, pressurizer level, and reactor coolant temperatures, gives an indication of leakage rate. All of these indications are provided in the control room.

B. Sump Pump Operation

Since a leak in the primary system will result in reactor coolant flowing into the containment sump, gross leakage can be indicated by an increase in frequency of operation of the containment sump pump. Pump operation and sump level are monitored from the control room.

C. Liquid Inventory

Gross 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.

5.2.5.2.3 Component Operation

5.2.5.2.3.1 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. The leakage flow rate can be determined from the count rate when the specific background radioactivity present before the leakage is known. 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. The containment air particulate monitor minimum detectable concentration is 10-9 mCi/cm3.

5.2.5.2.3.2 Containment Gaseous Radioactivity Monitors. 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 one year of normal operation) increases almost linearly for the first several hours after beginning of the leak. The leakage flow rate can be determined from the count rate when the specific background radioactivity present before the leakage begins, is known. The background activity is dependent upon the power level, percent failed fuel and natural radioactivity brought in by the containment purge. The containment gaseous activity monitor minimum detectable concentration is 10-6 mCi/cm3 for Kr-85.

5.2.5.2.3.3 Containment Humidity Monitoring System. Relative humidity sensors are provided for measurement in each reactor coolant pump area, the reactor cavity ventilation outlet and the containment purge outlet. These points are continuously monitored for increasing humidity which would be indicative of leakage into the containment. Containment humidity is recorded and high humidity is alarmed in the main control room.

5.2.5.2.3.4 Containment Sump Level and Flow Monitoring. Leakages from the primary reactor coolant systems is indicated by an increase in sump level. Indication of increasing sump level is provided by control room indicator and recorder. This system alarms high rate of change of level increases by monitoring level increases versus time to provide measurement of low leakages. A check of other instrumentation would be required to eliminate possible leakage from nonradioactive systems as a cause of an increase in sump level.

5.2.5.2.3.5 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.

5.2.5.2.3.6 Sump Pump Operation. Under normal conditions, the containment 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 radwaste building control room to alert the operators. Sump level indication in the main control room provides indication of sump pump operation.

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

5.2.5.2.3.7 Gross Leakage Detection Methods. These methods, listed below, indicate gross leakage but by their rather imprecise nature do not normally indicate small leaks as do the other systems.

A. Charging pump

The charging pump normally operates at 84 gal/min. 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 84 gal/min to maintain constant pressurizer level.

B. 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 control room flow indication will be checked frequently for unusual makeup rates not due to plant operations.

5.2.5.3 Leakage Detection Conversion to Leakage Equivalent

The procedures necessary for each leak detection method to be converted to a common leakage equivalent are indicated below.

5.2.5.3.1 Containment Radioactive Gas and Air Particulate and Purge Exhaust Monitoring

Upon actuation of a high activity alarm, the operator follows these steps:

A. Records particulate and gaseous activity measurements at equal time intervals

B. Determines the activity increases during these intervals

C. Estimates RCS activity level based on previous operating history

D. Estimates the coolant leakage rate from the radioactivity increases. This is accomplished by comparing the measured increases to a curve showing abnormal primary coolant leakage as a function of particulate and gaseous activity variation.

Estimated leakage and measured particulate and gaseous activities then are plotted to test the consistency of the data and to estimate the time elapsed since the beginning of the leak.

Nonconstant activity increases mean either:

A. Containment atmosphere activity is approaching steady-state, or

B. The abnormal leakage flow is not constant.

5.2.5.3.2 Sump Level Measuring System

The initiation of an additional or abnormal leak in the containment results in an increase in the flow rate to the sump. The additional sump flow initiates an excessive rate of change of level alarm in the control room as discussed in subparagraph 5.2.5.2.3.4.

Upon actuation of the excessive leakage alarm, the operator follows these steps:

A. Records sump level measurements at equal time intervals

B. Determines the sump level increase during these intervals

C. Determines the leakage from the measured sump level increases.

5.2.5.4 Safety Evaluation

Inasmuch as this system has no safety design basis, no safety evaluation is provided. Criteria for the selection of the safety design bases are stated in paragraph 1.1.2.2.

5.2.5.5 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 the calibration and maintenance procedures and frequencies for the containment radioactivity monitoring system is presented in ITS Chapter 1 3.4.15 357

The humidity measuring system is 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 the RCPB components are described in detail in subsection 5.2.4.

5.2.5.6 Instrumentation Application

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

A. Containment air particulate monitor - air particulate activity

B. Containment gaseous activity monitor - gaseous activity

C. Containment sumps level and rate of change of level alarm

D. Containment humidity measuring system - containment humidity (recording and alarm only)

E. Containment purge exhaust monitor - containment air particulate and gaseous activity. This monitor acts as a backup system to items A and B above.

F. Gross leakage detection methods - charging pump flow rate, letdown flow rate (indication and alarm only). 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. Indication of primary coolant makeup is provided.

For a further description of safety-related instrumentation, see section 7.5.

5.2.6 REFERENCES

1. "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972.

2. Campbell, L. A., "LOFTRAN Code Description," WCAP-7878 Rev. 1, January, 1977.

3. Golik, M. S., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary) and WCAP-7735 (Nonproprietary), August 1971.

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

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

6. "Dynamic Fracture Toughness to ASME SA508 Class 2 and ASME SA533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals," WCAP-9292, March 1978.

7. "Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.K.3.2 for Westinghouse NSSS Plants", WCAP-9804, February 1981.

3 8. "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loops," WCAP-8172, July 1973.

Table 5.2-1

APPLICABLE CODE ADDENDA FOR RCS COMPONENTS

Reactor Vessel	ASME III, 1977 Edition through Winter 1977
Steam Generator	ASME III, 1977 Edition through Winter 1977
Pressurizer	ASME III, 1977 Edition through Winter 1977
CRDM Housing	ASME III, 1977 Edition through Winter 1977
CRDM Head Adapter Housing	ASME III, 1977 Edition through Winter 1977
Reactor Coolant Pump	ASME III, 1977 Edition through Winter 1977
Reactor Coolant Pipe	ASME III, 1977 Edition through Winter 1977
Surge Lines	ASME III, 1977 Edition through Winter 1977
Valves	
Pressurizer Safety	ASME III, 1977 Edition through Winter 1977
Pressurizer PORV	ASME III, 1977 Edition through Summer 1979
Motor-Operated	ASME III, 1977 Edition through Winter 1977 and Winter 1978
Manual (3" and larger)	ASME III, 1977 Edition through Winter 1977
Control	ASME III, 1977 Edition through Winter 1977

Table 5.2-1a

APPLICABLE CODE CASES FOR RCS COMPONENTS

Reactor Vessel	*
Steam Generator	1528-3, 1484-3, N-265
Pressurizer	*
CRDM Housing	N-242-1
CRDM Head Adapter	*
Reactor Coolant Pump	N-242-1
Reactor Coolant Pipe	*
Surge Lines	*
Valves	
Pressurizer Safety	*
Pressurizer PORV	*
Motor-Operated	N-260
Manual (3" and larger)	*
Isolation	1649

*No code cases used.

Table 5.2-2

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS (Sheet 1 of 3)

Reactor Vessel Components	
Shell and Head Plates (other than core region)	SA533 Gr A, B or C, Class 1 or 2 (Vacuum treated)
Shell Plates (core region)	SA533 Gr A or B, Class 1 (vacuum treated)
Shell, Flange & Nozzle Forgings Nozzle Safe Ends	SA508 Class 2 or 3 SA182 Type F304 or F316
CRDM Head Adapter Housings - and/or ECCS Appurtenances - Upper Head	SB166 or 167 and SA182 Type F304
Instrumentation Tube Appurtenances - Lower Head	SB166 or 167 and SA182 Type F304, F304L or F316
Closure Studs, Nuts, Washers, Inserts and Adaptors	SA540 Class 3 Gr B23 or B24
Core Support Pads	SB166 with Carbon less than 0.10%
Monitor Tubes and Vent Pipe	SA312 or 376 Type 304 or 316 or SB166 or SB167 or SA182 Type 316
Vessel Supports, Seal Ledge and Heat Lifting Lugs	SA516 Gr 70 Quenched and Tempered or SA533 Gr. A, B or C, Class 1 or 2. (Vessel supports may be of weld metal buildup of strength equivalent to 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	SA533 Gr A, Class 2
Pressure Forgings	SA508 Class 2a

Table 5.2-2

CLASS 1 PRIMARY COMPONENTS MATERIAL
SPECIFICATIONS (Sheet 2 of 3)

Steam Generator Components (Continued)	
Nozzle Safe Ends	Stainless Steel Weld Metal Analysis A-8
Channel Heads	SA216 Grade WCC
Tubes	SB163 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	SA193 Gr B-7
Pressurizer Components	
Pressure Plates	SA533 Gr A, Class 2
Pressure Forgings	SA508 Class 2
Nozzle Safe Ends	SA182 316L Forging
Cladding and Buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Closure Bolting	SA193 Gr B-7
Reactor Coolant Pump	
Pressure Forging, Bolting Ring	SA182 F 304, F 316, F-347 or F 348 SA508 Class 2
Pressure Casting	SA351 Gr CF8, CF8A or CF8M
Tube and Pipe	SA213, SA376 or SA312 Seamless Type 304 or 316
Pressure Plates	SA240 Type 304 or 316
Bar Material	SA479 Type 304 or 316

Table 5.2-2

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS (Sheet 3 of 3)

Reactor Coolant Pump (Continued)	
Closure Bolting	SA193, SA320, SA540, SA453 - Gr 660
Flywheel, Motor (Non-pressure boundary component)	SA533 Gr B, Class 1
Reactor Coolant Piping	
Reactor Coolant Pipe	SA351 Gr CF8A Centrifugal Casting
Reactor Coolant Fittings	SA351 Gr CF8A and
Branch Nozzles	SA182 Gr 316N
Surge Line	SA376 Gr 304
Auxiliary Piping 1/2" through 12" 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
Control Rod Drive Mechanism	
Latch Housing	SA336 Gr F8
Rod Travel Housing	SA336 Gr F8
Cap	SA479 Type 304
Welding Materials	Stainless Steel Weld Metal Analysis A-8

Table 5.2-3

CLASS 1 AND 2 AUXILIARY COMPONENTS MATERIAL SPECIFICATIONS (Sheet 1 of 2)

Auxiliary Pressure Vessels, Tanks, Filters, etc.	
Shells and Heads	SA240 Type 304 or SA264 consisting of SA537 Cl1 with Stainless Steel Weld Metal Analysis A-8 Cladding
Flanges and Nozzles	SA182 Gr F304 and SA105 or SA350 Gr LF2, LF3 with Stainless Steel Weld Metal Analysis A-8 Cladding
Piping	SA312 and SA240 TP304 or TP316 Seamless
Pipe Fittings	SA403 WP304 Seamless
Closure Bolting and Nuts	SA193 Gr B7 and SA194 Gr 2H
Auxiliary Pumps	
Pump Casing and Heads	SA351 Gr CF8 or CF8M, SA182 Gr F304 or F316
Flanges and Nozzles	SA182 Gr F304 or F316, SA403 Gr WP316L Seamless
Valves	
Bodies	SA182 Type F316 or SA351 Gr CF8 or CF8M
Bonnets	SA182 Type F316 or SA351 Gr CF8 or CF8M
Discs	SA182 Type F316 or SA564 Gr 630
Pressure Retaining Bolting	SA453 Gr 660
Pressure Retaining Nuts	SA453 Gr 660 or SA194 Gr 6

Table 5.2-3

CLASS 1 AND 2 AUXILIARY COMPONENTS MATERIAL
SPECIFICATIONS (Sheet 2 of 2)

Auxiliary Heat Exchangers	
Heads	SA240 Type 304
Nozzle Necks	SA182 Gr F304
Tubes	SA213 TP304
Tube Sheets	SA182 Gr F304
Shells	SA240 and SA312 Type 304
Piping	SA312 TP304 or TP316 Seamless
Stuffing or Packing Box Cover	SA351 Gr CF8 or CF8M, SA240 TP304 or TP304L or TP316
Pipe Fittings	SA403 Gr WP316L Seamless
Closure Bolting and Nuts	SA193 Gr B6, B7 or B8M and SA194 Gr 2H or Gr 8M, SA193 Gr F6, B7 or B8M; SA453 Gr 660; and Nuts, SA194 Gr 2H, Gr 8M, and Gr 6

Table 5.2-4

REACTOR VESSELS INTERNALS FOR CORE SUPPORT

Forgings	SA182 Type F304
Plates	SA240 Type 304
Pipes	SA312 Type 304 Seamless
Tubes	SA213 Type 304
Bars	SA479 Type 304
Bolting	SA193 Gr. B8M (65 MYS/90MTS) Code Case 1618 Inconel 750 SA461 Gr. 688
Nuts	SA193 Gr. B-8
Locking Devices	SA479 Type 304

Table 5.2-5

REACTOR COOLANT WATER CHEMISTRY
SPECIFICATION (Sheet 1 of 2)

Electrical Conductivity	Determined by the concentration of boric acid and alkali present.	
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C. 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 - 50 cc (STP)/kg H_2O	
Suspended Solids[d]	1.0 ppm, maximum	
pH Control Agent (LiOH)[e]	0.2 - 3.5 ppm as Li	255
Boric Acid	Variable from 0 . 4000 ppm as B	
Silica[f]	1.0 ppm, maximum	255
Sulfate	0.15 ppm, maximum	
Aluminum[f]	0.05 ppm, maximum	
Calcium[f]	0.05 ppm, maximum	
Magnesium[f]	0.05 ppm, maximum	

a. Oxygen concentration must be controlled to less than 0.1 ppm in the reactor coolant at temperatures above 250°F by scavenging with hydrazine. During power operation with the specified hydrogen concentration maintained in the coolant, the residual oxygen concentration must not exceed 0.005 ppm. 37

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Table 5.2-5

REACTOR COOLANT WATER CHEMISTRY
SPECIFICATION (Sheet 2 of 2)

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

c. Hydrogen must be maintained in the reactor coolant for all plant operations with nuclear power above 1MWt. The normal operating range should be 30 ~ 40 cc/kg H_2O.

d. Solids concentration determined by filtration through filter having 0.45 micron pore size.

e. The startup lithium and boron are coordinated to maintain $pH_{Tave} \geq 6.9$. Lithium concentration can be maintained constant, at or below a target of 3.5 ppm, until a $pH_{Tave} = 7.1$ has been reached. Once a pH_{Tave} has been reached 7.1, lithium can be controlled so as to continue operation with $pH_{Tave} = 7.1$ 255 365

f. These limits are included in the table of reactor coolant specifications as recommended standards for monitoring coolant purity. Establishing coolant purity with 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.

Table 5.2-6

DESIGN COMPARISON WITH REGULATORY GUIDE 1.45 DATED MAY 1973 TITLED
REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEM
(Sheet 1 of 4)

Regulatory Guide 1.45 Position	KNU 5 & 6
C. Regulatory Position 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.	1. Complies. Flow to RCDT can be established, is monitored and is separated from unidentified leakage.
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 or better.	2. Complies. The instrumentation provided is such that over a period of time the collected flow rate can be determined with an accuracy of better than one gallon per minute.

Table 5.2-6

DESIGN COMPARISON WITH REGULATORY GUIDE 1.45 DATED MAY 1973 TITLED
REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEM
(Sheet 2 of 4)

Regulatory Guide 1.45 Position	KNU 5 & 6
3. At least three separate detection methods should be employed and two of these methods should be (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may be selected from the following: A. monitoring of the 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.	3. Complies. The methods provided are sump level and flow (level versus time) monitoring, airborne particulate radioactivity monitoring airborne gaseous radioactivity monitoring, containment humidity monitoring and containment purge exhaust monitor.
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.	4. Complies. Refer to subparagraph 5.2.5.2.1.2, subsection 9.3.3 and section 11.5.

Table 5.2-6

DESIGN COMPARISON WITH REGULATORY GUIDE 1.45 DATED MAY 1973 TITLED
REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEM
(Sheet 3 of 4)

Regulatory Guide 1.45 Position	KNU 5 & 6
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 gallon per minute in less than one hour.	5. Complies, as described in subparagraph 5.2.5.2.3.
6. The leakage detection system 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 SSE.	6. Complies. The airborne particulate radioactivity system is designed to remain functional when subjected to SSE. Refer to subparagraphs 11.5.3.2.1 and 11.5.3.2.2. The remaining leakage detection systems can reasonably to 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. 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.	7. Complies, as described in subparagraph 5.2.5.2.3 and paragraph 5.2.5.6.

Table 5.2-6

DESIGN COMPARISON WITH REGULATORY GUIDE 1.45 DATED MAY 1973 TITLED REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEM
(Sheet 4 of 4)

Regulatory Guide 1.45 Position	KNU 5 & 6
8. The leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation.	8. Complies. Refer to paragraph 5.2.5.5.
9. The technical specification 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.	9. Complies. Refer to ITS Chapter 1 3.4.13 and 3.4.15. 357

본 도면은 참고용 도면임
관련 도면목록 :Table 1.7-2(Sh. 4 of 12)
관리도면 : P&ID , 3-M-BC-F001

KOREA ELECTRIC POWER CORPORATION
KRN 3&4 FSAR

RESIDUAL HEAT REMOVAL SYSTEM
FIGURE 5.2-1

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

SAFETY INJECTION SYSTEM

Figure 5.2-2

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KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

SAFETY INJECTION SYSTEM
Figure 5.2-3

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KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

SAFETY INJECTION SYSTEM

Figure 5.2-4

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

REACTOR COOLANT SYSTEM

Figure 5.2-5

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 subsection 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:

Element	Base Metal (percent)	As-Deposited Weld Metal (percent)
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)

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. The vessel flange, head flange, and nozzles are manufactured as forgings. The cylindrical portion of the vessel is made up of several shells, each consisting of formed plates joined by full-penetration longitudinal and girth-weld seams. The reactor vessel parts are joined by welding, using the single or multiple wire submerged arc and the shielded metal arc processes.

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

C. The control rod drive mechanism head adaptor 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 beads are Inconel weld metal in order to prevent cracking.

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

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

G. The procedure qualification for cladding low alloy steel (SA508 Class 2) requires a special evaluation to assure freedom from underclad cracking.

H. Minimum preheat requirements have been established for pressure boundary welds using low alloy material. The preheat is maintained either until an intermediate post-weld heat treatment is completed or until the completion of welding. In the latter case, upon completion of welding, a low temperature (400F minimum) post-weld heat treatment is applied for four hours, followed by allowing the weldment to cool to ambient temperature. This practice is specified for all pressure boundary welds except for the installation of nozzles. For the primary-nozzle-to-shell welds, the preheat temperature is maintained until a high temperature (greater than 800F) post-weld heat treatment is applied in accordance with the requirements of the ASME Code Section III. This method is followed because higher restraint stresses may be present in the nozzle-to-shell weldments.

5.3.1.3 Special Methods for Nondestructive Examination

The nondestructive examination of the reactor vessel and its appurtenances is conducted in accordance with the 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 nondestructive examination program is given in table 5.3-1.

5.3.1.3.1 Ultrasonic Examinations

A. In addition to the design code straight beam ultrasonic test, 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.

B. In addition to ASME Section III nondestructive examination, all full-penetration ferritic pressure boundary welds in the reactor vessel are ultrasonically examined during fabrication. This test is 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

A. The partial penetration welds for the control rod drive mechanism head adaptors and the bottom instrumentation tubes are inspected by dye penetrant after the root pass in addition to code requirements. Core support block attachment welds are inspected by dye penetrant after first layer of weld metal and after each 1/2-inch of weld metal.

B. Surface Examinations

   1. Dye penetrant examinations of all clad surfaces and other vessel and head internal surfaces after the hydrostatic test.

   2. Base metal or weld surfaces which are exposed to mechanical or thermal straightening operations are dye-penetrant-examined immediately after this operation.

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 are 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 shall be examined by magnetic particle methods. The acceptable standards shall be in accordance with section lll of the ASME Code:

A. Surface Examinations

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

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

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

B. Weld Examination

Magnetic particle examination of the weld metal buildup for vessel supports, and welds attaching the closure head lifting lugs and 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 shall be examined after back-chipping or back-grinding operations.

5.3.1.4 Special Controls for Ferritic and Austenitic stainless Steels

Welding of ferritic steels and austenitic stainless steels is discussed in subsection 5.2.3. Paragraph 5.2.3.4 includes discussions which indicate the degree of conformance with Regulatory Guides 1.44, "Control of the Use of Sensitized Stainless Steel." Appendix 3A discusses the degree of conformance with Regulatory Guides 1.43, "Control of stainless Steel Weld Cladding of Low-Alloy Steel Components," 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steels," 1.71, "Welder Qualification for Areas of Limited Accessibility," and 1.99, "Radiation Embrittlement of Reactor Vessel Materials." 321

5.3.1.5 Fracture Toughness

Assurance of adequate fracture toughness of ferritic materials in the reactor coolant pressure boundary (ASME section lll Class 1 Components) is provided by compliance with the requirements for fracture toughness testing included in NB 2300 to Section lll of the ASME Boiler and Pressure Vessel Code and Appendix G of 10 CFR 50.

The vessel fracture toughness data is provided in table 5.3-2 for Unit 5 and table 5.3-3 for unit 6.

5.3.1.6 Material Surveillance

In the surveillance program, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile and 1/2 T (thickness) compact tension (CT) fracture mechanics test specimens. The program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach. The program will conform with ASTM-E-185 "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," and 10 CFR 50, Appendix H. 321

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

Dosimeters, including Ni, Cu, Fe, Co-Al, Cd shielded Co-Al, Cd shielded Np-237, and Cd shielded U-238, are placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low-melting-point alloys are included to monitor the maximum temperature of the specimens. The specimens are enclosed in a tight-fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity.

The complete capsule is helium leak-tested. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent will be made for surveillance material and as deposited weld metal.

Each of the six capsules contains the following specimens:

Material	Number of Charpys	Number of Tensiles	Number of CT's
Limiting Base Material[a]	15	3	4
Limiting Base Material[b]	15	3	4
Weld Metal[c]	15	3	4
Heat-Affected Zone	15	-	-

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

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

c. Weld metal to be selected per ASTM E185.

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 (579F melting point)

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

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall with the specimens being located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition temperature shift measurements are representative of the vessel at a later time in life. Data from CT fracture toughness specimens are expected to provided 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 subparagraph 5.3.1.6.1. They have indicated good agreement. The calculations of the integrated flux at the vessel wall are conservative. The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution will be considered in the evaluation of the surveillance specimen date. Verification and possible readjustment of the calculated wall exposure will be made by use of data on all capsules withdrawn. The 321 schedule for removal of the capsules for post-irradiation testing will conform with ASTM-E185 and Appendix H of 10 CFR 50. Adequate dosimetry program shall be established and confirmed periodically to monitor neutron fluence during the period of operation in accordance with Appendix H of 10 CFR 50 after all surveillance capsules have been withdrawn. The withdrawal schedule for capsules 318 and the examination schedule of dosimetry program are listed in the Table 5.3-7. 488

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

In order to effect a correlation between fast neutron (E>1.0MeV) 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:

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

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

The use of activation detectors such as those listed above dose 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 will 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 Fe^{54} (n,p) Mn^{54} reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron energy spectrum, is similar.

The analysis of the passive monitors and the subsequent derivation of the average neutron flux requires completion of two procedures. First, the disintegration rate of product isotope per unit mass of monitor must be determined. Second, in order to define a suitable spectrum-averaged reaction cross-section, the neutron energy spectrum at the monitor location must be calculated.

The Mn^{54} product of the Fe^{54} (n,p) Mn^{54} reaction has a half-life of 314 days and emits gamma rays of 0.84 Mev energy. The activity of each monitor is determined by means of a Lithium-drifted germanium, Ge (Li), gamma spectrometer. The overall standard deviation of this measured data is a function of the sample weighing, the uncertainty in counting, and the acceptable error in detector calibration. For typical samples removed from the reactor vessel surveillance capsules, the overall 2σ deviation in the measured data will be on the order of ± 10 percent.

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

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

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

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

where:

D	=	Induced Mn^{54} activity	(dps/gm_{Fe})
No	=	Avogadro's number	(atoms/gm-atom)
A	=	Atomic weight of iron	(gm/gm-atom)
f_i	=	Weight fraction of Fe^{54} in the detector	
$\sigma(E)$	=	Energy-dependent activation cross-section for the $Fe^{54}(n,p)Mn^{54}$ reaction	(barns)
$\phi(E)$	=	Energy-dependent neutron flux at the detector at full reactor power	$(n/cm^2\text{-}sec)$
λ	=	Decay constant of Mn^{54}	(1/sec)
F_j	=	Fraction of full reactor power during the jth time interval, t_j	
t_j	=	Length of the jth irradiation period	(sec)
τ_d	=	Decay time following the jth irradiation period	(sec)

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

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

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

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

$\bar{\phi}_{E}TH$ = Average neutron flux above energy, E_{TH}

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

$\phi_s(E)$ = Multigroup energy spectra at the detector location obtained from the DOT analysis

Thus,

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

or, solving for the threshold flux:

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

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

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

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

Because of the relatively long half-life of Mn^{54}, the fluence may be accurately calculated in this manner for irradiation periods up to about two 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/cm^2-sec.

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

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

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

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

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

The neutron flux at the surveillance capsule is given by:

$$\phi_C = \phi(E,R_C,\theta_C,Z_C)$$

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

$$\phi_{v-max} = \phi(E,R_v,\theta_{v-max},Z_{v-max})$$

The lead factor then becomes:

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

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

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure studs, nuts, and washers are designed and fabricated in accordance with the requirements of ASME III.

The closure studs are fabricated of SA 540, Class 3, Grade B24. The closure stud material meets the fracture toughness requirements of ASME 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 ASME III. Bolting materials fracture toughness data is provided in tables 5.3-4 for unit 5 and table 5.3-5 for unit 6.

Westinghouse refueling procedures require the studs, nuts and washers to be removed from the reactor closure and be placed in storage racks during preparation for refueling. The storage racks are then removed from the refueling cavity and stored at convenient locations on the containment operating deck prior to removal of the reactor closure head and refueling cavity flooding. Therefore, the reactor closure studs are never exposed to the borated refueling cavity water. 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 before removing the reactor closure, thus preventing leakage of the borated refueling water into the stud holes.

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Limit Curves

Startup and shutdown operating limitations will be based on the properties of the core region materials of the reactor pressure vessel, reference 2. Actual material property test data will be used. The methods outlined in Appendix G section III of the ASME Code will be employed for the shell regions in the analysis of protection against non-ductile failure. The initial operating curves are calculated assuming a period of reactor operation such that the beltline material will be limiting. The heatup and cooldown curves are given in the technical specifications. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil-ductility temperature which includes a reference nil-ductility temperature shift (ΔRT_{NDT}). Predicted ΔRT_{NDT} values are obtained using the procedure in Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials,": the effect of fluence, and copper and nickel contents on the shift of RT_{NDT} for the reactor vessel steels at 1/4T(thickness) and 3/4T locations (tips of the code reference flaw when the flaw is assumed at inside diameter and outside diameter locations respectively). 321

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 will be limiting in the analysis.

The operating curves, including pressure-temperature limita-
tions, are calculated in accordance with 10 CFR 50, Appendix G,
and ASME Code section XI, Appendix G, requirements. Changes
in fracture toughness of the core region plates or forgings,
weldments and associated heat-affected zones due to radiation
damage will be monitored by a surveillance program which con- 321
forms with ASTM E-185, "Standard Practice for Conducting
Surveillance Tests for Light-Water Cooled Nuclear Power Reactor
Vessels," and 10 CFR 50, Appendix H.
The evaluation of the radiation damage in this surveillance
program is based on pre-irradiation testing of Charpy V-notch
and tensile specimens and post-irradiation testing of Charpy
V-notch, tensile, and 1/2T compact tension specimens. The
post-irradiation testing will be carried out during the life- 321
time of the reactor vessel. Specimens are irradiated in cap-
sules located near the core midheight and removable from the
vessel at specified intervals.

Compliance with Regulatory Guide 1.99, "Radiation Embrittlement 321
of Reactor Vessel Materials," 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 paragraph 3.9.1.1. These transients are representative of the operating conditions that should prudently be considered to occur during plant operation. The transients selected form a conservative basis for evaluation of the reactor coolant system (RCS) to ensure the integrity of the RCS equipment.

Those transients listed as upset condition transients are listed in table 3.9-1. None of these transients will result in pressure-temperature changes which exceed the heatup and cooldown limitations as described in paragraph 5.3.2.1 and in the technical specifications.

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 the other outside the outer O-ring. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains head adaptors. These head adaptors are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contains acme threads for the assembly of control rod drive mechanisms or instrumentation adaptors. The seal arrangement at the upper end of these adaptors consists of a welded flexible canopy seal. Inlet and outlet nozzles are located symmetrically around the vessel. Outlet nozzles are arranged on the vessel to facilitate optimum layout of the reactor coolant system equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The 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-overlaid with 0.125-inch minimum of stainless steel and Inconel. The exterior of the reactor vessel is insulated with canned stainless steel reflective sheets. The insulation is a minimum of three inches thick and contoured to enclose the top, side, and bottom of the vessel. Provisions are made for the removability of the insulation covering the closure and bottom heads to allow access for inservice inspection; access to the vessel side insulation is limited by the surrounding concrete.

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

Principal design parameters of the reactor vessel are given in table 5.3-6. 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. In the unlikely event that an annealing operation is required to restore the properties of the vessel material opposite the reactor core because of neutron irradiation damage, a metal temperature of approximately 850F would be applied for a period of 168 hours.

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

Cyclic loads are introduced by normal power changes, reactor trip, startup, and shutdown operations. These design 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 ASME III. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heatup and cooldown rates imposed by plant operating limits are 50F per hour for normal operations and 100F per hour under abnormal or emergency conditions. The rate of 100F per hour is reflected in the vessel design specifications as a normal condition for conservatism.

5.3.3.2 Materials of Construction

The materials used in the fabrication of the reactor vessel are discussed in subsection 5.2.3.

5.3.3.3 Fabrication Methods

The reactor vessels for Korea Nuclear Units 5 & 6 will be manufactured by Combustion Engineering Corporation.

The fabrication methods used in the construction of the reactor vessel are described in paragraph 5.3.1.2.

5.3.3.4 Inspection Requirements

The nondestructive examinations performed on the reactor vessel are described in paragraph 5.3.1.3.

5.3.3.5 Shipment and Installation

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

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 housing. All head openings are sealed to prevent the entrance of moisture and an adequate quantity of desiccant bags is placed inside the head.

These are placed in a wire mesh basket attached to the head cover. All carbon steel surfaces are painted with heat-resistant paint before shipment. A lifting frame is provided for handling the vessel head.

5.3.3.6 Operating Conditions

Operating limitations for the reactor vessel are presented in subsection 5.3.2, as well as in the technical specifications.

Actuation of the emergency core cooling system (ECCS) following a loss-of-coolant accident produces relatively high thermal stresses in the regions of the reactor vessel that come into contact with ECCS water. Primary regions of interest include the reactor vessel beltline region and the reactor vessel primary coolant nozzle.

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

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

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses perpendicular to the major plane of the crack) the stress intensity factor is designated as K_1 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 LEFM analysis, methods in ASME XI Appendix A and ASME III Appendix G are used to perform the fracture evaluation of postulated flaws to establish that the vessel integrity is maintained. This LEFM analysis is considered accurate in the elastic range and conservative in the elastic-plastic range. 321

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Therefore, for faulted condition 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.

An example of a faulted condition evaluation carried out according to the procedure discussed above is given in reference 3. This report discusses the evaluation procedure in detail as applied to a severe faulted condition (a postulated loss of coolant accident).

5.3.3.7 Inservice Surveillance

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

The closure head is examined visually during each refueling. Optical devices permit a selective inspection of the cladding, control rod drive mechanism housings 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 inspections, dye penetrant or magnetic particle, and ultrasonsic testing. The closure studs can be inspected periodically using visual, magnetic particle, and/or 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 irradiated reactor vessel, are available for nondestructive examination:

A. Vessel shell - from the inside surface.

B. Primary coolant nozzles - from the inside surface.

C. Closure head - from the inside and outside surfaces. Bottom head - from the outside surface.

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

The design considerations that have been incorporated into the system design to permit the above inspections 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. All reactor vessel studs, nuts, and washers can be removed to dry storage during refueling.

D. Removable plugs are provided in the primary shield. The insulation covering the nozzle-to-pipe welds may be removed.

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

A. Shop ultrasonic examinations are performed on all internally clad surfaces to acceptance and repair standards that assure an adequate cladding bond that allows later ultrasonic testing of the base metal from the 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 are 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 in addition to ASME Code Section III requirements.

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

5.3.4 REFERENCES

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

2. Hazelton, W. S., et al, "Basis for Heatup and Cooldown Limit Curves," WCAP-7924-A, April, 1975.

3. Buchalet, 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.

Table 5.3-1

REACTOR VESSEL NONDESTRUCTIVE EXAMINATION PROGRAM

	RT(a)	UT(a)	PT(a)	MT(a)
<u>Forgings</u>				
Flanges		yes		yes
Studs, Nuts		yes		yes
CRD Head Adaptor Flange		yes	yes	
CRD Head Adaptor Tube		yes	yes	
Instrumentation Tube		yes	yes	
Main Nozzles		yes		yes
Nozzle Safe-Ends		yes	yes	
<u>Plates</u>		yes		yes
<u>Weldments</u>				
Main Seam	yes	yes		yes
CRD Head Adaptor-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	
Nozzle to Safe-Ends After Hydrotest		yes	yes	
CRD Head Adaptor Flange to CRD Head Adaptor Tube	yes		yes	
All Full-Penetration Ferritic Pressure Boundary Welds Accessible After Hydrotest		yes		yes
Seal Ledge				yes
Head Lift Lugs				yes
Core Pad Welds			yes	

a. RT - Radiographic
   UT - Ultrasonic
   PT - Dye Penetrant
   MT - Magnetic Particle

Table 5.3-2

KORI Unit 5 Reactor Vessel Material Properties

Component	Code No.	Material Spec. No.	Cu (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Upper Shelf Energy NMWD (ft lb)	Upper Shelf Energy MWD (ft lb)
Closure Dome	R6016-1	A533B CL.1	.07	.014	-20	-20	152	-
Head Flange	R6002-1	A508 CL.3	-	.010	-50	-50	107	-
Vessel Flange	R6001-1	A508 CL.3	.02	.007	-50	-50	117	-
Inlet Nozzle	R6003-1	A508 CL.3	.04	.008	-30	-30	135	-
Inlet Nozzle	R6003-2	A508 CL.3	.03	.008	-30	-30	130	-
Inlet Nozzle	R6003-3	A508 CL.3	.04	.009	-30	-30	141	-
Outlet Nozzle	R6004-1	A508 CL.3	.07	.008	-70	-70	138	-
Outlet Nozzle	R6004-2	A508 CL.3	.09	.008	-70	-70	110	-
Outlet Nozzle	R6004-3	A508 CL.3	.08	.007	-100	-90	104	-
Upper Shell	R6006-1	A533B, CL.1	.07	.006	-20	-20	132	-
Upper Shell	R6006-2	A533B, CL.1	.03	.005	0	0	142	-
Upper Shell	R6006-3	A553B, CL.1	.07	.006	-20	-20	132	-
Inter. Shell	R6007-1	A533B, CL.1	.04	.005	-30	-30	155	229
Inter. Shell	R6007-2	A533B, CL.1	.04	.005	-30	-30	152	171
Lower Shell	R6008-1	A533B, CL.1	.04	.007	-40	-20	132	145
Lower Shell	R6008-2	A533B, CL.1	.06	.008	-50	10	128	159
Bottom Head Dome	R6014-1	A533A, CL.1	.12	.014	-40	10	69	-
Bottom Head Dome	R6015-1	A533A, CL.1	.13	.013	-30	20	76	-
Core Region Welds	E3.07	A533A, CL.1	.05	.008	-70	-70	97	-

NMWD - Normal-to-Major Working Direction
MWD - Major Working Direction

Table 5.3-3

KORI Unit 6 Reactor Vessel Material Properties

Component	Code No.	Material Spec. No.	Cu (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Upper Shelf Energy NMWD (ft lb)	Upper Shelf Energy MWD (ft lb)
Closure Head Dome	R6307-1	A533B, CL.1	.10	.009	-30	-30	140	-
Closure Head Flange	R6306-1	A508, CL.3	-	.008	-40	-40	147	-
Vessel Flange	R6303-1	A508, CL.3	-	.008	-60	-60	135	-
Inlet Nozzle	R6304-1	A508, CL.3	.08	.007	-20	0	117	-
Inlet Nozzle	R6304-2	A508, CL.3	.08	.007	-20	-20	132	-
Inlet Nozzle	R6304-3	A508, CL.3	.09	.008	-20	-10	117	-
Outlet Nozzle	R6305-1	A508, CL.3	.10	.009	-40	-40	121	-
Outlet Nozzle	R6305-2	A508, CL.3	.10	.008	-60	-60	109	-
Outlet Nozzle	R6305-3	A508, CL.3	.09	.009	-40	-40	127	-
Upper Shell	R6301-1	A533B, CL.1	.05	.011	-20	55	102	-
Upper Shell	R6301-2	A533B, CL.1	.06	.011	0	10	114	-
Inter. Shell	R6302-1	A533B, CL.1	.04	.006	-20	10	137	160
Inter. Shell	R6302-2	A533B, CL.1	.03	.005	-40	-10	145	174
Lower Shell	R6309-1	A533B, CL.1	.04	.007	10	10	124	157
Lower Shell	R6009-2	A533B, CL.1	.05	.007	10	20	113	146
Bottom Head Dome	R6311-1	A533A, CL.1	.15	.013	-10	-10	66	-
Bottom Head Dome	R6310-1	A533A, CL.1	.16	.009	-30	-10	64	-
Core Region Welds	E5.07	A533A, CL.1	.03	.009	-60	-60	94	-

NMWD - Normal-to-Major Working Direction
MWD - Major Working Direction

Table 5.3-4

KORI Unit 5 Reactor Vessel Closure Head Bolting Material Properties
(Sheet 1 of 2)

CLOSURE HEAD STUDS

Heat No	Material Spec. No.	Bar No.	0.2% Yield Strength (Ksi)	Ultimate Tensile Strength (Ksi)	Elongation (%)	Reduction In Area (%)	Energy At 10F (Ft Lb)	Lateral Expansion (Mils)
88826	SA540,B24	261	139.5	155.0	17.5	56.0	60,60,62	38,38,40
88826	SA540,B24	261-1	142.5	156.0	17.0	54.9	60,62,64	37,34,41
88826	SA540,B24	262	140.8	155.5	17.5	54.9	64,64,66	40,41,43
88826	SA540,B24	262-1	141.0	157.0	17.0	54.4	57,57,54	36,35,33
88826	SA540,B24	268	143.8	157.5	17.0	57.3	64,61,60	46,40,40
88826	SA540,B24	268-1	145.0	160.0	16.5	55.7	55,54,56	34,36,36
88452	SA540,B24	233	143.5	159.5	15.5	53.3	51,51,51	30,29,29
88452	SA540,B24	233-1	142.5	158.0	16.0	55.0	53,54,54	32,32,34
88452	SA540,B24	237	145.0	159.0	15.5	54.7	54,54,53	29,32,28
88452	SA540,B24	237-1	145.0	160.0	15.5	53.8	47,46,47	27,26,28
88452	SA540,B24	226	140.0	156.0	16.5	54.7	55,54,55	34,33,33
88452	SA540,B24	226-1	151.2	166.0	16.0	54.4	50,49,50	30,28,30
88452	SA540,B24	232	144.8	159.0	16.0	54.1	54,55,56	32,33,32
88452	SA540,B24	232-1	141.2	157.0	17.0	55.2	53,54,53	32,31,33
87069	SA540,B24	922	137.5	153.0	18.5	58.8	57,60,60	36,40,39
87069	SA540,B24	922-1	142.8	157.0	18.0	57.5	57,58,59	35,35,39
87069	SA540,B24	925	142.5	156.0	17.5	56.5	54,55,56	36,36,36
87069	SA540,B24	925-1	148.0	162.0	16.5	55.2	56,53,55	37,33,34
87069	SA540,B24	929	147.8	162.0	17.0	55.5	59,57,59	39,36,38
87069	SA540,B24	929-1	142.5	156.5	17.5	56.5	54,56,55	35,35,34
87069	SA540,B24	936	142.8	156.5	16.5	57.8	58,58,57	38,38,36
87069	SA540,B24	936-1	146.2	160.0	16.0	56.5	57,57,54	37,36,35
86532	SA540,B24	940	148.8	162.5	16.0	53.8	50,52,52	30,32,32
86532	SA540,B24	940-1	143.5	162.0	16.0	54.9	54,53,52	34,33,32
86532	SA540,B24	944	145.0	160.0	16.5	57.3	52,51,53	33,31,33
86532	SA540,B24	944-1	147.5	161.0	16.0	54.7	53,52,53	35,32,32

Table 5.3-4

KORI Unit 5 Reactor Vessel Closure Head Bolting Material Properties
(Sheet 2 of 2)

CLOSURE HEAD NUTS AND WASHERS

Heat No	Material Spec. No.	Bar No.	0.2% Yield Strength (Ksi)	Ultimate Tensile Strength (Ksi)	Elongation (%)	Reduction In Area (%)	Energy At 10F (Ft Lb)	Lateral Expansion (Mils)
86633	SA540,B24	275	143.0	156.5	17.0	57.3	57,58,57	36,35,36
86633	SA540,B24	275-1	141.2	154.5	17.5	56.5	60,62,59	38,39,37
86633	SA540,B24	278	136.5	150.0	17.5	58.6	64,65,63	41,42,41
86633	SA540,B24	278-1	140.5	155.0	17.0	56.8	59,57,57	39,36,36
86633	SA540,B24	280	140.5	154.0	17.5	58.3	59,56,60	37,38,40
86633	SA540,B24	280-1	140.0	154.0	17.0	56.0	58,58,60	36,36,39
87025	SA540,B24	286	144.8	159.0	16.0	54.1	55,57,56	37,39,36
87025	SA540,B24	286-1	144.5	158.5	16.5	54.7	56,55,55	36,33,34
84080	SA540,B24	125	149.8	162.0	16.5	54.9	51,49,48	30,28,26
84080	SA540,B24	125-1	151.0	163.0	16.5	55.5	51,53,53	29,31,31

Table 5.3-5

KORI Unit 6 Reactor Vessel Closure Head Bolting Material Properties
(Sheet 1 of 2)

CLOSURE HEAD STUDS

Heat No	Material Spec. No.	Bar No.	0.2% Yield Strength (Ksi)	Ultimate Tensile Strength (Ksi)	Elongation (%)	Reduction In Area (%)	Energy At 10°F (Ft Lb)	Lateral Expansion (Mils)
84906	SA540,B24	532	155.0	167.5	17.0	55.7	46,45,46	29,26,26
84906	SA540,B24	532-1	142.2	156.0	17.5	57.8	50,54,53	32,35,33
84906	SA540,B24	537	145.0	160.0	17.0	54.7	49,48,50	29,30,28
84906	SA540,B24	537-1	149.5	164.0	17.0	53.8	47,49,48	30,30,30
84906	SA540,B24	538	142.0	158.0	17.0	54.1	47,46,47	30,29,33
84906	SA540,B24	538-1	137.5	153.0	18.0	56.5	55,56,53	35,35,33
84906	SA540,B24	543	137.2	152.0	17.0	53.8	48,49,48	32,30,30
84906	SA540,B24	543-1	137.0	152.5	17.0	55.2	50,48,49	31,30,32
86532	SA540,B24	837	142.0	157.0	16.0	53.3	57,57,56	36,35,34
86532	SA540,B24	837-1	143.0	158.0	16.5	54.1	56,57,55	33,36,36
86532	SA540,B24	841	144.0	158.5	16.5	54.7	55,57,57	35,38,37
86532	SA540,B24	841-1	147.5	161.0	16.5	55.5	56,54,56	34,32,37
86532	SA540,B24	842	143.0	158.0	17.0	55.5	56,55,56	34,34,35
86532	SA540,B24	842-1	146.5	162.0	17.0	54.7	53,51,52	32,30,33
86532	SA540,B24	846	138.8	155.0	17.5	57.5	59,60,59	39,40,39
86532	SA540,B24	846-1	140.2	155.5	17.0	56.5	59,62,64	38,41,43
88826	SA540,B24	261	139.5	155.0	17.5	56.0	60,60,62	38,38,40
88826	SA540,B24	261-1	142.5	156.0	17.0	54.9	60,62,64	37,34,41
88826	SA540,B24	262	140.8	155.5	17.5	54.9	64,64,66	40,41,43
88826	SA540,B24	262-1	141.0	157.0	17.0	54.4	57,57,54	36,35,33
88826	SA540,B24	268	143.8	157.5	17.0	57.3	64,61,60	46,40,40
88826	SA540,B24	268-1	145.0	160.0	16.5	55.7	55,54,56	34,36,36
88452	SA540,B24	233	143.5	159.5	15.5	53.3	51,51,51	30,29,29
88452	SA540,B24	233-1	142.5	158.0	16.0	55.0	53,54,54	32,32,34
88452	SA540,B24	237	145.0	159.0	15.5	54.7	54,54,53	29,32,28
88452	SA540,B24	237-1	145.0	160.0	15.5	53.8	47,46,47	27,26,28
88452	SA540,B24	226	140.0	156.0	16.5	54.7	55,54,55	34,33,33
88452	SA540,B24	226-1	151.2	166.0	16.0	54.4	50,49,50	30,28,30
88542	SA540,B24	232	144.8	159.0	16.0	54.1	54,55,56	32,33,32
88542	SA540,B24	232-1	141.2	157.0	17.0	55.2	53,54,53	32,31,33
87069	SA540,B24	922	137.5	153.0	18.5	58.8	57,60,60	36,40,39
87069	SA540,B24	922-1	142.8	157.0	18.0	57.5	57,58,59	35,35,39
87069	SA540,B24	925	142.5	156.0	17.5	56.5	54,55,56	36,36,36
87069	SA540,B24	925-1	148.0	162.0	16.5	55.2	56,53,55	37,33,34
87069	SA540,B24	929	147.8	162.0	17.0	55.5	59,57,59	39,36,38
87069	SA540,B24	929-1	142.5	156.5	17.5	56.5	54,56,55	35,35,34
87069	SA540,B24	936	142.8	156.5	16.5	57.8	58,58,57	38,38,36
87069	SA540,B24	936-1	146.2	160.0	16.0	56.5	57,57,54	37,36,35
86532	SA540,B24	940	148.8	162.5	16.0	53.8	50,52,52	30,32,32

Table 5.3-5

KORI Unit 6 Reactor Vessel Closure Head Bolting Material Properties
(Sheet 2 of 2)

Heat No	Material Spec. No.	Bar No.	0.2% Yield Strength (Ksi)	Ultimate Tensile Strength (Ksi)	Elongation (%)	Reduction In Area (%)	Energy At 10°F (Ft Lb)	Lateral Expansion (Mils)
CLOSURE HEAD STUDS								
86532	SA540,B24	940-1	148.5	162.0	16.0	54.9	54,53,52	34,33,32
86532	SA540,B24	944	145.0	160.0	16.5	57.3	52,51,53	33,31,33
86532	SA540,B24	944-1	147.5	161.0	16.0	54.7	53,52,53	35,32,32
CLOSURE HEAD NUTS & WASHERS								
86633	SA540,B24	275	143.0	156.5	17.0	57.3	57,58,57	36,35,36
86633	SA540,B24	275-1	141.2	154.5	17.5	56.5	60,62,59	38,39,37
86633	SA540,B24	278	136.5	150.0	17.5	58.6	64,65,63	41,42,41
86633	SA540,B24	278-1	140.5	155.0	17.0	56.8	59,57,57	39,36,36
86633	SA540,B24	280	140.5	154.0	17.5	58.3	59,56,60	37,38,40
86633	SA540,B24	280-1	140.0	154.0	17.0	56.0	58,58,60	36,36,39

Table 5.3-6

REACTOR VESSEL DESIGN PARAMETERS

Design/Operating Pressure (psig)	2485/2235
Design Temperature, (F)	650
Overall Height of Vessel and Closure Head, ft-in (Bottom Head Outside Diameter to top of Control Rod Mechanism Adaptor (ft.-in.)	42-7 3/16
Thickness of Insulation, minimum, (in.)	3
Number of Reactor Closure Head Studs	58
Diameter of Reactor Closure Head/Studs, (in.) (minimum shank)	6
Inside Diameter of Flange, (in.)	149-9/16
Outside Diameter of Flange, (in.)	184
Inside Diameter at Shell, (in.)	157
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
Vessel Belt-Line Thickness, minimum, (in.)	7-7/8
Closure Head Thickness, (in.)	6-3/16

Table 5.3-7

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
(UNIT 3 CAPSULE WITHDRAWAL SCHEDULE)
(Sheet 1 of 2)

Capsule Identification	Orientation	Lead Factor(1)	Approximate Removal Time
U	343°	3.22	0.89 EFPY
V	107°	4.00	3.41 EFPY
X	287°	3.70	6.83 EFPY
W	110°	3.32	11.58 EFPY
Y	290°	3.37	15.37 EFPY
Z	340°	3.37	15.37(2) EFPY
HU3-A(3)	107°		31~34 EFPY
HU3-B(3)	110°		34~37 EFPY
HU4-A(4)	287°		31~34 EFPY
HU4-B(4)	290°		34~37 EFPY

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Note :
(1) Calculated value based on the End of 15th Fuel Cycle.
(2) Removed at the End of 15th Fuel Cycle.
(3) Capsule of Hanul Unit 3 was inserted at the End of 23rd Fuel Cycle.
(4) Capsule of Hanul Unit 4 was inserted at the End of 23rd Fuel Cycle. 530

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
(UNIT 3 DOSIMETRY WITHDRAWAL SCHEDULE)

Number	Dosimetry Identification	Installation Fuel Cycles(2)	Withdrawal Fuel Cycle(2)	Remarks
1	IS-1	15	16	1st Installation (1)
2	IS-2	16	17	Baseline
3	IS-3	17	18	Baseline
4	IS-4	18	19	Before Uprating
5	IS-5	19	20	After Uprating
6	IS-6	20	23	
7	IS-7	23	26	
8	IS-8	26	29	
9	IS-9	29	32	EOL

Note :
(1) The 1st Ex-Vessel Neutron Dosimetry System was installed on January 21, 2004.
(2) The Schedule may be altered whenever there is a significant change in neutron exposure.

Amendment 530
2015.08.21

Table 5.3-7

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
(UNIT 4 CAPSULE WITHDRAWAL SCHEDULE)
(Sheet 2 of 2)

Capsule Identification	Orientation	Lead Factor(1)	Approximate Removal Time
U	343°	3.31	0.89 EFPY
V	107°	3.63	3.29 EFPY
X	287°	3.51	7.05 EFPY
W	110°	3.07	13.18 EFPY
Y	290°	3.06	14.43 EFPY
Z	340°	3.14	15.76(2) EFPY
HU5-A(3)	107°		31~34 EFPY
HU5-B(3)	110°		34~37 EFPY
HU6-A(4)	287°		31~34 EFPY
HU6-B(4)	290°		34~37 EFPY

Note :
(1) Calculated value based on the End of 14th Fuel Cycle.
(2) Removed at the End of 15th Fuel Cycle.
(3) Capsule of Hanul Unit 5 was inserted at the End of 23rd Fuel Cycle.
(4) Capsule of Hanul Unit 6 was inserted at the End of 23rd Fuel Cycle.

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REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
(UNIT 4 DOSIMETRY WITHDRAWAL SCHEDULE)

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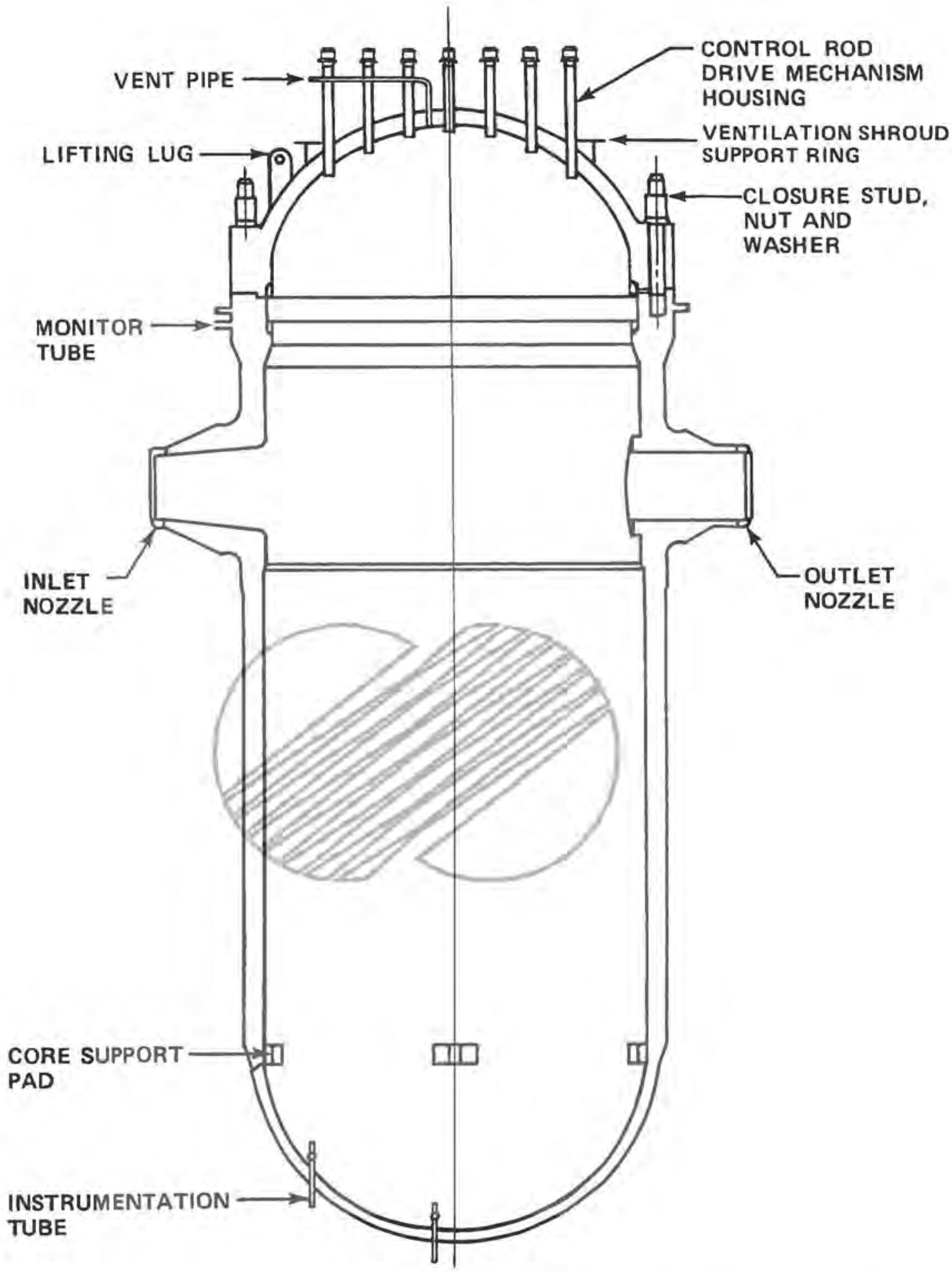
Number	Dosimetry Identification	Installation Fuel Cycle(2)	Withdrawal Fuel Cycle(2)	Remarks
1	IS-1	15	16	1st Installation (1)
2	IS-2	16	17	Baseline
3	IS-3	17	18	Baseline Before Uprating
4	IS-4	18	19	After Uprating
5	IS-5	19	22	
6	IS-6	22	25	
7	IS-7	25	28	
8	IS-8	28	31	
9	IS-9	31	34	EOL

Note :
(1) The 1st Ex-Vessel Neutron Dosimetry System was installed on September 30, 2004.
(2) The Schedule may be altered whenever there is a significant change in neutron exposure.



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FSAR

REACTOR VESSEL

Figure 5.3-1

5.4

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

5.4.1.1 Design Bases

The reactor coolant pump ensures an adequate core cooling flow rate for sufficient heat transfer to maintain a departure from nucleate boiling ratio (DNBR) greater than 1.3 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.

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

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.

5.4.1.2 Pump Description

5.4.1.2.1 Design Description

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

The pump consists of three major areas: the hydraulics, the seals, and the motor.

A. The hydraulic section consists of the casing, thermal barrier flange, impeller/diffuser, and diffuser adapter.

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

C. The motor is a totally enclosed 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

The 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 on 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 reactor coolant system and the remainder flowing 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.

The reactor coolant pump motor bearings are of conventional design. The radial bearings are of 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 internal lower bearing oil cooler.

The motor is a totally enclosed water/air cooled, Class F thermalastic epoxy insulated, squirrel cage induction motor with a 7000 Hp rating. 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 anti-reverse rotation device.

The internal parts of the motor are cooled by air. Integral vanes on each end of the rotor draw air in through cooling slots in the motor frame. This air passes through the motor with particular emphasis on the stator end turns. It is then routed to the external water/air heat exchangers which are supplied with component cooling water. Each motor has two such coolers, mounted diametrically opposed to each other. The air is recirculated from motor to coolers and back to motor. Coolers are sized to maintain optimum motor operating temperature. No heat is rejected from the motor to the containment.

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 motion shaft probes mounted on top of the pump seal housing; the probes are located ninety degrees apart in the same horizontal plane and are 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. Proximeters and converters linearize the probe output which is displayed on monitor meters in the control room. The monitor meters automatically indicate the highest output from the relative probes and seismoprobes; manual selection allows monitoring of individual probes. Indicator lights display caution and danger limits of vibration.

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

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

5.4.1.3 Design Evaluation

5.4.1.3.1 Pump Performance

The reactor coolant pumps are sized to deliver flow at rates which equal or exceed the required flow rates. Initial reactor coolant system 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 ensures that pump operation is within the assumptions used for loss of coolant flow analyses, which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation.

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

The support of the stationary member of the 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 relatively small leakage rates would be maintained for a period of time which is sufficient to secure the pump; even if the number 1 seal fails entirely during normal operation, the number 2 seal would maintain these small leakage rates if the proper action is taken by the operator. The plant operator is warned of number 1 seal damage by the increase in number 1 seal leakoff rate. Following warning of excessive seal leakage conditions, the plant operator should close the number 1 seal leakoff line and secure the pump, as specified in the instruction manual. Gross leakage from the pump does not occur if the proper operator action is taken subsequent to warning of excessive seal leakage conditions.

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

5.4.1.3.2 Coastdown Capability

It is important to reactor protection that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow in a station blackout condition, each reactor coolant pump is provided with a flywheel. Thus, the rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow. The coastdown flow transients are provided in the figures in section 15.3. The pump/motor system is designed for the safe shutdown earthquake at the site. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a blackout coincident with the safe shutdown earthquake. Core flow transients and figures are provided in section 15.3.

5.4.1.3.3 Bearing Integrity

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

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

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

5.4.1.3.4 Locked Rotor

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft fails in torsion just below the coupling to the motor, disengaging the flywheel and motor from the shaft. This constitutes a loss of coolant flow in the loop. Following such a postulated seizure, the motor continues to run without any overspeed, and the flywheel maintains its integrity, as it is still supported on a shaft with two bearings. Flow transients are provided in the figures in section 15.3 for the assumed locked rotor.

There are no other 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 anti-rotation pin in the seal ring. 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 indicators respectively. Following these signals, pump vibration levels are checked. Excessive vibration indicates mechanical trouble and the pump is shut down for investigation.

5.4.1.3.5 Critical Speed

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

5.4.1.3.6 Missile Generation

Precautionary measures taken to preclude missile formation from primary coolant pump components, assure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing. Further discussion and analysis of missile generation are contained in reference 1.

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 236-foot head (approximately 101 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 from this source.

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.

An electrical fault requiring immediate trip of the generator (with resulting turbine trip) could result in an overspeed condition. However, the turbine control system and the turbine intercept valves limit the overspeed to less than 120 percent. As additional backup, the turbine protection system has a mechanical overspeed protection trip, usually set at about 110 percent (of turbine speed). In case a generator trip deenergizes the pump buses, the reactor coolant pump motors will be transferred to offsite power within 6 to 10 cycles. Further discussion of pump overspeed considerations is contained in reference 1.

5.4.1.3.9 Anti-Reverse Rotation Device

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

At an approximate forward speed of 70 rpm, the pawls drop and bounce across the ratchet plate; as the motor continues to slow, the pawls drag across the ratchet plate. After the motor has 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 anti-reverse rotation device has shown high reliability of operation.

5.4.1.3.10 Shaft Seal Leakage

Leakage along the reactor coolant pump shaft is controlled by three shaft seals arranged in series such that reactor coolant leakage to the containment is essentially zero. Seal injection flow is directed to each reactor coolant pump via a seal water injection filter. It enters the pumps through a connection on 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 reactor coolant system; the remainder flows up the shaft through the seals. This flow provides a back pressure 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 the number 3 seal leakoff. This arrangement assures essentially zero leakage of reactor coolant or trapped gases from the pump to the containment.

5.4.1.3.11 Seal Discharge Piping

The number 1 seal drops the system pressure to that of the volume control tank. Water from each pump number 1 seal is piped to a common manifold, and through the seal water return filter and through the seal water heat exchanger where the temperature is reduced to that of the volume control tank. The number 2 and number 3 leakoff lines dump numbers 2 and 3 seal leakage to the reactor coolant drain tank and the containment sump, respectively.

5.4.1.4 <u>Tests and Inspections</u>

The reactor coolant pumps can be inspected in accordance with ASME Section XI, Code for Inservice Inspection of Nuclear Reactor Coolant Systems.

The pump casing may be cast in one or two pieces, thus having at the most one circumferential weld 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 surface of the pump casing.

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

5.4.1.5 Pump Flywheels

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

5.4.1.5.1 Design Basis

The calculated stresses at operating speed are based on stresses due to centrifugal forces. The stress resulting from the interference fit of the flywheel on the shaft is less than 2000 psi at zero speed, but this stress becomes zero at approximately 600 rpm because of radial expansion of the hub. The primary coolant pumps run at approximately 1190 rpm and may operate briefly at overspeeds up to 109 percent (1295 rpm) during loss of outside load. For conservatism, however, 125 percent of operating speed was selected as the design speed for the primary coolant pumps. The flywheels are given a preoperational test of 125 percent of the maximum synchronous speed of the motor.

5.4.1.5.2 Fabrication and Inspection

The flywheel consists of two thick plates bolted together. The flywheel material is produced by a process 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. 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 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 finished machined bores, keyways, and drilled holes are subjected to magnetic particle or liquid penetrant examinations in accordance with the requirements of Section III of the ASME Code. The finished flywheels, as well as the flywheel material (rolled plate), are subjected to 100 percent volumetric ultrasonic inspection using procedures and acceptance standards specified in Section III of the ASME Code.

The reactor coolant pump motors are designed such that, by removing the 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. For a description of the inservice inspection program for the flywheels, refer to subsection 5.2.4.

5.4.1.5.3 Material Acceptance Criteria

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

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

B. A minimum of three Charpy V-notch impact specimens from each plate shall be tested at ambient (70F) 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 70F and therefore, an RT_{NDT} of 10F can be assumed. An evaluation of flywheel overspeed has been performed which concludes that flywheel integrity will be maintained.

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 suppliers' certification data. The degree of compliance with Regulatory Guide 1.14 is further discussed in appendix 3A.

5.4.2 STEAM GENERATORS

5.4.2.1 Steam Generator Materials

5.4.2.1.1 Selection and Fabrication of Materials

All pressure boundary materials used in the steam generator are selected and fabricated in accordance with the requirements of Section III of the ASME Code. A general discussion of materials specifications is given in subsection 5.2.3, 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 subsection 5.2.3, particularly in paragraphs 5.2.3.3 and 5.2.3.4.

Testing has justified the selection of corrosion-resistant Inconel 600, a nickel-chromium-iron alloy (ASME SB-163), for the steam generator tubes. The channel head divider plate is Inconel (ASME SB-168). The interior surfaces of the reactor coolant channel head, nozzles, 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 tubes are then seal welded to the tube sheet cladding. These fusion welds are performed in compliance with Sections III and IX of the ASME Code and are dye penetrant inspected and leakproof tested before each tube is expanded the full depth of the tube sheet bore.

Code cases used in materials selection are discussed in subsection 5.2.1. The extent of conformance with Regulatory Guides 1.84 and 1.85 is also discussed in appendix 3A.

During manufacture, cleaning is performed on the primary and secondary sides of the steam generator in accordance with written procedures which follow the guidance of Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants, and 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 paragraph 5.2.3.4.

The fracture toughness of the materials is discussed in paragraph 5.2.3.3. Adequate fracture toughness of ferritic materials in the RCPB is provided by compliance with Appendix G of 10 CFR 50 and with Article NB-2300 of Section III of the ASME Code. As discussed in paragraph 5.4.2.3, consideration of fracture toughness is only necessary for materials in Class 1 components.

5.4.2.1.2 Steam Generator Design Effects on Materials

Several features have been introduced into the Model F steam generator to minimize the deposition of contaminants from the secondary side flow. Such deposits could otherwise produce a local environment in which adverse conditions could develop and result in material attack. The support plates are made of corrosion-resistant stainless steel 405 alloy and incorporate a four-lobe shaped tube hole design that provides greater flow area adjacent to the tube outer surface and eliminates the need for interstitial flow holes. The resulting increase in flow provides higher sweeping velocities at the tube/tube support plate intersections. Figure 5.4-3 is an illustration of the quatrefoil broached holes. This modification in the support plate design is a major factor contributing to the increased circulation ratio. The increased circulation results in better flow in the interior of the bundle, as well as increased horizontal velocity across the tube sheet, which reduces the tendency for sludge deposition there. The effect of the increased circulation on the vibrational stability of the tube bundle has been analyzed with consideration given to flow-induced excitation frequencies. The unsupported span length of tubing in the U-bend region and the corresponding optimum number of antivibration bars has been determined. The antivibration bars are fabricated from square Inconel barstock which is then chromium plated to improve frictional characteristics. Also, due to the increased circulation ratio, the moisture separating equipment has been modified to maintain an adequate margin with respect to the allowed moisture carryover. To provide added strength as well as resistance to vibration, the quatrefoil tube support plate thickness has been increased. In addition, 14 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, fluidelastic excitation, and turbulence, were studied.

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

For fluidelastic excitation, the ratios of the effective crossflow velocity to the critical velocity of the first 20 vibration modes were calculated. The results indicate that no fluidelastic vibration will occur during steady state conditions as well as operational transients.

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 plugging limit for a Model F steam generator tube.

This model steam generator is also provided with a flow distribution baffle in the region of the tube sheet, to control where the flow enters the bundle. The baffle also determines the location of the blowdown pipe inlet in order to effectively remove any impurities that might accumulate at the point of zero horizontal velocity (i.e., due to the 90 degree turn upward). Blocking devices located adjacent to the downcomer regions and at the innermost U-bend tube row, at the tube sheet, minimize bypass flow, promoting flow into the central regions of the bundle. To avoid extensive crevice areas at the tube sheet, the tubes are full length roll expanded within the tube sheet bore.

5.4.2.1.3 Compatibility of Steam Generator Tubing with Primary and Secondary Coolants

As mentioned in subparagraph 5.4.2.1.1, 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 or under the same environmental conditions (water chemistry, sludge composition).

Localized steam generator tube diameter reductions were 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 0.71-inch diameter 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.

Denting is a term which describes a group of related phenomena resulting from corrosion of carbon steel in the crevices formed between the tubes and the tube support plates. The term "denting" has been applied to the secondary effects which include:

A. Tube diameter reduction

B. Tube support plate hole reduction

C. Tube support plate flow hole distortion, flow slot hour-glassing

D. Tube support plate expansion

E. Tube leakage

F. Wrapper distortion.

The mechanism which produces the effects cited involves an acid chloride environment in the tube crevices. In sequence, the process appears to occur as follows:

The crevice between the tube and the support plate is blocked as a result of deposition of chemical species present in the bulk water, including phosphate compounds, secondary system corrosion products, and minimal tube corrosion products. Once plugged, the annulus provides a site for concentration of various nominally soluble contaminants, such as chlorides, sulfates, etc. Recent studies indicate that in the absence of nonvolatile, alkalizing species, there may exist the potential for production of an acid solution by hydrolysis of such compounds as magnesium chloride, nickel phosphate, copper chloride, various ferrous salts, etc. In an acid chloride solution, the corrosion film on the carbon steel is converted from protective in character, to a thick, non-protective oxide of low density which assumes a laminar configuration subject to disruption due to the volume mismatch between the oxide and the base metal. The buildup of the thick oxide in the nominal 14 mil radial gap between the tube and the support plate causes sufficient force to be exerted against the tube to cause plastic deformation locally. The reaction to these forces can cause distortion of the circulation holes in the plate, in both the flow holes between the tubes and in the central flow slots between the inlet and outlet halves of the tube bundle. In the most extreme cases, as corrosion proceeds and in-plate forces accumulate, the entire plate increases in diameter and the ligaments between the holes in the plate may crack. Ovalization of the tubes at the intersections results in high strains, leading to tensile stress on the tube ID and possible leakage by intergranular cracking. A similar result may occur at the apex of first row; i.e., the smallest radius U-bends, if sufficient distortion of the top support plate flow slots occurs, resulting in leg displacement, ovalization, and high strains.

The tube leakage and support plate effects do not pose a safety problem with respect to release of radioactivity or effects on accident calculations, but the frequency of leakage and resultant repair shutdowns does present an economic concern to the operators. The utilization of preventive plugging, therefore, serves to maintain availability and to permit orderly planning of long-term corrective action.

The occurrence of denting has thus far been associated exclusively with plants having a history of chloride contamination due to condenser leakage. Moreover, it has recently been noted that Maine Yankee and Millstone Point 2, non-Westinghouse plants which have used AVT exclusively, have apparently incurred denting also; sea water is used for cooling the condensers at both of these plants.

Research into the causes of denting was initiated shortly after the discovery of the denting condition. Initially, dented tubes were removed for laboratory examination. Subsequently, tube support plate samples containing sections of tubing were also removed for analysis from operating plants.

The initial hard data on the nature of the denting phenomenon were derived from these tube/support plate samples which revealed the thick oxide buildup, the tube diameter reduction, and chemical makeup of the crevice-filling materials. It was observed that there was only minor corrosive attack on the tube material, approximately 0-2 mil circumferential thinning, and that the crevice contained a thick layer of almost pure magnetite (Fe_3O_4); other chemical constituents included Inconel-metal-phosphate corrosion products close to the tube, and general secondary system contaminants between the Fe_3O_4 and the phosphate layer. There was evidence of copper deposits, and the oxide was laced with chlorides.

Armed with those general observations, a series of crevice-with-contaminants test geometries were evaluated. Denting was produced first in reverse as "bulging" when a carbon steel plug was inserted into an Inconel tube to form the crevice; later, heated crevice assemblies with heat transfer were shown to be effective dent simulators; finally, denting in model boilers equipped with plant-type geometrical configurations was demonstrated. While pure, uncontaminated AVT environments have to date been found to be innocuous, it has been shown that the PO_4 to AVT transition was unnecessary to initiate the denting process. Only the presence of acid chloride solutions has been found to be a common factor. Nickel chloride, ferrous, or cupric chloride solutions have been shown to be corrosive, and have also produced measurable denting. Thus far, test data indicate that phosphates, calcium hydroxide, and borates seem to retard the dent process; morpholine, among the common volatile amines, shows a beneficial effect on the corrosion rate of carbon steel.

Model boiler tests have been used to evaluate the adequacy of the AVT chemistry specifications adopted in 1974. With one significant alternation, the specifications appear to be adequate to preserve tube integrity: the frequency and the length of time above the chloride limit for normal operation (0.15 ppm) must be limited. Westinghouse is working to prepare a uniform specification to be applied to all plants, which will limit the chloride concentration, the number of consecutive days beyond the normal specification, and the number of incidents per year.

As has been discussed, tube denting is a result of support plate corrosion products compressing the tube while dilating the tube hole. Therefore, measures to inhibit denting concentrate on removing the corrosion mechanisms, medium, and materials.

The tube support plates used in the Model F will be 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. The support plates will also be designed with broached tube holes rather than drilled holes. The broached tube hole design promotes high velocity flow among the tubes, sweeping any impurities away from the support plate location.

Additional measures are incorporated in the Model F design to prevent areas of dryout in the steam generator and accumulations of sludge in low-velocity areas. Modifications to the wrapper have increased water velocities across the tube sheet. A flow distribution baffle is provided which forces the low flow area to the center of the bundle. Increased capacity blowdown pipes have been added to enable continuous blowdown of the steam generators at a high volume. The intakes of these blowdown pipes are located below the center cut out section of the flow distribution baffle in the low-velocity region where sludge may be expected to accumulate. Continuous blowdown provides maximum protection against inleakage of impurities from the condenser.

Although the tubes themselves are not significantly corroding, they are affected by the results of the tube support plate corrosion. The continued corrosion of the carbon steel support plates results in stresses being induced in the tubes at the location of the deformation. Stress may also be induced in the U-bend section when support plate movement results from the tube hole dilation. Thermal treatment of Inconel tubes has been shown to be effective in limiting stress corrosion cracking. Tubing used in the Model F will be thermally treated in accordance with a laboratory-derived treatment process.

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.

To eliminate these localized areas of corrosion over the long-term operation of the unit, it was decided that the use of phosphates for steam generator water chemistry control would be eliminated. The adoption of the all volatile treatment (AVT) 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 concentrations of impurities in the steam generator water, thus reducing the potential for formation of highly concentrated solutions in low 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 program minimizes the possibility for recurrence 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 has 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.

Among the plants with Inconel tubes which have operated successfully with AVT, without evidence of tubing corrosion, are the Hanford N-Reactor and Prairie Island No. 2. The Hanford N-Reactor has operated with AVT since 1964; there have been no tube leaks, and annual eddy current inspections have revealed no corrosion defects.

Additional extensive operating data are presently being accumulated with the conversion to AVT chemistry. A comprehensive program of steam generator inspections, including the requirements of Regulatory Guide 1.83, 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.

5.4.2.1.4 Cleanup and Secondary Side Materials

Several methods are employed to clean operating steam generators of corrosion-causing secondary side deposits. Sludge lancing, a procedure in which a hydraulic jet inserted through an access opening (handhole) loosens deposits which are removed by means of a suction pump, can be performed when the need is indicated by the results of steam generator tube inspection. Six 6-inch access ports are provided for sludge lancing and inspection. Three of these are located above the tube sheet and three above the flow distribution baffle. Continuous blowdown is performed to regulate water chemistry. The location of the blowdown piping suction, adjacent to the tube sheet and in a region of relatively low flow velocity, facilitates the efficient removal of impurities that have accumulated on the tube sheet.

5.4.2.2 Steam Generator Inservice Inspection

The steam generator 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 the edition of the ASME Code, Division 1, Section XI required by 10 CFR 50.55a, effective January 5, 1977. 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.

Regulatory Guide 1.83 provides recommendations concerning the inspection of tubes. Those recommendations cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspections, methods of recording, and required actions based on findings. Agreement with Regulatory Guide 1.83 is discussed in appendix 3A. Regulatory Guide 1.121 provides recommendations concerning tube plugging. The minimum requirements for inservice inspection of steam generators, including tube plugging criteria, are established as part of the technical specifications.

5.4.2.3 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, the current philosophy is to design all pressure-retaining parts of the steam generator, and thus both the primary and secondary pressure boundaries, to satisfy the criteria specified in Section III of the ASME Code for Class 1 components. The design stress limits, transient conditions, and combined loading conditions applicable to the steam generator are discussed in subsection 3.9.1. 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. The accident analysis of a steam generator tube rupture is discussed in chapter 15.

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 reactor coolant system surfaces. The water chemistry of the steam side and its effectiveness in corrosion control are discussed in chapter 10. Compatibility of steam generator tubing with both primary and secondary coolants is discussed further in subparagraph 5.4.2.1.3.

The steam generator is designed to prevent unacceptable damage from mechanical or flow-induced vibration. Tube support adequacy is discussed in subparagraph 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 subsection 3.9.1. Further consideration is given in subparagraph 5.4.2.5.4 to the effect of tube wall thinning on accident condition stresses.

5.4.2.4 Design Description

The steam generator is a Model F, vertical shell and U-tube evaporator, with integral moisture separating equipment. Figure 5.4-4 shows the model, indicating several of its improved design features which are described in the following paragraphs:

On the primary side, the reactor coolant flows through the inverted U-tubes, entering and leaving through nozzles located in the 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 through an annulus between the tube wrapper and shell. The ring is offset, with respect to the tube bundle, in order to distribute the colder feedwater in a manner to maximize heat transfer in the bundle. 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 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 in the relatively low-velocity fluid for sludge deposition. Flow blockers discourage the feedwater from flowing up the bypass lane as it enters the tube bundle where it is converted to a steam-water mixture. Subsequently the steam-water mixture from the tube bundle rises into the steam drum section, where sixteen individual centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary separators for further moisture removal, increasing its quality to a minimum of 99.75 percent. The moisture separators reintroduce the separated water, which is combined with entering feedwater to flow back down the annulus between the wrapper and shell for another recirculation through the steam generator. The dry steam exits from the steam generator through the outlet nozzle, which is provided with a steam flow restrictor, as described in subsection 5.4.4.

5.4.2.5 Design Evaluation

5.4.2.5.1 Forced Convection

The limiting case for heat transfer capability is the "nominal 100 percent design" case. The 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 parameter selection is 1503 Btu/hr-ft^2-F. This coefficient is approximately 5 to 10 percent less than the heat transfer performance experienced at a number of operating plants.

The coefficient incorporates a specified fouling factor resistance of 0.00006 hr-ft^2-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 Flow

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. Thus, natural circulation is assured for the removal of decay heat during hot shutdown in the unlikely event of loss of forced circulation.

5.4.2.5.3 Mechanical and Flow-Induced Vibration Under Normal Operation

In the design of steam generators, the possibility of vibratory failure of tubes due to either mechanical or flow-induced excitation is 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 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 vibratory failure in heat exchanger components could be the effect of hydrodynamic excitation by the secondary fluid on the outside of the tubes.

Consideration of secondary flow-induced vibration involves 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) fluid elastic vibration. The steam generator is analyzed to ensure that the tube natural frequency is well above the anticipated vortex shedding frequency and that unstable fluid elastic vibration does not exist. In order to achieve this, adequate tube supports must be provided. Evaluations using the specific parameters for the appropriate steam generator model confirm the integrity of the support system.

While the behavior of tube arrays under cross flow in actual operating units is given consideration, the high temperature and pressure limit the amount and quality of information obtained. As a result, it was deemed prudent to undertake a research program that would allow in-depth study and testing in this area of interest. Facilities included a water tunnel and wind tunnel, which were specifically built to study the vibration behavior of tubes in arrays.

The results of this research confirm both the vortex shedding and the fluid elastic mechanisms. Both mechanisms have been considered in the steam generator design. Testing is also conducted using specific parameters of the steam generator in order to show that the support system is adequate.

Summarizing the results of analysis and tests of steam generator tubes for flow-induced vibration, it can be stated that a check of support adequacy has been made using all published techniques believed appropriate to heat exchanger tube support design. In addition, the tube support system is consistent with accepted standards of heat exchanger design utilized throughout the industry (spacing, clearance, etc.). Furthermore, the design techniques are supplemented with a continuing research and development program to understand the complete mechanism of fluid-structural interaction, and it should be noted that successful operational experience with several steam generator designs has given confidence in the overall approach to the tube support design problem.

5.4.2.5.4 Allowable Tube Wall Thinning

Over a period of time under the influence of the operating loads and environment in the steam generator, some tubes may become degraded in local areas. To determine the condition of the tubing, in-service inspection using eddy-current techniques is performed in accordance with the guidelines of US NRC Regulatory Guide 1.83, Reference 2. Partially degraded tubes are satisfactory for continued service provided that defined stress and leakage limits are satisfied, and that the prescribed structural limit is adjusted to take into account possible uncertainties in the eddy current inspection, and an operational allowance for continued tube degradation until the next scheduled inspection. 5 321

The US NRC Regulatory Guide 1.121, Reference 3, describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection shall be removed from service. The level of acceptable, degradation is referred to as the "repair limit."

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5.4.2.6 Quality Assurance

The steam generator quality assurance program is given in table 5.4-4.

Radiographic inspection and acceptance standards shall be 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:

Nozzle to shell

Support brackets

Instrument connection (secondary)

Temporary attachments after removal

All accessible pressure-retaining welds after hydrostatic test.

Magnetic particle inspection and acceptance standards are in accordance with 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.

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.

In addition, the heat transfer tubes are subjected to a hydrostatic test pressure prior to installation into the vessel, which is not less than 1.25 times the primary side design pressure.

5.4.3 REACTOR COOLANT PIPING

5.4.3.1 Design Bases

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

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

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

Stainless steel 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 the scope of ANSI B36.19 conforms to ANSI B36.10.

The minimum wall thicknesses of the loop pipe and fittings are not less than that calculated using the ASME III Class 1 formula of Paragraph NB-3641.1 (3) with an allowable stress value of 17,550 psi. The pipe wall thickness for the pressurizer surge line is Schedule 160. The minimum pipe bend radius is 5 nominal pipe diameters; ovality does not exceed 6 percent.

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

Processing and minimization of sensitization are discussed in subsection 5.2.3.

Flanges conform to ANSI B16.5.

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

Inservice inspection is discussed in subsection 5.2.4.

5.4.3.2 Design Description

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

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.

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

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. There will be no electroslag welding on these components. All smaller piping that comprises part of the RCS, such as the pressurizer surge line, spray and relief line, loop drains, and connecting lines to other systems is also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is stainless steel. All joints and connections are welded, except for the pressurizer code safety valves, where flanged joints are used. Thermal sleeves are installed on the pressurizer spray line and surge line nozzles.

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

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

B. Letdown, excess letdown, loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.

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

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

E. Two of the three taps in each resistance temperature detector hot leg connection.

F. The hot leg sample connections and the loop 3 thermowell, both located on the horizontal centerline.

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

A. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the 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 narrow range temperature detector hot leg wells are located in scoops that extend into hot legs of the reactor coolant to detect a representative temperature.

D. The wide and narrow range temperature detectors are located in resistance temperature detector wells that extend into both hot and cold legs of the reactor coolant pipes.

The resistance temperature detectors in thermowells for each reactor coolant loop hot and cold leg are provided so that individual temperature signals may be developed for use in the reactor control and protection system. The resistance temperature detectors in thermowells within scoops detect a representative hot leg temperature, and are extended into the flow stream at locations 120° apart in the cross-sectional plane, on the reactor coolant leg.

The resistance temperature detector thermowell on each cold leg is located downstream of the pump discharge. Because of the mixing action of the pump, only one thermowell is required to obtain a representative sample. This thermowell is located as close as possible to the weld connection at the pump discharge and is in the same relative position in each loop. 84

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Signals from these instruments are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{hot}, minus 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.

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 two of the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.

D. Residual heat removal suction lines from the reactor coolant loops up to the designated 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.

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H. Loop fill, loop drain, sample[a], and instrument[a] lines to or from the designated isolation valve to or from the reactor coolant loops.

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

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

붙임 2-1 : INSERT

INSERT 'A' : The narrow range temperature detector hot leg wells are located in scoops that extend into hot legs of the reactor coolant to detect a representative temperature.

INSERT 'B' : The resistance temperature detectors in thermowells within scoops detect a representative hot leg temperature, and are extended

INSERT 'C' : Hot leg temperature measurement on each loop will be accomplished using three fast response, narrow range, well type dual element RTDs located within the three scoops, which extend into the flow stream at locations 120 degrees apart in the cross sectional plane on the reactor coolant hot leg. Cold leg temperature measurement on each loop will be accomplished using one fast response, narrow range, well type dual element RTD located in cold leg at the discharge of the reactor coolant pump.

INSERT 'D' : RTD(including thermowell) response time + electronic delay time ≤ 8 seconds and electronic delay time ≤ 4 seconds. RTD(including thermowell) response time is the characteristic time constant for the RTD(including thermowell) and is used as a first order lag in the analysis. The electronic delay is the trip circuit channel electronics delay plus the time for the reactor trip breakers to open and time for the CRDM stationary grippers to disengage(gripper release time)

붙임4. 개정(안)

located above the elevation of the reactor vessel nozzles to permit valve repair during cold shutdown, without draining the RCS.

In addition, vents and drains are provided in each manifold to be used, in conjunction with the isolation valve, for maintenance.

삭제

Signals from these instruments are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{hot}, minus 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.

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 two of the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.

D. Residual heat removal suction lines from the reactor coolant loops up to the designated 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. Resistance temperature detector manifold bypass loop piping.~~

삭제

H. Loop fill, loop drain, sample[a], and instrument[a] lines to or from the designated isolation valve to or from the reactor coolant loops.

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

---

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

J. Resistance temperature detector scoop element, pressurizer spray scoop, sample connection[(a)] with scoop, reactor coolant temperature element installation boss, and the temperature element well itself.

K. All branch connection nozzles attached to reactor coolant loops.

L. 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.

M. Seal injection water lines to the reactor coolant pump to the designated check valve (injection line).

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

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

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

5.4.3.3 Design Evaluation

Piping and load and stress evaluation for normal operating loads, seismic loads, blowdown loads; and combined normal, blowdown, and seismic loads is discussed in section 3.9.

5.4.3.3.1 Material Corrosion/Erosion Evaluation

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

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

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

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

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality listed in table 5.2-5. Maintenance of the water quality to minimize corrosion is accomplished using the chemical and volume control system and sampling systems which are described in chapter 9.

5.4.3.3.2 Sensitized Stainless Steel

Sensitized stainless steel is discussed in subsection 5.2.3.

5.4.3.3.3 Contaminant Control

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

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

5.4.3.4 Tests and Inspections

The RCS piping quality assurance program is given in table 5.4-6.

Volumetric examination is performed throughout 100 percent of the wall volume of each pipe and fitting in accordance with the applicable requirements of Section III of the ASME Code for 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 ASME Section III. Acceptance standards are in accordance with the applicable requirements of ASME Section III.

The pressurizer surge line conforms to SA-376 Grade 304, 304N, or 316 with supplementary requirements S2 (transverse tension tests), The S2 requirement applies to each length of pipe. An ultrasonic test is performed over the entire volume per Section III of the

ASME Code requirements, except that the acceptance standard is a 3 percent "V" notch. In addition, a dye penetrant examination is performed on all accessible surfaces.

The end of pipe sections, branch ends and fittings are machined back to provide a smooth weld transit ion 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. Several protective advantages are thereby provided: Rapid rise in containment pressure is prevented, the rate of heat removal from the reactor coolant is kept within acceptable limits, thrust forces on the main steam line piping are reduced, and stresses on internal steam generator components, particularly the tube sheet and tubes, are limited. The restrictor is also designed to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

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

5.4.4.3 Design Evaluation

The 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 4.0 psi. This is based on a design flowrate of 4.31×10^6 - pounds per hour. 382
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 inspections beyond those during fabrication are required.

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 subsection is not applicable to pressurized water reactors.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

The residual heat removal system (RHRS) transfers heat from the reactor coolant system (RCS) to the component cooling system (CCS) to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal or safety-grade plant cooldown, and maintains this temperature until the plant is started up again.

Parts of the RHRS also serve as parts of the emergency core cooling system (ECCS) for accident mitigation (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 refueling operations.

Nuclear plants employing a similar RHRS design are discussed in section 1.3.

5.4.7.1 Design Bases

The RHRS design parameters are listed in table 5.4-8.

The RHRS is designed to operate in conjunction with other plant systems to reduce the temperature of the RCS during the second phase of normal plant cooldown.

The RHRS is placed in operation approximately 4 hours after reactor shutdown when the temperature and pressure of the RCS are approximately 350F and 400 psig, respectively. Assuming that two heat exchangers and two pumps are in service, and that each heat exchanger is supplied with component cooling water at design flow and temperature, the RHRS is designed to reduce the temperature of the reactor coolant from 350 to 140F within 16 hours. 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 following reactor shutdown from an extended run at full power.

Assuming that only one heat exchanger and pump are in service and that the heat exchanger is supplied with component cooling water at 5200 gal/min and 105F, the RHRS is capable of reducing the RCS temperature from 350F to less than 200F within 30 hours.

The RHRS is also designed to operate in conjunction with the other systems of the cold shutdown design in order to address the functional requirements proposed by Regulatory Guide 1.139, Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown. The cold shutdown design enables the nuclear steam supply to be taken from hot standby to cold shutdown conditions using only safety-grade systems, with or without offsite power, and with the most limiting single failure. The cold shutdown design also enables the RCS to be taken from hot standby to conditions that will permit initiation of RHRS operation within 36 hours. The reliability of the cold shutdown design is discussed in subparagraph 5.4.7.2.6.

The RHRS is designed to be isolated from the RCS whenever the RCS pressure exceeds the RHRS design pressure. The RHRS is isolated from the RCS on the suction side by two motor-operated valves in series on each suction line. Each of the normally closed motor-operated valves is interlocked to prevent its opening if RCS pressure is greater than approximately 425 psig and to automatically close if RCS pressure exceeds 750 psig. (These interlocks are discussed in more detail in subparagraph 5.4.7.2.4 and subsection 7.6.2.) The RHRS is isolated from the RCS on the discharge side by 3 check valves in each return line. Also provided on the discharge side is a normally open motor-operated valve downstream of each RHR heat exchanger.

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

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve designed to relieve the maximum possible back leakage through the valves isolating the RHRS from the RCS.

The RHRS is designed for a single nuclear power unit and is not shared among nuclear power units.

The RHRS is designed to be fully operable from the control room for normal operation. Remote-manual operations required of the operator include: opening the suction isolation valves, positioning the flow control valves downstream of the residual heat exchangers, and starting the residual heat removal pumps. Manual actions are also discussed in subparagraphs 5.4.7.2.6 and 5.4.7.2.7. By nature of its redundant two-train design, the RHRS is designed to accept all major component single failures with the only effect being an extension in the required cooldown time. There are no motor-operated valves in the RHRS that are subject to flooding. Provisions to protect equipment from flooding are discussed in section 3.4. Although Westinghouse considers it to be of low probability, spurious operation of a single motor-operated valve can be accepted without loss of function as a result of the redundant two-train design.

Provisions incorporated in the design to ensure that the system will operate when needed are discussed further in subparagraph 5.4.7.2.6.

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

5.4.7.2 System Design

5.4.7.2 Schematic Piping and Instrumentation Diagrams

The RHRS, as shown in figures 5.4-5 (piping and instrumentation diagram) and 5.4-6 (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 legs 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 normally closed motor-operated valves in series located inside the containment. Separate Class 1E power sources are provided for the four suction isolation valves. This arrangement ensures that single failure requirements for RHR initiation will be met. Each discharge line is isolated from the RCS by three check valves 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 not part of the RHRS; these valves are shown as part of the ECCS: see figure 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 requirements of the reactor vessel and by the number 1 seal differential pressure and net positive suction head (NPSH) 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 residual heat exchangers. A line containing a flow control valve bypasses each residual heat exchanger and is used to maintain a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature, and total flow.

The RHRS is also 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 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 injection of borated water from the refueling water storage tank into the RCS cold legs during the injection phase following a loss-of-coolant accident (LOCA).

In its capacity as the low-head portion of the ECCS, the RHR provides long-term recirculation capability for core cooling following the injection phase of the LOCA. This function is accomplished by aligning the RHRS to take fluid from the containment recirculation sump, cooling it by circulation through the residual heat exchangers, and supplying it to the core directly as well as via the centrifugal charging pumps. 253 321

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

5.4.7.2.1.1 Description of Component Interlocks. In order to perform their ECCS function, the residual heat removal pumps are interlocked to start automatically on receipt of a safety injection signal (see section 6.3).

The RHRS suction isolation valves in each inlet line from the RCS are separately interlocked to prevent their being opened when RCS pressure is greater than approximately 425 psig, and to automatically close if RCS pressure exceeds 750 psig. These interlocks are described in more detail in subparagraph 5.4.7.2.4 and subsection 7.6.2.

The RHRS suction-isolation valves from the RCS are also interlocked to prevent their being opened unless the isolation valves in the following lines are closed:

A. Recirculation line from the residual heat exchanger outlet to the suction of the high head safety injection pumps.

B. Residual heat removal pump suction line from the refueling water storage tank.

The motor-operated valves in the RHRS miniflow bypass lines are interlocked to open when the residual heat removal pump discharge flow is less than approximately 500 gal/min and close when the flow exceeds approximately 1000 gal/min.

The motor-operated isolation valves in the recirculation lines from the residual heat exchanger outlet to the suctions of the high-head safety injection pumps are interlocked such that they cannot be opened unless either of the series RHRS suction isolation valves from the RCS in the corresponding subsystem is closed.

5.4.7.2.2 Equipment and Component Descriptions

The materials used to fabricate RHRS components are in accordance with the applicable code requirements. All parts of components in contact with borated water are fabricated or clad with austenitic stainless steel or equivalent corrosion-resistant material. Component parameters are given in table 5.4.7-3.

5.4.7.2.2.1 Residual Heat Removal Pumps. Two pumps are installed in the RHRS. The pumps are sized to deliver reactor coolant flow through the residual heat exchangers to meet the plant cooldown requirements. The use of two separate residual heat removal trains ensures that cooling capacity is only partially lost should one pump become inoperative.

The residual heat removal pumps are protected by the miniflow bypass lines which ensure flow to the pump suction should the pump discharge be isolated or the RCS pressure be above the shutoff head of the pump. A valve located in each miniflow line is controlled 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 500 gal/min and close when the flow exceeds approximately 1000 gal/min.

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, in-line, 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 residual heat removal pumps also function as the low-head safety injection pumps in the ECCS. (See section 6.3 for further information and for typical residual heat removal pump performance curves.)

5.4.7.2.2.2 Residual Heat Exchangers. Two residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and minimal temperature differences between reactor coolant and component cooling water existing 20 hours after reactor shutdown.

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

The residual heat exchangers are of shell and U-tube design, vertically mounted. 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).

5.4.7.2.2.3 Residual Heat Removal System Valves. Valves that perform a modulating function are equipped with two sets of packings and an intermediate leakoff connection that discharges to the drain header.

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.

5.4.7.2.3 System Operation

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

At initiation of 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 to augment control of the 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 open. 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.

Steam bubble formation in the pressurizer is accomplished by energizing the pressurizer heaters and by increasing the letdown flow above the charging flow. After the pressurizer bubble has been formed, the reactor coolant pumps are started to heat up the system. When the pressurizer water level reaches the no-load programmed setpoint, pressurizer level control is shifted to the normal operational means. The RHRS is then isolated from the RCS and the system pressure is controlled by normal letdown, pressurizer spray, and pressurizer heaters.

5.4.7.2.3.2 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.

5.4.7.2.3.3 Plant Shutdown. Plant shutdown is defined as the operation which brings the plant from no-load temperature and pressure to a cold shutdown condition, i.e., to a subcritical condition with the reactor coolant temperature no greater than 200F.

5.4.7.2.3.4 Normal Cold Shutdown. The initial phase of a normal plant shutdown is accomplished by transferring heat from the RCS to the steam and power conversion system. Circulation of the reactor coolant is provided by the reactor coolant pumps and heat removal is accomplished by using the steam generators and dumping steam to the condenser.

In conjunction with this portion of the cooldown, the reactor coolant is borated to the concentration required for cold shutdown and depressurized to a pressure permitting RHRS operation. Boration and makeup for the contraction of the RCS due to cooling are performed using the charging, letdown, and makeup control portions of the CVCS. The depressurization function is performed by initiating pressurizer spray from the discharge of the operating reactor coolant pump.

When the reactor coolant temperature and pressure are reduced to approximately 350F and 400 psig, approximately 4 hours after reactor shutdown, the second phase of cooldown starts with the RHRS being placed in operation.

Startup of the RHRS includes a warmup period during which time reactor coolant flow through the heat exchangers 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. As the reactor coolant temperature decreases, the reactor coolant flow through the residual heat exchanger is increased by adjusting the control valve in each heat exchanger tube side outlet line.

During plant shutdown with the RHRS in operation, operation with a steam bubble in the pressurizer is maximized to provide RCS pressure control. Pressure control is augmented by regulating the charging flowrate and the rate of letdown from the RHRS to the CVCS.

After the reactor coolant is reduced below a temperature of 160F and the reactor coolant pump is stopped, cooling of the pressurizer is continued by providing auxiliary spray from the CVCS.

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

5.4.7.2.3.5 Safety-Grade Cold Shutdown. It is expected that the systems normally used for cold shutdown will be available any time the operator chooses to perform a reactor cooldown. However, to ensure that the plant can be taken to cold shutdown at any time, the safety-grade cold shutdown design enables the RCS to be taken from no-load temperature and pressure to cold conditions using only safety-grade systems, with only onsite or offsite power available, and assuming the most limiting single failure.

Should portions of normal shutdown systems be unavailable, the operator should maintain the plant in a hot standby condition while making the normal systems functional. Local manual actions could be performed where such are permitted by the prevailing environmental conditions. Appropriate procedures are provided for the use of safety-grade backups contingent upon the inability to make normal systems available. The operator should use any of the normal systems that are available in combination with the safety-grade backups for the systems that cannot be made operable. The safety-grade provisions are to be used only upon the inability to make available the equipment normally used for the given function.

The safety-grade cold shutdown design enables the operator to maintain the plant in hot standby for at least 4 hours. Since it is assumed that the reactor coolant pumps are not available, circulation of the reactor coolant is provided by natural circulation with the reactor core as the heat source and the steam generators as the heat sink. Heat removal is accomplished via the steam generator power-operated relief valves and auxiliary feedwater system.

For the purpose of establishing conservative requirements for auxfeed capacity, it is assumed that the RCS is borated to cold shutdown concentration prior to cooling the RCS. The charging pumps are used to provide 4 weight percent boric acid from the boric acid tanks to the RCS at a rate of approximately 50 gal/min. The borated water is delivered to the RCS cold legs via the high head safety injection lines. Reactor coolant pump seal injection is also maintained. To accommodate this addition to RCS inventory, continuous letdown is discharged from the reactor vessel head letdown line to the pressurizer relief tank.

Following boration to cold shutdown concentration, the safety-grade cooldown is accomplished by increasing the steam dump from the steam generator power-operated relief valves to attain a rate of primary side cooling of at least 35F/hr. In conjunction with this portion of the cooldown, the charging pumps are used to deliver refueling water to make up for primary contraction due to cooling. Makeup is also provided for the RCS inventory discharged when the reactor vessel head letdown path is periodically cycled to provide head cooling. Upon approaching the end of this phase of cooldown, the RCS is depressurized by venting the pressurizer through use of the safety-grade pressurizer power-operated relief valves.

To ensure that the accumulators do not repressurize the RCS, the accumulator discharge valves are closed prior to the RCS pressure dropping below the accumulator discharge pressure. Additionally, each accumulator is provided with two Class 1E solenoid-operated valves in parallel to ensure that the accumulator may be vented should it fail to be isolated from the RCS.

When the reactor coolant temperature and pressure are reduced to approximately 350F and 400 psig respectively, the second phase of cooldown starts with the RHRS being placed in operation.

As safety-grade cooldown continues, the reactor vessel head letdown line is periodically opened to increase head cooling and to accommodate any additional input to the RCS, such as reactor coolant pump seal injection. Since loss of nonsafety-grade equipment results in a loss of the air supply to the flow control valves that are normally used to limit the initial RHRS cooldown rate, the operator may choose to use only one of the residual heat removal subsystems. Should a single failure, such as that of a RHRS component or of an emergency power train (when only onsite power is available), limit operation to one of the residual heat removal subsystems, the operator would open the series isolation valves in the suction of only the operable residual heat removal subsystem. In this case, the operator would also close the cross-connect isolation valves between the subsystems. Residual heat removal would continue under these conditions until the redundant subsystem could be made available.

For power uprate operation, a computer code analysis was performed to evaluate the capability of the cold shutdown design to enable RCS to be taken from hot standby to RHR entry condition through natural circulation cooldown operation. This analysis utilizes the methods and assumptions approved by the USNRC and was performed assuming the restrictions of USNRC Branch Technical Position (BTP) RSB 5-1. The restrictions include the use of only safety-grade equipment, the concurrent loss of offsite power, a single failure, and auxiliary feedwater usage within the minimum available capacity. As a natural circulation cooldown analysis, the RCS was proved to be cooled and depressurized to RHR entry conditions, and the amount of safety-grade auxiliary feedwater used is about 260,000 gallons which is well within the minimum available CST capacity of 470,000 gallons. 321

A result of computer code simulation for a natural circulation cooldown event of the KNU 5 & 6 nuclear steam supply system from hot standby to RHR entry conditions is presented in Appendix 5A.

5.4.7.2.3.6 Refueling. Both residual heat removal pumps are utilized during refueling to pump borated water from the refueling water storage tank to the refueling cavity. During this operation, the residual heat removal pumps are stopped.

the isolation valves in the inlet lines of the RHRS are closed, the isolation valves to the refueling water storage tank are opened, and the residual heat removal pumps are restarted.

The reactor vessel head is lifted slightly. The refueling water is then pumped into the reactor vessel through the normal RHRS return lines and into the refueling cavity through the open reactor vessel. The reactor vessel head is gradually raised as the water level in the refueling cavity increases. After the water level reaches the normal refueling level, the residual heat removal pumps are stopped, the inlet isolation valves are opened, the refueling water storage tank supply valves are closed, the residual heat removal pumps are restarted, and residual heat removal is resumed.

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

Following refueling, the residual heat removal pumps are used to drain the refueling cavity to 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 flow path reestablished. The remainder of the water is removed from the refueling canal via a drain connection in the bottom of the canal.

5.4.7.2.4 Control

Each inlet line to the RHRS is equipped with a pressure relief valve sized to relieve the combined flow of two of the charging pumps at the relief valve set pressure. These relief valves also protect the system from inadvertent overpressurization during plant shutdown or startup. Each valve has a relief choke flow capacity of 850 gal/min at a set pressure of 450 psig. An analysis has been conducted to confirm the capability of the RHRS relief valve to prevent overpressurization in the RHRS. All credible events were examined for their potential to overpressurize the RHRS. These events included normal operating conditions, infrequent transients, and abnormal occurrences. The analysis confirmed that one relief valve has the capability to keep the RHRS maximum pressure within code limits.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the valve separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gal/min at a set pressure of 600 psig. These relief valves are located in the ECCS (see figure 6.3-1).

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 equipment drain tank of the radioactive drain system.

The design of the RHRS includes two normally closed motor-operated gate isolation valves in series on each inlet line between the high-pressure RCS and the lower pressure RHRS. The valves are closed during normal operation and are only opened for residual heat removal during plant shutdown after the RCS pressure is reduced to approximately 400 psig or lower, and RCS temperature is reduced to approximately 350F. During a plant startup the valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above 400 psig. These isolation valves are provided with independent and diverse "prevent-open" and "auto-closure" interlocks which are designed to prevent possible exposure of the RHRS to normal RCS operating pressure.

The two normally closed inlet isolation valves in each residual heat removal subsystem are independently and diversely interlocked with diverse RCS wide-range pressure instruments to prevent their being opened whenever the RCS pressure is greater than approximately 425 psig. The valves automatically close if the RCS pressure increases to 750 psig during a plant startup. The use of independently powered motor-operated valves in each of the two inlet lines, along with independent and diverse pressure interlock signals to each of the series valves, ensures a design that meets applicable single-failure criteria. Independence is accomplished by aligning separate RCS pressure transmitters to each suction valve. The transmitters are powered by the same vital bus as the motor-operated valve (MOV). Diversity is accomplished through the use of two types of RCS wide-range pressure transmitters that employ different pressure-sensing principles. 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, in combination with plant operating procedures, provide the means for accomplishing the protective function. For further information on the instrumentation and control features, see subsection 7.6.2.

The RHR inlet isolation valves are provided with control switches with integral 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 three check valves in series. These check valves are located in the ECCS and their testing is described in paragraph 6.3.4.2.

5.4.7.2.5 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. Component codes and classifications for the RHRS and the other systems relied upon for safety-grade cold shutdown are given in section 3.2.

5.4.7.2.6 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 ensure that this function can be performed. The safety-grade systems that perform this function for all plant conditions except a LOCA are: the RCS and steam generators (NSSS), which operate in conjunction with the steam generator safety valves; the steam generator power-operated relief valves; the auxiliary feedwater system (BOP); and the RHRS (NSSS), which operates in conjunction with the component cooling water system and service water system (BOP). For LOCA conditions, the safety-grade system that performs the function of removing residual heat from the reactor core is the ECCS (NSSS), which operates in conjunction with the component cooling water system and the service water system.

The auxiliary feedwater system, along with the steam generator safety and power-operated relief valves (described in sections 10.4 and 10.3, respectively), 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 350F. The auxiliary feedwater system is capable of performing this function for an extended period of time following plant shutdown.

In order to achieve conditions that will permit initiation of RHRS operation, two other functions (boration and depressurization) must be performed. The boration function is normally provided by the CVCS. When the reactor coolant pumps are not available, due to loss of offsite power or following a manual pump trip, the depressurization function is also provided by the CVCS. The normal function and inherent reliability of the CVCS is discussed in detail in paragraph 9.3.4.1.

Should it be necessary to take the plant to a cold shutdown using only safety-grade systems, portions of the RCS (described in section 5.1) and the ECCS (described in section 6.3) are also relied upon for boration, letdown, makeup, and depressurization. These safety-grade provisions would be used only upon failure of the equipment normally used for the given function. Reliability of these backup systems is demonstrated in the following paragraphs.

Boration is accomplished by using the centrifugal charging pumps to supply boric acid from the boric acid tanks to the RCS via the safety injection lines in the ECCS. Four weight percent boric acid is delivered from one of the two boric acid tanks to the suction of the charging pumps by either of two boric acid transfer pumps. Makeup flow is supplied from the refueling water storage tank to the charging pumps through the redundant suction headers provided for safety injection. Redundant charging pumps are available to deliver borated water via either of the two high-head safety injection headers. Solenoid valves powered by different power trains are provided in these paths to further ensure the modulation capability necessary to establish the desired rate of boration or makeup.

Letdown to accommodate boration and any other addition to the RCS inventory is provided by the safety-grade reactor vessel head letdown path in the RCS. To ensure reliability of this function the letdown line is provided with parallel solenoid valves. The valves are designed to fail close such that both lines can always be isolated, and the two valves in the same line are powered by the same power train such that at least one line can always be made available. Downstream of these isolation valves, the letdown flow is directed either to the pressurizer relief tank via parallel solenoid valves or to the excess letdown heat exchanger located in the CVCS.

Depressurization is accomplished by discharging RCS inventory via the safety-grade pressurizer power-operated relief valves. Two parallel lines are provided with solenoid valves which can be remotely operated to relieve to the pressurizer relief tank. The ECCS accumulators are also provided with safety-grade isolation and venting capability in order to ensure that depressurization can be completed.

The pressurizer relief tank, the vessel head letdown valves, and the pressurizer relief valves are described in subsections 5.4.11, 5.4.12, and 5.4.13, respectively, and are shown in figure 5.1-1.

The systems used for boration/inventory control and for depressurization are remotely operable with either onsite or offsite power available and assuming the most limiting single failure. A failure modes and effects analysis (FMEA) of the portions of the RCS, ECCS, and CVCS that are used for safety-grade cold shutdown is included in the RHRS - Safety Grade Cold Shutdown Operations - FMEA (table 5.4-7). The reliability of these systems ensures that conditions permitting RHRS operation can be attained.

The RHRS is provided with two residual heat removal pumps and two residual heat exchangers arranged in two separate, independent flow paths. To ensure reliability, each residual heat removal pump is connected to a different power train. Each train is isolated from the RCS on the suction side by two normally closed motor-operated valves in series. The power supplies are designed such that a single failure neither prevents opening of at least one of the two subsystems nor prevents isolation of both subsystems. Each suction isolation valve is also independently and diversely interlocked to prevent exposure of the RHRS to the normal operating pressure of the RCS (see subparagraph 5.4.7.2.4).

The RHRS operation for normal conditions, even with a major failure, 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 a major single failure, such as failure of a residual heat removal pump, valve, or heat exchanger, or of an emergency power source, without impact on the redundant train continued heat removal. The only effect would be an extension of the time required for cooldown. The capability of the RHRS to accommodate a single component failure and still perform a normal or safety-grade cooldown is demonstrated in the RHRS - Safety-Grade Cold Shutdown Operations - FMEA (see table 5.4-7).

5.4.7.2.7 Manual Actions

The RHRS is designed to be fully operable from the control room for normal operation. Manual operations required of the operator include: opening the suction and discharge isolation valves, positioning the flow control valves downstream of the residual heat exchangers, and starting the residual heat removal pumps. No manual actions are required outside of the control room.

Assuming the most limiting single failure, the RHRS can still be operated without any operator action required outside of the control room, with the only effect being an extension in the cooldown time.

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, RHRS, and the CCS at stepwise intervals following the initiation of RHRS operation. Heat removal through the residual heat and component cooling heat exchangers is calculated at each interval by use of standard water-to-water heat exchanger performance correlations; the resultant fluid temperatures for the RCS, RHRS, and the CCS are calculated and used as input to the next interval heat balance calculation.

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

A. RHRS operation is initiated 4 hours after reactor shutdown.

B. RHRS operation begins at a reactor coolant temperature of 350F.

C. Thermal equilibrium is maintained throughout the RCS during the cooldown.

D. Component cooling water temperature during cooldown is limited to a maximum of 120F for no more than 4 hours.

E. RCS cooldown rates of 50F per hour are not exceeded.

Cooldown curves for one-train and two-train RHR operation are depicted in figure 5.4-7.

5.4.7.4 Preoperational Testing

Preoperational testing of the RHRS is addressed in chapter 14.

5.4.8 REACTOR WATER CLEANUP SYSTEM

This system is not applicable to PWRS.

5.4.9 MAIN STEAM LINE AND FEEDWATER PIPING

Discussions pertaining to the main steam line and feedwater piping are contained in the following sections or subsections:

A. Main Steam Line Piping - section 10.3

B. Main Feedwater Piping - subsection 10.4.7

C. Auxiliary Feedwater Piping - subsection 10.4.9

D. Inservice Inspection of A, B, and C - section 6.6.

5.4.10 PRESSURIZER

5.4.10.1 Design Bases

The general configuration of the pressurizer is shown in figure 5.4-9. 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 provides a point in the RCS where liquid and vapor can be maintained in equilibrium under saturated conditions for pressure control purposes for steady-state operations and during transients.

5.4.10.1.1 Pressurizer Surge Line

The surge line is sized to minimize the pressure drop between the RCS and the safety valves with maximum allowable discharge flow from the safety valves.
The pressurizer surge line nozzle diameter is given in table 5.4-10 and the pressurizer surge line dimensions are shown in figure 5.1-1.

5.4.10.1.2 Pressurizer Volume

The volume of the pressurizer is equal to, or greater than, the minimum volume of steam, water, or the 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 50 percent reduction from full load with automatic reactor control and 64 percent 321
at upper T_{avg} 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 will not empty following reactor trip and turbine trip.

F. The emergency core cooling signal is not activated during reactor trip and turbine trip.

5.4.10.2 Design Description

5.4.10.2.1 Pressurizer Surge Line

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

5.4.10.2.2 Pressurizer Vessel

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. A stainless steel liner is used on the pressurizer spray nozzle.

The surge line nozzle and removable electric heaters are installed in the lower pressurizer head. The heaters are removable for maintenance or replacement.

A retaining screen is located above the nozzle to prevent any foreign matter from entering the RCS. Baffles in the lower section of the pressurizer prevent an insurge of cold water from flowing directly to the steam/water interface and assist mixing.

Spray line nozzles, relief, and safety valve connections are located in the top head of the vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves also can be operated manually by a switch in the control room.

A small continuous spray flow is provided through a manual bypass valve around the power-operated spray valves to minimize boron concentration differences between pressurizer liquid and reactor coolant and to prevent excessive cooling of the spray piping.

During an outsurge from the pressurizer, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the low-pressure reactor trip point. During an insurge from the RCS, the spray system, which is fed from two cold legs, condenses steam in the pressurizer to prevent reaching the setpoint of the power-operated relief valves. 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 subsection 3.9.1. Additional details on the pressurizer design cycle analysis are given in subparagraph 5.4.10.3.5.

Pressurizer Support

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

Pressurizer Instrumentation

Refer to section 7.1 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

Spray Line Temperatures

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. Insufficient flow in the spray lines will result in low spray line temperature alarms.

Safety and Relief Valve Discharge Temperatures

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 through the associated valve.

5.4.10.3 Design Evaluation

5.4.10.3.1 System Pressure

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

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

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

During startup and shutdown, the rate of temperature change in the RCS is controlled by the operator. Heatup rate is controlled by reactor coolant pump energy and by the pressurizer electrical heating capacity.

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 flowrate via the residual heat removal system.

5.4.10.3.2 Pressurizer Performance

The normal operating water volume at full load conditions is about 56 percent at upper T_{avg} and about 48 percent at lower T_{avg} of the span volume. Under part-load conditions, the water volume in the vessel is reduced for proportional reductions in plant load to 22 percent of the span volume at zero power level. The various plant operating transients are analyzed and the design pressure is not exceeded with the pressurizer design parameters as given in table 5.4-10. 321

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 setpoints pressure, 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 valve is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open, and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping routed to the pressurizer forms a water seal which prevents the steam buildup back to the control valves. The spray rate is selected to prevent the pressurizer from reaching the operating setpoint of the power-operated 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 adequate spray, using as the driving force the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop in order to utilize the velocity head of the reactor coolant loop flow to add to the spray driving force. The spray valves and spray line connections are arranged so that the spray 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 chemical and volume control system to the pressurizer spray line is also provided. This additional facility provides auxiliary spray to the vapor space of the pressurizer during cooldown 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.

5.4.10.3.5 Pressurizer Design Analysis

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

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

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

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

C. Pressurizer spray is assumed to be initiated instantaneously to its design flowrate as soon as the RCS pressurizer pressure increases above 2275 psia. Spray is assumed to be terminated as soon as the pressurizer pressure decreases to 2275 psia.

D. Consistent with C above, unless otherwise noted, pressurizer spray is assumed to be initiated once per occurrence of each transient condition. The pressurizer surge nozzle is also assumed to be subject to one temperature transient per transient condition, unless otherwise noted.

E. At the end of each upset condition transient, the RCS is assumed to return to a no-load condition, with pressure and temperature changes controlled within normal limits.

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

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

5.4.10.3.6 Pressurizer Heater Power Supply - Conformance with NUREG 0737, Item II.E.3.1

The KNU 5 & 6 pressurizer backup heaters are powered and controlled from Class 1E sources. The motive and control power interfaces with the emergency buses are qualified in accordance with safety-grade requirements. The backup heaters consist of four heater banks which trip on any of the following signals: safety injection (SI), containment isolation-phase B (CIS-B), or 4.16-kV Class 1E bus loss of voltage. After SI or CIS-B reset and level recovery in the pressurizer, any combination of backup heater banks can be aligned for automatic operation. In the event of a loss of offsite power or safety injection signal, any of the backup heater banks can be manually activated in the main control room 60 seconds after emergency power becomes available. The required operator actions will be specified in the Emergency Operating Procedures.

A study on behalf of the Westinghouse Operating Plants Owner's Group (WOG), which applied to KNU 5 & 6, identified the minimum number of pressurizer heaters sufficient to maintain natural circulation and to determine the time available for connection of the emergency power source following a loss of offsite power. Results of this study have indicated that a heat input to the pressurizer heaters in the amount of 125 kW would be required at 60 minutes after initial heater trip to maintain the reactor coolant system conditions needed for natural circulation. The time available for connection of the emergency power is adequate due to the significant secondary water inventory maintained in the steam generators.

5.4.10.4 Tests and Inspections

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

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

Support skirt to the pressurizer lower head

Surge nozzle to the lower head

Safety, relief, and spray nozzles to the upper head

Nozzle to safe end attachment welds

All girth and longitudinal full penetration welds

Manway attachment welds

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

Peripheral support rings are furnished for the removable insulation modules.

The pressurizer Quality Assurance Program is given in table 5.4-12.

5.4.11 PRESSURIZER RELIEF DISCHARGE SYSTEM

The pressurizer relief discharge system collects, cools, and directs, for processing, the steam and water discharged from various safety and relief valves in the containment. Additionally, letdown from the reactor vessel head vent may also be directed to the pressurizer relief discharge system. The system consists of the pressurizer relief tank, the pressurizer safety and relief valve discharge piping, the relief tank internal spray nozzles and associated piping, the tank nitrogen supply, and the drain to the liquid waste management system.

5.4.11.1 Design Basis

The system design, including the pressurizer relief tank design volume, is based on the requirement to condense and cool a discharge of steam equivalent to 110 percent of the full power pressurizer steam volume, without exceeding a pressure/temperature condition of 50 psig/200F in the pressurizer relief tank.

These values are well below the pressurizer relief tank design conditions of 100 psig and 340F. Additional design data for the tank are given in table 5.4-13.

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 120F, which corresponds to the design maximum containment temperature for normal conditions. Provision is made to permit cooling of the water in the tank when the temperature exceeds 120F during normal plant operation. Following a design basis discharge to the tank, the final operating temperature is 200F, which allows the tank to be drained to the liquid waste management system without cooling.

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

The pressurizer safety and relief valve piping and support arrangement, is designed such that the effect of thrust forces on the piping system from valve operations is minimized. The piping analysis is discussed in section 3.9.

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.

The steam and water discharged from the pressurizer safety and relief valves are routed to the pressurizer relief tank. Several liquid relief valves inside containment also discharge reactor grade water to the pressurizer relief tank. 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 pressurizer safety and relief valve piping and support arrangement shown in figure 5.4-9 shows the valve discharge piping, as well as the piping upstream of the safety and relief valves. The piping upstream of the valves, which is not considered part of the pressurizer relief discharge system, includes:

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

B. A line from the pressurizer relief nozzle branching to the three power-operated relief valves, which have individual water seals and motor-operated isolation valves.

The pressurizer safety and relief valve discharge piping consists of:

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

B. Safety valve discharge lines to the manifold

C. Relief valve discharge lines to the manifold, and

D. A manifold downcomer discharge line to the pressurizer relief tank.

The main support structure for the safety and relief valve piping consists of four column assemblies equally spaced around the pressurizer top circumference. These columns are mechanically attached to the four pairs of internal valve support brackets on the pressurizer. No support structure welding to the pressurizer is required. The tops of the four main columns are attached to the square-shaped manifold at midpoint of each side. To increase natural frequency of the system, tubular braces are attached to the main column and the manifold, which are interconnected by auxiliary cross members. The safety valves are provided with a bottom saddle-type support coupled to brace column junction. The relief valves are positioned above the manifold and the relief valve lines are supported from the manifold.

The pressurizer safety, relief valve piping, and manifold are constructed of austenitic stainless steel. The support columns and other nonpressure boundary members are structural carbon steel except supports welded to the manifold, which are austenitic stainless steel.

The piping from the pressurizer to the safety and relief valves is designed and fabricated in accordance with ASME B&PV Code, Section III, Class 1 (NB) requirements. The downstream piping from the safety and relief valves to the manifold discharge tee is considered nonnuclear safety. The support structure is designed and fabricated in accordance with ASME B&PV Code, Section III, Class 1 (NF) requirements. It should be noted that the combination of discharge piping, also acting as the structural support, is unique in its dual function and is designed and fabricated in accordance with ASME B&PV Code, Section III, Class 1 (NF) requirement. Design data for the pressurizer safety and relief valve piping are given in table 5.4-15.

The general configuration of the pressurizer relief tank is shown in figure 5.4-10. The tank is a horizontal, cylindrical vessel with elliptical dished heads. The vessel is constructed of austenitic stainless steel and is overpressure protected in accordance with ASME B&PV Code, Section Ⅷ, Division 1, by means of two safety heads with stainless steel rupture discs. Also shown in figure 5.4-10 is the flanged connection for the pressurizer safety and relief valve discharge line, the spray water inlet, the bottom drain connection, the gas vent connection, and the vessel supports. The tank is designed and fabricated to Section Ⅷ, Division 1, ASME Code.

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

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 sized such that the pressure following a design basis steam discharge does not exceed 86 psig, assuming an initial pressure of 8.5 psig. This pressure is consistent with rupture disc minimum bursting pressure. Provision is made to permit the gas in the tank to be periodically analyzed to determine the concentration of hydrogen and/or oxygen. 321

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

5.4.11.3 Safety 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 reactor coolant system (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.

Regulatory Guide 1.67 is not applicable since the system is not an open discharge system.

The pressure relief discharge system is capable of handling a design discharge of steam without exceeding the design pressure and temperature. The volume of nitrogen in the pressurizer relief tank is that required to limit the maximum pressure accompanying the design basis discharge to 86 psig, rupture disc minimum bursting pressure. The volume of water in the pressurizer relief tank is capable of absorbing the heat from the assumed discharge while maintaining the water temperature below 200F. 321

If a discharge results in a pressure that exceeds the design, the rupture discs on the tank will 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 and rupture discs holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

The discharge piping from the pressurizer safety and relief valves to the pressurized 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 following instrumentation is provided on the main control board:

A. The pressurizer relief tank pressure transmitter provides signals to an indicator and to an alarm actuated on high tank pressure.

B. The pressurizer relief tank level transmitter supplies a signal to an indicator. High- and low-level alarms are also provided.

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

D. The temperature of the safety and relief valve discharge lines is displayed by indicators. Alarms actuated by high temperature notify the operator of steam discharge due to either leakage or valve actuation.

5.4.11.5 Inspection and Testing Requirements

The nondestructive examinations performed during fabrication of the forged piping from the pressurizer to the downcomer tee connection are identified in table 5.4-16. The pressurizer relief tank is subject to nondestructive and hydrostatic testing during construction and after installation in accordance with Section VIII, of the ASME B&PV Code.

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

5.4.12 VALVES

5.4.12.1 Design Bases

As noted in section 5.2, all reactor coolant pressure boundary valves out to and including the second valve normally closed or capable of automatic or remote closure, larger than 3/4-inch, are American Nuclear Society (ANS) Safety Class 1, and ASME B&PV Code, Section III, Class 1 valves. All 3/4-inch or smaller valves in lines connected to the Class 1 reactor coolant system (RCS) piping are Class 2, since the interface with the Class 1 piping is provided with suitable flow limiting orificing for such valves. Design data for the RCS valves are given in table 5.4-17.

To ensure that the valves meet the design objectives, the materials of construction minimize corrosion/erosion and are compatible with the environment. Leakage is minimized to the extent practicable by design.

5.4.12.2 Design Description

All RCS valves are constructed primarily of stainless steel. Other materials in contact with the coolant are corrosion resistant.

All manual and motor-operated RCS valves which are 3 inches and larger are provided with double-packed stuffing boxes and intermediate lantern-ring leakoff connections. All throttling control valves, are provided with double-packed stuffing boxes and with stem leakoff connections. In general, RCS leakoff connections are piped to a closed collection system. Leakage to the atmosphere is essentially zero for these valves.

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

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 disc hinge is serviced through the bonnet. All operating parts are contained within the body. The disc has limited rotation to provide a change of seating surface and alignment after each valve opening.

5.4.12.3 Design Evaluation

The design 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 demonstrate the ability of the valves to operate as required during anticipated and postulated plant conditions.

Reactor coolant chemistry parameters are specified in the design specifications to ensure the compatibility of valve construction materials with the reactor coolant. To ensure that the reactor coolant continues to meet these parameters, the chemical composition of the coolant will be analyzed periodically.

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

5.4.12.4 Tests and Inspections

The tests and inspections discussed in paragraph 3.9.3.2 are performed to ensure the operability of active valves.

There are no full-penetration welds within valve body walls. Valves are accessible for disassembly and internal visual inspection to the extent practical. Plant layout configurations determine the degree of inspectability. The valve nondestructive examination program is given in table 5.4-18. Inservice inspection is discussed in subsection 5.2.4.

5.4.13 SAFETY AND RELIEF VALVES

5.4.13.1 Design Bases

The combined capacity of the pressurizer safety valves is designed to accommodate the maximum surge resulting from complete loss of load. This objective is met without reactor trip or any operator action by the opening of the steam generator safety valves when steam pressure reaches the steamside 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 power loss.

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

5.4.13.2 Design Description

The pressurizer safety valves are 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 drains in the bottom of the pressurizer. If the pressurizer pressure exceeds the set pressure 372
of the safety valves, they start lifting during the accumulation period.

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

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 through the associated valve.

Leakage through the pressurizer safety and relief valves can also be detected by the acoustic leak monitoring system. A localized leak through a pressure boundary or a valve seat generates metalborne acoustic waves which are detected by acoustic transmitters mounted on the piping adjacent to the valves.

The power-operated relief valves provide a safety-related means for reactor coolant system depressurization to achieve cold shutdown. For a discussion of the use of these valves to achieve safety-grade cold shutdown, see subsection 5.4.7.

Design parameters for the pressurizer safety and power-operated relief valves are given in table 5.4-19.

5.4.13.3 Design Evaluation

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

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

5.4.13.4 Tests and Inspections

All safety and relief valves are subjected to hydrostatic tests, seat leakage tests, operational tests, and inspections as required. For safety and relief valves that are required to function during a faulted condition, additional tests are performed. These tests are described in section 3.9.

The valve nondestructive examination program is given in table 5.4-18. There are no full-penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection.

5.4.14 COMPONENT SUPPORTS

5.4.14.1 Design Bases

Component supports allow virtually unrestrained lateral thermal movement of the loop during plant operation and provide restraint to the loops and components during accident conditions. The loading combinations and design stress limits are discussed in subsection 3.9.1. Support design is in accordance with the ASME Code, Section III, Subsection NF. The design maintains the integrity of the reactor coolant system boundary for normal and accident conditions and satisfies the requirements of the piping code.

5.4.14.2 Description

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

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

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

Reactor Pressure Vessel

Supports for the reactor vessel, as shown in figure 5.4-11, are individual air-cooled rectangular box structures beneath the vessel nozzles welded to an embedment in the primary shield wall. Each box structure consists of a horizontal top plate that receives loads from the reactor vessel shoe, a horizontal bottom plate supported by and transferring loads to the primary shield wall and connecting vertical plates. The supports are air-cooled to maintain the supporting concrete temperature within acceptable levels.

Steam Generator

As shown in figure 5.4-12, the steam generator supports will consist of the following elements:

A. Vertical Support

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

B. Lower Lateral Support

Lateral support is provided at the generator tubesheet by fabricated steel girders and struts. These are bolted to the compartment walls and include bumpers that bear against the steam generator but permit unrestrained movement of the steam generator during changes in system temperature. The beam is attached to the walls such that any displacement of the walls due to compartment pressurization causes no additional stresses in the beam.

C. Upper Lateral Support

Lateral support of the steam generator is provided by a ring band at the operating deck. Two-way acting snubbers restrain sudden seismic or blowdown induced motion, but permit the normal thermal movement of the steam generator. Movement perpendicular to the thermal growth direction of the steam generator is prevented by lateral keyed supports.

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 will be provided by three lateral tension tie bars. The pump supports are shown in figure 5.4-13.

Pressurizer

The supports for the pressurizer, as shown in figure 5.4-14, consist of:

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

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

CRDM Support System

The support system for the control rod drive mechanisms (CRDM) provides lateral restraint to limit CRDM deflections due to seismic or pipe break loadings. The CRDM support system consists of the following:

A. Square plates at the top of the rod position indicator coil on each CRDM assembly. Square plates are also placed in locations where there are no CRDM assemblies to complete the square plate array.

B. A support platform which surrounds the square plate array. The support platform is constructed as a box frame welded to a girder ring at the outside periphery.

C. Three columns pinned to the reactor vessel head (reactor vessel head lifting legs). The support platform is held vertically by the lifting legs.

D. Radial and tangential tension tie rods, pinned to the refueling canal wall, provide lateral restraint for the support platform.

The CRDM support system is shown in figure 5.4-15.

Pipe Restraints

A. Crossover Leg Restraints

Restraint at each elbow of the reactor coolant pipe between the pump and the steam generator (crossover leg) is required to prevent excessive stresses on the system resulting from postulated breaks in this pipe. The restraint includes pipe bumpers with straps and steel thrust blocks as shown in figure 5.4-16. A whip restraint is provided to prevent whipping of the crossover leg pipe following a postulated break at the steam generator outlet nozzle. This restraint is attached to the vertical run of the crossover leg pipe and extends horizontally to the supporting concrete structure as shown in figure 5.4-17.

B. Hot Leg Restraint

A restraint is located at the 50 degree elbow in the hot leg 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 as shown in figure 5.4-18.

C. Primary Shield Wall Restraints

Pipe restraints are provided in the primary shield wall at the reactor coolant pipe as close to the elbow on the cold leg as possible, and on the hot leg as close to the RPV outlet nozzle safe end as possible. The function of these restraints is to limit the break area for guillotine breaks at the RPV safe end such that cavity pressures are limited. These restraints are shown in figure 5.4-19.

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 acceptibility. 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 loss of coolant accident conditions. Loads which the system is expected to encounter often during its lifetime (thermal, weight, pressure) are applied and stresses are compared to allowable values. The modeling and analysis methods are discussed in subsection 3.9.1.

The reactor vessel supports are not designed to provide restraint to vertical uplift movement resulting from seismic and pipe break loadings. However, reactor pressure vessel motion resulting from seismic and pipe break events is included in the reactor coolant system analyses described in subsection 3.9.1.

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 lll, Subsection NF and in the applicable equipment specifications.

5.4.15 REFERENCES

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

2. In-Service Inspection of Pressurizer Water Reactor Steam Generator Tubes, USNRC Regulatory Guide 1.83, Revision 1, July 1975. 321

3. Based for Plugging Degraded PWR Steam Generator Tubes (for comment), USNRC Regulatory Guide 1.121, Aug 1976.

Table 5.4-1

REACTOR COOLANT PUMP DESIGN PARAMETERS
(Sheet 1 of 2)

Unit Design Pressure, psig	2485
Unit Design Temperature, °F	650[a]
Unit Overall Height, ft	27.5
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Cooling Water Flow, gpm per RCP	516
Maximum continuous Cooling Water Water Inlet Temperature, °F	105
<u>Pump</u>	
Capacity, gpm	102,900
Developed Head, ft	279
NPSH Required, ft	Figure 5.4-2
suction Temperature, °F	556.7
Pump Discharge Nozzle, Inside Diameter, in.	27-1/2
Pump Suction Nozzle, Inside Diameter, in.	31
Speed, rpm	1188

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

Table 5.4-1

REACTOR COOLANT PUMP DESIGN PARAMETERS
(Sheet 2 of 2)

<u>Pump (continued)</u>	
Water Volume, ft^3	80[(b)]
Weight (dry), lbs	207,000
<u>Motor</u>	
Type	Drip proof, squirrel cage induction, water/ air cooled, totally enclosed
Power, Hp	7000
Voltage, volts	13,200
Phase	3
Frequency, Hz	60
Insulation Class	Class F, thermalastic epoxy insulation
Current	
Starting	1900 amp @ 13,200 volts
Input, hot reactor coolant	271 amp
Input, cold reactor coolant	338 amp
Pump Moment of Inertia, $lb\text{-}ft^2$ maximum	
Flywheel	70,000
Motor	22,500
Shaft	520
Impeller	980

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b. Composed of reactor coolant in the casing and of injection and cooling water in the thermal barrier.

Table 5.4-2

REACTOR COOLANT PUMP QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]
Castings	yes		yes	
Forgings				
1. Main Shaft		yes	yes	
2. Main Studs		yes	yes	(PT or MT)
3. Flywheel (Rolled Plate)		yes		(PT or MT of finished base and keyway only.)
Weldments				
1. Circumferential	yes		yes	
2. Instrument Connections			yes	

a. RT - Radiographic

UT - Ultrasonic

PT - Dye Penetrant

MT - Magnetic Particle

Table 5.4-3

STEAM GENERATOR DESIGN DATA

Design Pressure, reactor coolant side, psig	2485
Design Pressure, steam side, psig	1185
Design Pressure, primary to secondary, psi	1600	321
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	5626
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	0.00006
Steam Flow, lbs/hr	4.31×10^6	321

Table 5.4-4

STEAM GENERATOR QUALITY ASSURANCE PROGRAM
(Sheet 1 of 2)

	NDT Method*				
	RT	UT	PT	MT	ET
<u>Tubesheet</u>					
Forging		yes		yes	
Cladding		yes(+)	yes		
<u>Channel Head</u> <u>(if fabricated)</u>					
Fabrication	yes(++)	yes(+++)		yes	
Cladding			yes		
<u>Secondary Shell and Head</u>					
Plates		yes			
<u>Tubes</u>		yes			yes
<u>Nozzles (Forgings)</u>		yes		yes	
<u>Weldments</u>					
Shell, longitudinal	yes			yes	
Shell, circumferential	yes			yes	
Cladding (channel head-tubesheet 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	

3

Table 5.4-4

STEAM GENERATOR QUALITY ASSURANCE PROGRAM
(Sheet 2 of 2)

	NDT Method*				
	RT	UT	PT	MT	ET
Tube to tubesheet			yes		
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		

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current
(+) - Flat Surface Only
(++) - Weld Deposit
(+++) - Base Material Only

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
Reactor Coolant Loop Piping	
Design/Operating Pressure, psig	2485/2235
Design Temperature, °F	650
Pressurizer Surge Line	
Design Pressure, psig	2485
Design Temperature, °F	680
Pressurizer Safety Valve Inlet Line	
Design Pressure, psig	2485
Design Temperature, °F	680
Pressurizer (Power-Operated) Relief Valve Inlet Line	
Design Pressure, psig	2485
Design Temperature, °F	680

Table 5.4-6

REACTOR COOLANT PIPING QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]
Fittings and Pipe (Castings)	Yes		Yes
Fittings and Pipe (Forgings)		Yes	Yes
Weldments			
1. Circumferential	Yes		Yes
2. Nozzle to runpipe (Except no RT for nozzles less than 6 inches)	Yes		Yes
3. Instrument connections			Yes

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

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 1 of 14)

Component[(a)]	Failure Mode	Function[(b)]	Effect on System Operation[(b)]	Failure Detection Methods[(c)]	Remarks[(d)]
1. Motor-operated gate valve 8701A (8701B analogous)	a. Fails to open on demand	Provides isolation of fluid flow from the RCS to the suction of RHR pump 1 (pump 2)	a. Failure blocks reactor coolant flow from hot leg of RC loop 1 (loop 3) through train "A" (train "B") of RHRS. Failure 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 3 (loop 1) through train "B" (train "A") of RHRS; however, time required to reduce RCS temperature will be extended.	a. Valve open/close position indication at CB; RC loop 1 (loop 3) hot leg pressure indication at CB; RHR train "A" (train "B") discharge flow indication and low flow alarm at CB; and RHR pump 1 (pump 2) discharge pressure indication at CB.	1. Valve is electrically interlocked with RWST to RHR suction line isolation valve 8809A (8809B), with RHR to charging pump suction line isolation valve 8706A (8706B) and with a "prevent-open" pressure interlock PT403 (PT-409) of RC loop 1 (loop 3) 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 425 psig. The valve can be manually opened.
	b. Once open, fails closed due to false autoclosure interlock signal		b. Same as item 1.a.	b. Same as item 1.a.	
	c. Once open, fails to close if RCS pressure exceeds 750 psig.		c. Failure reduces redundancy of RHRS automatic isolation at RCS pressure > 750 psig. No effect on safety for system operation. Automatic isolation will be provided by valve 8702 A (8702 B)	c. Valve open/close position indication at CB.	2. Valve is electrically interlocked with "auto-close" pressure interlock PT403 (PT-409) to close if RC loop pressure exceeds 750 psig.

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 2 of 14)

Component[a]	Failure Mode	Function[b]	Effect on System Operation[b]	Failure Detection Methods[c]	Remarks[d]
2. Motor-operated gate valve 8702A (8702B analogous).	Same as item 1.	Same as item 1.	Same as item 1.	Same as item 1.	Same as item 1, except for pressure interlock PT408 (PT-402) control.
3. RHR pump 1 (RHR pump 2 analogous)	Fails to deliver working fluid	Provides fluid flow of reactor coolant through RHR heat exchanger 1 (heat exchanger 2) to reduce RCS temperature during cooldown operation.	Failure results in loss of reactor coolant flow from hot leg of RC loop 1 (loop 3) through train "A" (train "B") of RHRS. Failure 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 3 (loop 1) flowing through train "B" (train "A") 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 (loop 3) hot leg pressure indication at CB; RHR train "A" (train "B" discharge flow indication and low flow alarm at CB; and pump discharge pressure indication at CB.	The RHRS shares components with the ECCS. Pumps are tested as part of the ECCS testing program (see section 6.3.4).
4. Motor-operated gate valve FCV-602A (FCV-602B analogous)	a. Fails closed	Provides regulation of fluid flow through miniflow bypass line to suction of RHR pump 1 (pump 2) to protect against overheating of the pump and loss of discharge flow from the pump.	a. Failure blocks miniflow line to suction of RHR pump 1 (pump 2) during cooldown operation. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop 3 (loop 1) flowing through train "B" (train "A") of RHRS. However, time required to reduce RCS temperature will be extended.	a. Valve open/close position indication at CB; and RHRS train "A" (train "B") discharge flow indication at CB.	1. Valve is automatically controlled to open when pump discharge is less than 500 gpm and close when the discharge exceeds 1000 gpm.

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 3 of 14)

Component[a]	Failure Mode	Function[b]	Effect on System Operation[b]	Failure Detection Methods[c]	Remarks[d]
	b. Fails open		b. Failure allows for a portion of RHR heat exchanger 1 (heat exchanger 2) discharge flow to be bypassed to suction of RHR pump 1 (pump 2). RHRS train "A" (train "B") is degraded for the regulation of coolant temperature by RHR heat exchanger 1 (heat exchanger 2). No effect on safety for system operation. Cooldown of RCS within established specification cooldown rate may be accomplished through operator action of adjusting throttle valves HCV-603A (HCV-603B) and FCV-605A (FCV-605B) to compensate for the open miniflow line and controlling cooldown with redundant RHRS train "B" (train "A").	Same as item 4.a.	
5. Air diaphragm-operated butterfly valve FCV-605A (FCV-605B analogous)	a. Fails to open on demand for flow increase ("Auto" mode CB switch selection)	Controls rate of fluid flow bypassed around RHR heat exchanger 1 (heat exchanger 2) during cooldown operation.	a. Failure prevents coolant discharged from RHR pump 1 (pump 2) from bypassing RHR heat exchanger 1 (heat exchanger 2) resulting in mixed mean temperature of coolant flow to RCS being low. RHRS train "A" (train "B") is degraded for the regulation of control-	a. RHR pump 1 (pump 2) discharge flow temperature and RHRS train "A" (train "B") discharge to RCS cold leg flow temperature recording at CB; and RHRS train "A" (train "B") discharge to RCS cold leg flow indication at CB.	1. 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"

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS (Sheet 4 of 14)

Component(a)	Failure Mode	Function(b)	Effect on System Operation(b)	Failure Detection Methods(c)	Remarks(d)
			ling 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-603A (HCV-603B) and controlling cooldown with redundant RHRS train "B" (train "A").		to align RHRS for ECCS operation during plant power operation. 2. Valve is designed for normal plant cooldown operation. It is not required for safety-grade cold shutdown operations.
	b. Fails to close on demand for flow reduction ("Auto" mode CB switch selection)		b. Failure allows coolant discharged from RHR pump 1 (pump 2) to bypass RHR heat exchanger 1 (heat exchanger 2) resulting in mixed mean temperature of coolant flow to RCS being high. RHRS train "A" (train "B") 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-603A (HCV-603B) and controlling cooldown with redundant RHRS train "B" (train "A"); however, cooldown time will be extended.	b. Same as item 5.a.	

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 5 of 14)

Component[a]	Failure Mode	Function[b]	Effect on System Operation[b]	Failure Detection Methods[c]	Remarks[d]
6. Air diaphragm-operated butterfly valve HCV-603A (HCV-603B analogous)	a. Fails to close on demand for flow reduction	Controls rate of fluid flow through RHR heat exchanger 1 (heat exchanger 2) during cooldown operation.	a. Failure prevents control of coolant discharge flow from RHR heat exchanger 1 (heat exchanger 2) 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 RHS train "B" (train "A").	a. Same methods of detections as those stated for item 5.a. In addition, monitor light and alarm (valve closed) for group monitoring of components at CB.	1. Valve is designed to fail "open". Valve is normally "open" to align RHRS for ECCS operation during plant power operation.
	b. Fails to open on demand for flow increase		b. Same as item 6.a.	b. Same as item 6.a.	
7. Motor-operated gate valve 8809A (8809B analogous)	Fails to close on demand	Provides isolation of fluid from the RWST to suction of RHR pump 1 (pump 2) during cooldown operation	No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg loop 3 (loop 1) flowing through train "B" (train "A") of RHRS; however, time required to reduce RCS temperature will be extended.	Valve open/closed position indication at CB and valve (closed) monitor light and alarm at CB.	Valve is normally "open" to align RHRS for ECCS operation during plant power operation. Valve must be closed during plant cooldown to satisfy electrical interlock to permit valves 8701A and 8702A (8701B and 8702B) to be opened.

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 6 of 14)

Component[a]	Failure Mode	Function[b]	Effect on System Operation[b]	Failure Detection Methods[c]	Remarks[d]
8. Motor-operated gate valve 8887A (8887B analogous)	Fails to close on demand.	Provides separation between the two RHR trains during cooldown operation.	Failure reduces the redundancy for isolating RHR trains during cooldown. Negligible effect on system operation. Isolation valve 8887B (8887A) provides backup isolation between the two RHR trains.	Same as item 7.	
9. Centrifugal charging pump 1 (pump 3 analogous)	Fails to deliver working fluid	Provides fluid flow of borated water from the BAT or RWST to the RCS.	Failure reduces redundancy of providing borated water to the RCS at high RCS pressures. Fluid flow from charging pump 1 (pump 3) will be lost. Minimum flow requirements for boration and makeup will be met by charging pump 3 (pump 1).	Charging pump discharge header pressure and flow indication at CB. Open/close pump switchgear circuit breaker indication on CB. Circuit breaker close position monitor light for group monitoring of component at CB. Common breaker trip alarm at CB.	1. The charging pumps provide boration and makeup flow to the RCS during safety grade cold shutdown operations. 2. Analysis of charging pump 2 being on-line is analogous to that presented for charging pumps 1 and 3.
10. Motor-operated gate valve LCV-115C (LCV-115E analoguous)	Fails to close on demand.	Provides isolation of fluid discharge from the VCT to the suction of charging pumps.	Failure reduces redundancy of providing VCT discharge isolation. Negligible effect on safety for system operation. Alternate isolation valve LCV-115E (LCV-115C) provides backup tank discharge isolation.	Same as item 7.	The charging pumps' suction is isolated from the VCT and aligned to the BAT (for boration) or RWST (for makeup) during safety-grade cold shutdown operations.
11. Motor-operated gate valve LCV-115B (LCV-115D	Fails to open on demand.	Provides isolation of fluid discharge from the RWST to the suction of	Failure reduces redundancy of providing fluid flow from RWST to suction of charging pumps. Negligible	Valve open/close position indication at CB and valve (open) monitor light	The charging pumps' suction is aligned to the RWST for makeup to the

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 7 of 14)

Component[(a)]	Failure Mode	Function[(b)]	Effect on System Operation[(b)]	Failure Detection Methods[(c)]	Remarks[(d)]
analogous)		charging pumps.	effect on safety for system operation. Alternate isolation valve LCV-115D (LCV-115B) opens to provide backup flow path to suction of charging pumps.	and alarm at CB.	RCS during safety-grade cold shutdown operations.
12. Motor-operated gate valve 8107 (8108 analogous)	Fails to close on demand.	Provides isolation of fluid flow from the charging pump discharge header to the CVCS normal charging line to the RCS.	Failure reduces redundancy of providing isolation of charging pump discharge to normal charging line of CVCS. Negligible effect on safety for system operation. Alternate isolation valve 8108 (8107) provides backup normal CVCS charging line isolation.	Same as item 7 except no valve (closed) monitor alarm for group monitoring.	Normal charging line is isolated during safety-grade cold shutdown operations. Boration and makeup flow provided to RCS through redundant ECCS headers to the RCS cold legs.
13. Motor-operated gate valve 8130A (8130B analogous)	Fails to close on demand	Provides isolation barrier to isolate charging pump suction flow paths in the event of a MELB in charging pump suction header.	No negligible effect on safety for system operation. MELB isolation is provided by closing of alternate isolation valve 8130B (8130A).	Same as item 7.	
14. Motor-operated gate valve 8131A (8131B analogous)	Fails to close on demand	Same as item 13.	Negligible effect on safety for system operation. MELB isolation is provided by closing of alternate isolation valve 8131B (8131A)	Same as item 7.	
15. Motor-operated gate valve 8132A	Fails to close on demand	Provides isolation barrier to isolate	Negligible effect on safety for system operation. HELB isolation	Same as item 7.	

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 8 of 14)

Component(a)	Failure Mode	Function(b)	Effect on System Operation(b)	Failure Detection Methods(c)	Remarks(d)
(8132B analogous)		charging pump discharge flow paths in the event of a HELB in charging pump discharge header.	is provided by closing of alternate isolation valve 8132B (8132A).		
16. Motor-operated gate valve 8133A (8133B analogous)	Fails to close on demand.	Same as item 15.	Negligible effect on safety for system operation. HELB isolation is provided by closing of alternate isolation valve 8133B (8133A).	Same as item 7.	
17. Motor-operated gate valve 8803A	Fails to close on demand.	Provides isolation of fluid flow from charging pump discharge header to the inlet of the BIT.	Failure reduces redundancy of providing isolation of fluid flow from charging pumps to the BIT. Negligible effect on safety for system operation. Alternate isolation valve 8892 can be closed to isolate BIT and permit control of boration flow via valve HCV-937B	Same as item 11.	Valve opens upon receipt of an SI "S" signal.
18. Motor-operated gate valve 8803B	Fails to close on demand	Provides isolation of fluid flow from charging pump discharge header to the inlet of the BIT.	Failure prevents isolation of fluid flow from charging pumps to the BIT and control of boration flow via valve HCV-937B. Negligible effect on safety for system operation. Charging pumps can be used to provide boration flow through redundant boration path via valve HCV-937A.	Same as item 11.	Same as item 17.

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 9 of 14)

Component[a]	Failure Mode	Function[b]	Effect on System Operation[b]	Failure Detection Methods[c]	Remarks[d]
19. Solenoid-operated globe valve HCV-937A (HCV-937B)	Fails to open on demand.	Provides control of fluid flow from charging pump 1 (pump 3) to RCS during plant boration and makeup.	Failure reduces redundancy of controlling boration and makeup flow to the RCS. Negligible effect on safety for system operation. Alternate control valve HCV-937B (HCV-937A) controls flow from charging pump 3 (pump 1).	Valve position indication at CB; and charging pump 1 (pump 3) discharge header flow indication at CB.	Same as item 12.
20. Motor-operated globe valve 8891	Fails to open on demand.	Provides isolation of fluid flow from charging pump discharge header to RCS through valve HCV-937A	Failure reduces redundancy of providing boration flow to the RCS. Negligible effect on safety for system operation. Boration flow provided by charging pump 3 through valve HCV-937B.	Same as item 11.	
21. Motor-operated gate valve 8892	Fails to close on demand.	Provides isolation of fluid flow from charging pump discharge header to the inlet of the BIT.	Failure reduces redundancy of providing isolation of fluid flow from charging pump to the BIT. Negligible effect on safety for system operation. Either isolation valve 8803A can be closed to isolate BIT and permit control of boration flow through valve HCV-937B or boration flow can be provided by charging pump 1 through valve HCV-937A.	Same as item 11.	
22. Solenoid-operated globe	a. Fails to open on demand.	Provides isolation of fluid	a. Failure reduces redundancy of providing flow from	Valve open/close position indication at CB; and	The RV head letdown path to the PRT

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 10 of 14)

Component[a]	Failure Mode	Function[b]	Effect on System Operation[b]	Failure Detection Methods[c]	Remarks[d]
valve 8037A (8037B analogous)		flow from the RV head to the PRT.	the RV head to the PRT. Negligible effect on safety for system operation. RV head letdown flow provided by parallel head letdown path through alternate isolation valve 8037B (8037A).	RV head letdown high temperature indication and alarm at CB.	provides fluid flow out of the RCS to accommodate boration flow into the RCS.
	b. Fails to close on demand.		b. Failure reduces redundancy of isolating flow from the RV head to the PRT. Negligible effect on safety for system operation. RV head letdown flow isolation provided by alternate series isolation valve 8038A (8038B).		
23. Solenoid-operated globe valve 8038A (8038B analogous)	a. Fails to open on demand.	Same as item 22.	a. Same as item 22.a. except for alternate isolation valve 8038B (8038A).	Same as item 22.	Same as item 22.
	b. Fails to close on demand.		b. Same as item 22.b. except for alternate series isolation valve 8037A (8037B).		
24. Solenoid-operated globe valve HCV-443A (HCV-443B analogous)	Fails to open on demand.	Same as item 22.	Same as item 22.a except for alternate parallel isolation valve HCV-443B (HCV-443A).	Valve position indication at CB; RV letdown temperature indication at CB.	Same as item 22.

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 11 of 14)

Component[(a)]	Failure Mode	Function[(b)]	Effect on System Operation[(b)]	Failure Detection Methods[(c)]	Remarks[(d)]
25. Solenoid-operated power-operated relief valve PCV-445A (PCV-445B analogous)	a. Fails to open on demand. b. Fails to close on demand.	Provides isolation of fluid flow from pressurizer to PRT.	a. Failure reduces redundancy of providing flow from pressurizer to PRT. Negligible effect on safety for system operation. Pressurizer vent flow provided by a parallel pressurizer vent path through alternate isolation valve PCV-445B (PCV-445A). b. Failure reduces redundancy of isolating flow from the pressurizer to the PRT. Negligible effect on safety for system operation. Pressurizer vent flow isolation provided by series isolation valve 8000A (8000C).	Valve open/close position indication at CB; Pressurizer power-operated relief valve outlet temperature indication at CB.	Pressurizer vent path to the PRT provides fluid flow out of the RCS to permit RCS depressurization to RHRS initiation conditions.
26. Motor-operated gate valve 8000A (8000C analogous)	Fails to close on demand.	Same as item 25.	Same as item 25.b except pressurizer vent flow isolation provided by series isolation valve PCV-445A (PCV-445B).	Same as item 25.	Same as item 25.
27. Motor-operated gate valve 8808A (8808B and 8808C analogous)	Fails to close on demand.	Provides isolation of fluid flow from accumulator 1 (accumulator 2 and accumulator 3) to the RCS.	Failure prevents isolation of accumulator 1 (accumulator 2 and accumulator 3) from the RCS. Negligible effect on safety for system operation. Accumulator 1 (accumulator 2 or accumulator 3) is depressurized	Valve open/closed position indication at CB, valve (closed) monitor light and alarm at CB; and accumulator pressure indication and low alarm at CB.	Accumulators are isolated or vented during plant cooldown so as not to affect RCS depressurization

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 12 of 14)

Component[a]	Failure Mode	Function[b]	Effect on System Operation[b]	Failure Detection Methods[c]	Remarks[d]
			by opening vent isolation valves 8875A (8875B or 8875C) and HCV-936 or vent isolation valves 8876A (8876B or 8876C) and 8882.		to RHRS initiation conditions.
28. Solenoid-operated globe valve 8875A (8875B and 8875C analogous)	Fails to open on demand.	Provides venting of nitrogen gas from accumulator 1, (accumulator 2 and accumulator 3) to containment.	Failure reduces redundancy for venting accumulator 1 (accumulator 2 and accumulator 3) to containment. No effect on safety for system operation. Accumulator 1 (accumulator 2 and accumulator 3) can be vented by opening vent valves 8876A (8876B and 8876C) and 8882 or isolated from the RCS by closing isolation valve 8808A (8808B and 8808C).	Valve open/closed position indication at CB and accumulator pressure indication and low alarm at CB.	Same as item 27.
29. Solenoid-operated globe valve 8876A (8876B and 8876C analogous)	Fails to open on demand.	Same as item 28.	Failure reduces redundancy for venting accumulator 1 (accumulator 2 and accumulator 3) to containment. No effect on safety for system operation. Accumulator 1 (accumulator 2 and accumulator 3) to can be vented by opening vent valves 8875A (8875B and 8875C) and HCV-936 or isolated from the RCS by closing isolation valve 8808A (8808B and 8808C).	Same as item 28.	Same as item 27.

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN
OPERATION - FAILURE MODES AND EFFECTS ANALYSIS
(Sheet 13 of 14)

Component[a]	Failure Mode	Function[b]	Effect on System Operation[b]	Failure Detection Methods[c]	Remarks[d]
30. Solenoid-operated globe valve HCV-936	Fails to open on demand.	Provides venting of nitrogen gas from accumulators to containment.	Failure reduces redundancy for venting accumulators to containment. No effect on safety for system operation. Accumulators can be vented by opening vent valve 8882 or isolated from RCS by closing isolation valves 8808A, B, and C.	Valve position indication at CB and accumulator pressure indication and low alarm at CB.	Same as item 27.
31. Solenoid-operated globe valve 8882	Fails to open on demand.	Same as item 30.	Same as item 29 except for alternate vent valve HCV-936.	Same as item 30 except for alternate vent valve HCV-936.	Same as item 27.
32. Boric acid transfer pump 1 (pump 2 analogous)	Fails to deliver working fluid	Provides fluid flow of concentrated boric acid from BAT to charging pump suction.	Failure reduces redundancy of providing concentrated boric acid to charging pump suction. Fluid flow from boric acid transfer pump 1 (pump 2) will be lost. Minimum flow requirements for boration will be met by boric acid transfer pump 2 (pump 1).	Pump motor start relay position indication (open) at CB and local pump discharge pressure indication PI-110 (PI-105)	The boric acid transfer pumps provide boration flow to the charging pumps' suction during safety-grade cold shutdown operations.
33. Motor-operated globe valve 8104B (8104A analogous)	Fails to open on demand.	Provides isolation of fluid flow from either boric acid transfer pump to charging pump suction	Failure reduces redundancy of providing concentrated boric acid to charging pump suction. Negligible effect on safety for system operation. Concentrated boric acid provided to charging pump suction through alternate isolation valve 8104A (8104B).	Valve open/close position indication at CB; and boration flow indication (FI-110) at CB.	The charging pumps' suction is aligned to the BAT pumps for boration of the RCS during safety-grade cold shutdown operations.

Table 5.4-7

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY-GRADE COLD SHUTDOWN OPERATION - FAILURE MODES AND EFFECTS ANALYSIS (Sheet 14 of 14)

a. Components 7, 8, 17 through 21 and 27 through 31 are components of the ECCS that perform a safety-grade cold shutdown function. Components 9 through 16, 32, and 33 are components of the CVCS that perform a safety-grade cold shutdown function. Components 22 through 26 are components of the RCS that perform a safety-grade cold shutdown function.

b. List of acronyms and abbreviations

Auto	- Automatic
BAT	- Boric acid tank
BIT	- Boron injection tank
CB	- Control board
CVCS	- Chemical and volume control system
ECCS	- Emergency core cooling system
HELB	- High energy line break
MELB	- Moderate energy line break
PRT	- Pressurizer relief tank
RC	- Reactor coolant
RCS	- Reactor coolant system
RHR	- Residual heat removal
RHRS	- Residual heat removal system
RWST	- Refueling water storage tank
RV	- Reactor vessel
VCT	- Volume control tank

c. 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.

Table 5.4-8

DESIGN BASES FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION

Residual heat removal system initiation time after shutdown, hours	4	
Reactor coolant system initial pressure, psig	400	
Reactor coolant system initial temperature, °F	350	
Component cooling water temperature, °F	105	321
Cooldown time, after initiation of residual heat removal system operation, hours	~16	
Reactor coolant system temperature at end of cooldown, °F	140	
Decay heat generation at 20 hours after reactor shutdown, Btu/h	64.2×10^6	321

Table 5.4-9

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pump		
Number	2	
Design pressure, psig	600	
Design temperature, °F	400	
Design flow, gal/min	3000	
Design head, ft	270	
NPSH required at 4500 gal/min, ft	20	
Power, hp	300	
Residual Heat Exchanger		
Number	2	
Design heat removal capacity, Btu/h	31.2×10^6	
Estimated UA, Btu/h °F	2.0×10^6	
	Tube side	Shell side
Design pressure, psig	600	150
Design temperature, °F	400	200
Design flow, lb/h	1.5×10^6	2.6×10^6
Inlet temperature, °F	140	105
Outlet temperature, °F	119.2	117.2
Material	Austenitic stainless steel	Carbon steel
Fluid	Reactor coolant	Component cooling water

Table 5.4-10

PRESSURIZER DESIGN DATA

Design pressure, psig	2485
Design temperature, °F	680
Surge line nozzle diameter, in.	14
Heatup rate of pressurizer using heaters only, °F/hr	55
Internal volume, ft^3	1400
Span volume*, ft^3	1253

* : Volume of pressurizer level span 321

Table 5.4-11

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS (psig)

(See technical specifications for limiting values)

	Psig
Hydrostatic test pressure	3107
Design pressure	2485
Safety valves (begin to open)	2485
High pressure reactor trip	2385
High pressure alarm	2310
Power-operated relief valves	2335[a]
Pressurizer spray valves (full open)	2310
Pressurizer spray valves (begin to open)	2260
Proportional heaters (begin to operate)	2250
Operating pressure	2235
Proportional heater (full operation)	2220
Backup heaters on	2210
Low pressure alarm	2210
Pressurizer relief valve interlock	2185[b]
Low pressure reactor trip (typical, but variable)	1970

a. At 2335 psig, a pressure signal initiates actuation (opening) of these valves. Remote-manual control is also provided.

b. This interlock closes the power-operated relief valves and their block valves when pressure drops to indicated value.

Table 5.4-12

PRESSURIZER QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]
Heads				
1. Plates		yes		
2. Cladding			yes	
Shell				
1. Plates		yes		
2. Cladding			yes	
Heaters				
1. Tubing		yes	yes+	
2. Centering of element	yes			
Nozzle (Forgings)		yes	yes[b]	yes[b]
Weldments				
1. Shell, longitudinal	yes			yes
2. Shell, circumferential	yes			yes
3. Cladding			yes	
4. Nozzle safe end (if forging)	yes		yes	
5. Instrument connection			yes	
6. Support skirt, long seam	yes			yes
7. Support skirt to lower head		yes		yes
8. Temporary attachments (after removal)				yes
9. All external pressure boundary welds after shop hydrostatic test				yes

a. RT = Radiographic, UT = Ultrasonic, PT = Dye Penetrant, MT = Magnetic Particle, $\pm$ or eddy current testing

b. MT or PT

Table 5.4-13

PRESSURIZER RELIEF TANK DESIGN DATA

Design pressure, psig	100
Initial operating pressure, psig	3 (for low level alarm setpoint) 8.5 (for high level alarm setpoint)
Final operating pressure, psig	50 (for low level alarm setpoint) 86 (for high level alarm setpoint)
Design temperature, °F	340
Initial operating water temperature, °F[a]	120
Final operating water temperature, °F[a]	200
cooling time required following maximum discharge, approximate hour(s)	1 (feed & bleed process) 8 (recirculation through the reactor coolant drain tank heat exchanger)
Volume, ft^3	1,300
Initial operating water volume, ft^3	900
Initial operating gas volume, ft^3	400
Rupture disc release pressure:	
Nominal, psig	91
Range, psig	86 - 100
Total rupture disc relief capacity, lb/hr at 100 psig	1.14×10^6

382

a. For the design basis pressurizer steam discharge.

Table 5.4-14

RELIEF VALVE DISCHARGES TO THE PRESSURIZER RELIEF TANK

	Figure
<u>Reactor Coolant System</u>	
3 pressurizer safety valves	5.1-1
3 pressurizer power-operated relief valves	5.1-1
1 reactor vessel head vent letdown line	5.1-1
<u>Residual Heat Removal System</u>	
2 residual heat removal pump suction lines from the reactor coolant system hot legs	5.4-6
<u>Chemical and Volume Control System</u>	
1 seal water return line	9.3-7
1 letdown line	9.3-7

Table 5.4-15

PRESSURIZER SAFETY AND RELIEF VALVE PIPING DESIGN DATA

Safety and relief valve inlet line	
Design pressure, psig	2,485
Design temperature, °F	680
Safety and relief valve discharge piping	
Design pressure, psig	500
Design temperature, °F	470
Common discharge manifold nominal pipe size (inches)/schedule	12/100
Safety valve inlet lines and discharge piping nominal pipe size (inches)/schedule	6/160
Relief valve inlet line nominal pipe size (inches)/schedule	3/160
Relief valve discharge line nominal pipe size (inches)/schedule	6/160

Table 5.4-16

PRESSURIZER RELIEF DISCHARGE SYSTEM NONDESTRUCTIVE TESTING PROGRAM

	Radio-graphic	Ultra-sonic	Dye Penetrant
1. Fittings and pipe (casting)	Yes		Yes
2. Fittings and pipe (forgings)		Yes*	
3. Weldments			
a. Circumferential	Yes		Yes
b. Nozzle to runpipe	Yes*		Yes
c. Instrument connections			Yes

*Not applicable for PRT

Table 5.4-17

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

Table 5.4-18

REACTOR COOLANT SYSTEM VALVES NONDESTRUCTIVE EXAMINATION PROGRAM

	RT[a]	UT[a]	PT[a]
Castings (valve bodies, bonnets and discs)			
Larger than 4 inches	Yes (b)		Yes
2 inches to 4 inches	Yes (b)		Yes
Forgings (valve bodies, bonnets and discs)		Yes[c]	Yes[c]
2 inches to 4 inches		Yes	

a. RT = Radiograph, UT = Ultrasonic, PT = Dye Penetrant

b. Weld ends only

c. Either UT or PT

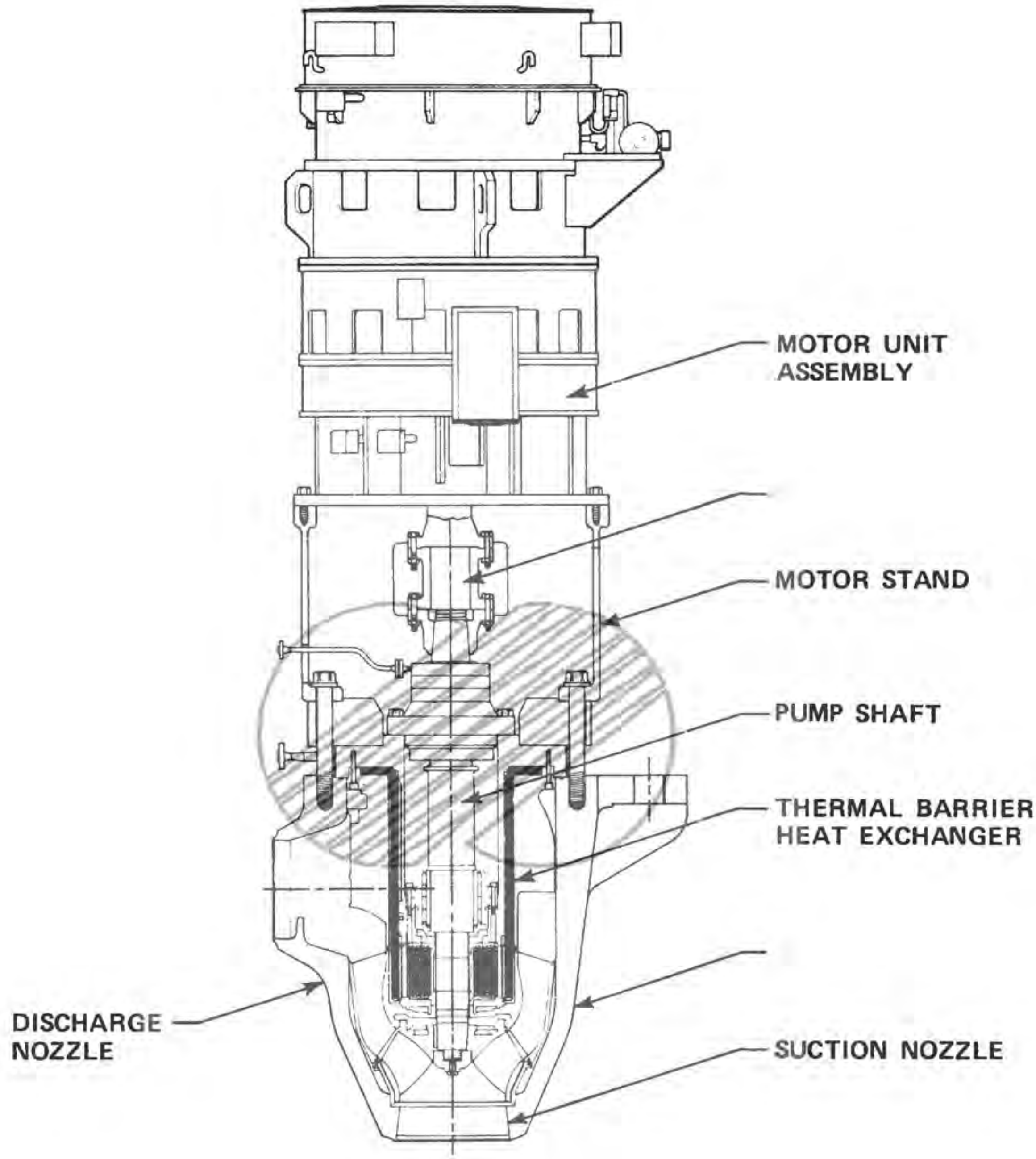
Table 5.4-19

PRESSURIZER SAFETY AND RELIEF VALVE DESIGN PARAMETERS

Pressurizer Safety Valves	
Number	3
Relieving capacity, ASME rated flow, lb/hr	380,000
Set pressure, psig	2,485
Design temperature, °F	650
Fluid	Saturated steam
Backpressure:	
Normal, psig	3 to 5
Expected during discharge, psig	350
Pressurizer Power-Operated Relief Valves	
Number	3
Design pressure, psig	2,485
Design temperature, °F	650
Relieving capacity at 2,350 psia, lb/hr	210,000
Fluid	Saturated steam

321

1880F-28



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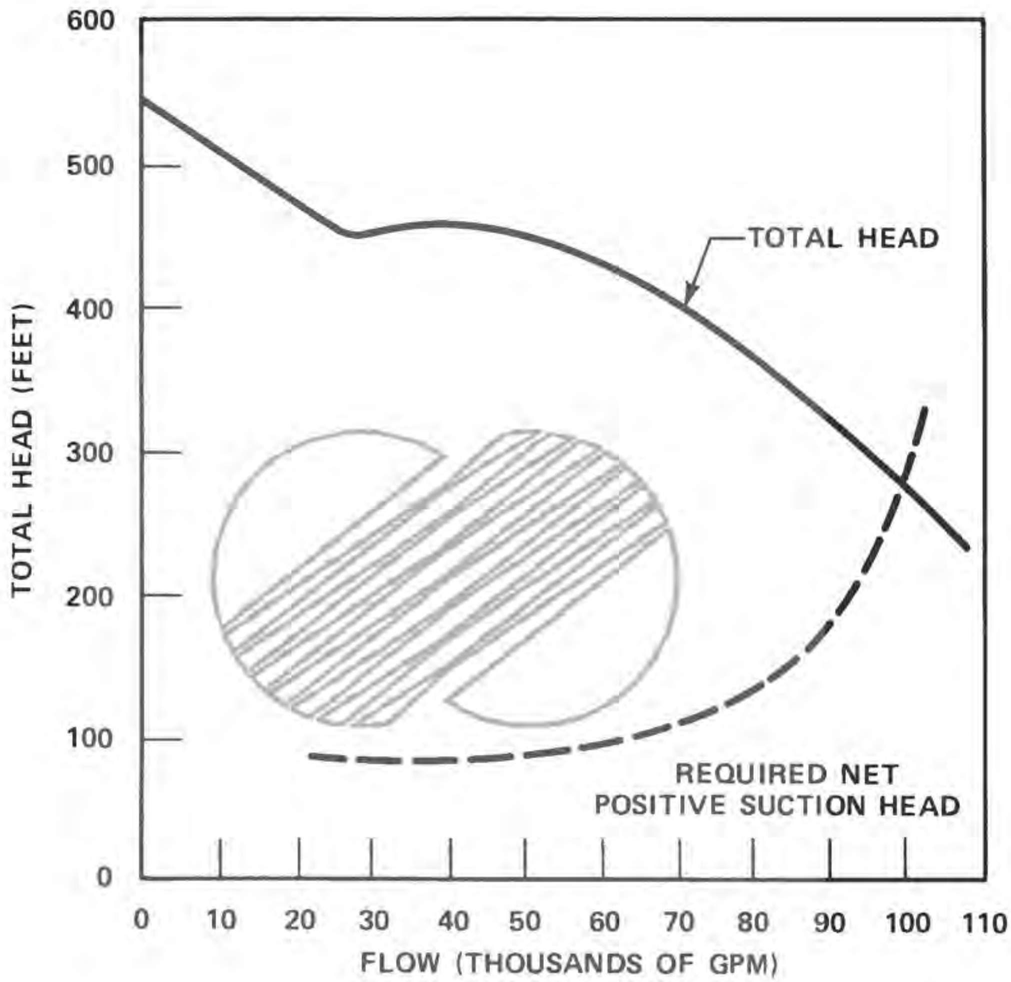
REACTOR COOLANT PUMP

Figure 5.4-1



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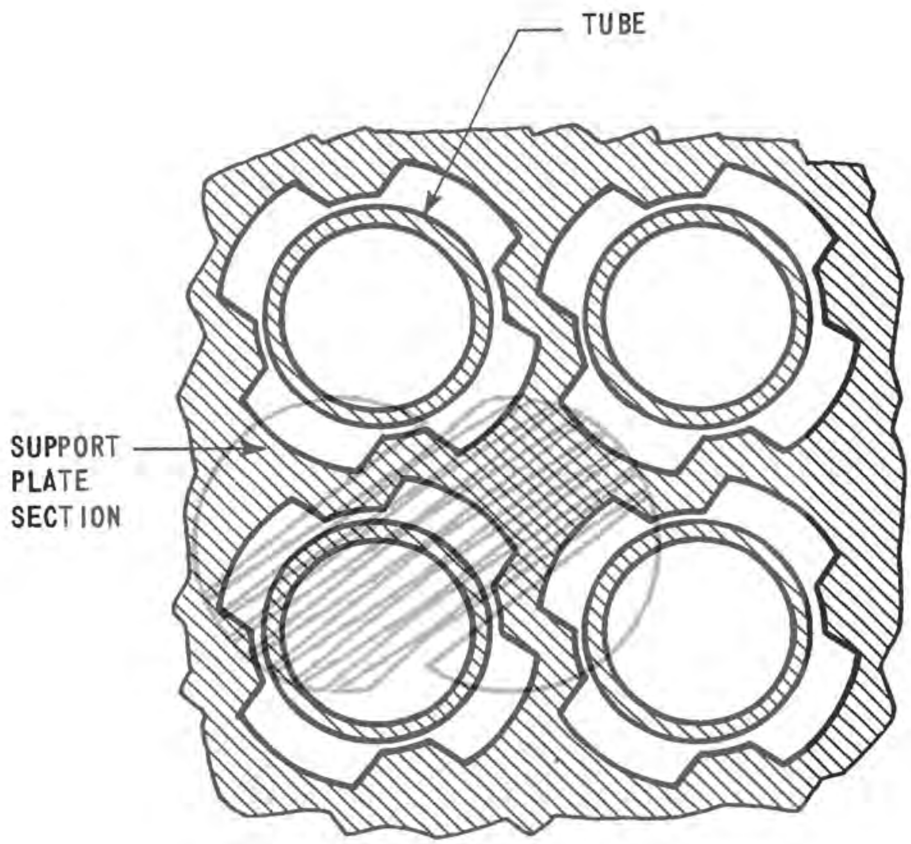
REACTOR COOLANT PUMP
ESTIMATED PERFORMANCE CHARACTERISTIC

Figure 5.4-2



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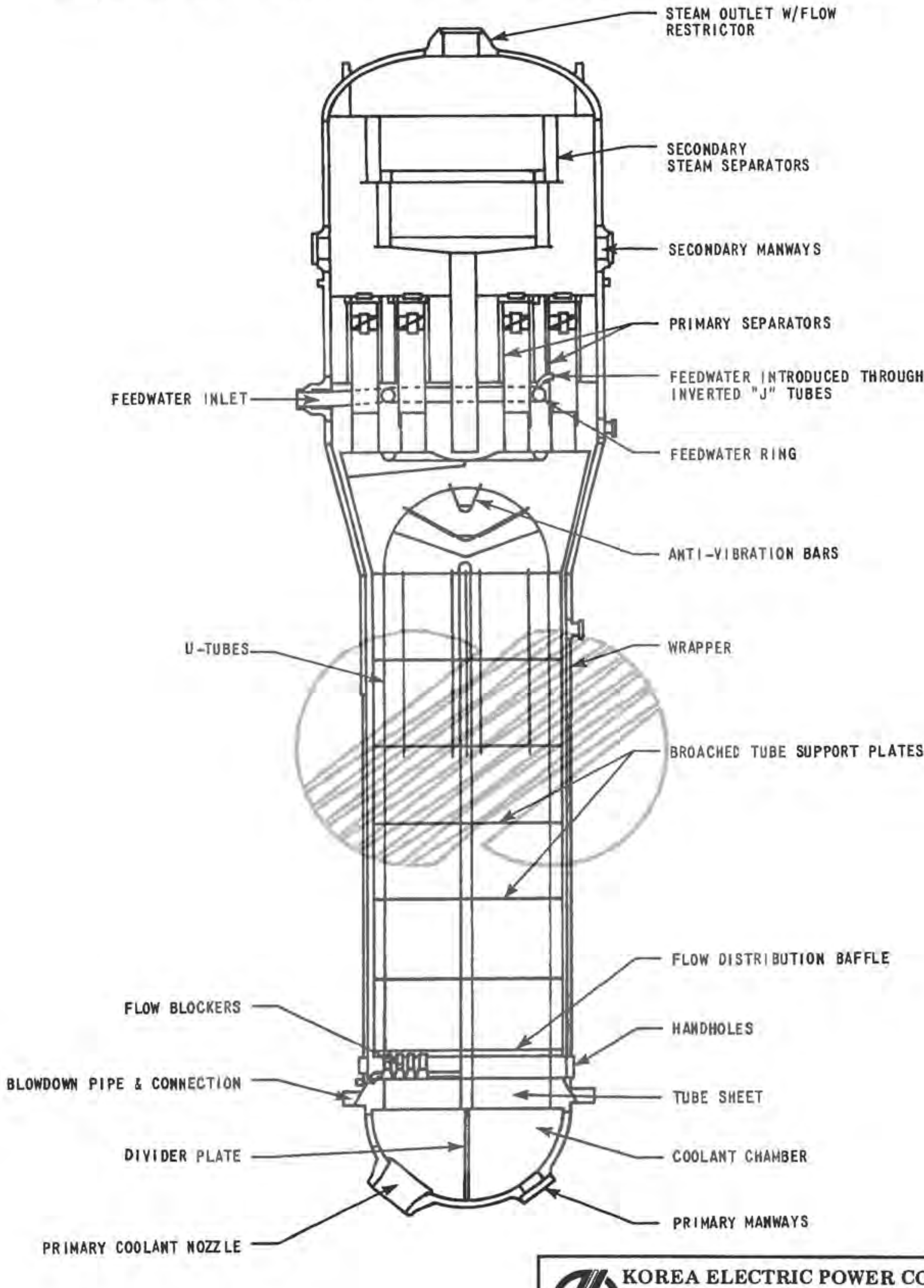
QUATREFOIL TUBE SUPPORT
DESIGN (CONCEPT)

Figure 5.4-3

1880B



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STEAM GENERATOR

Figure 5.4-4

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RESIDUAL HEAT REMOVAL SYSTEM
(PIPING INSTRUMENTATION DIAGRAM)
FIGURE 5.4-5

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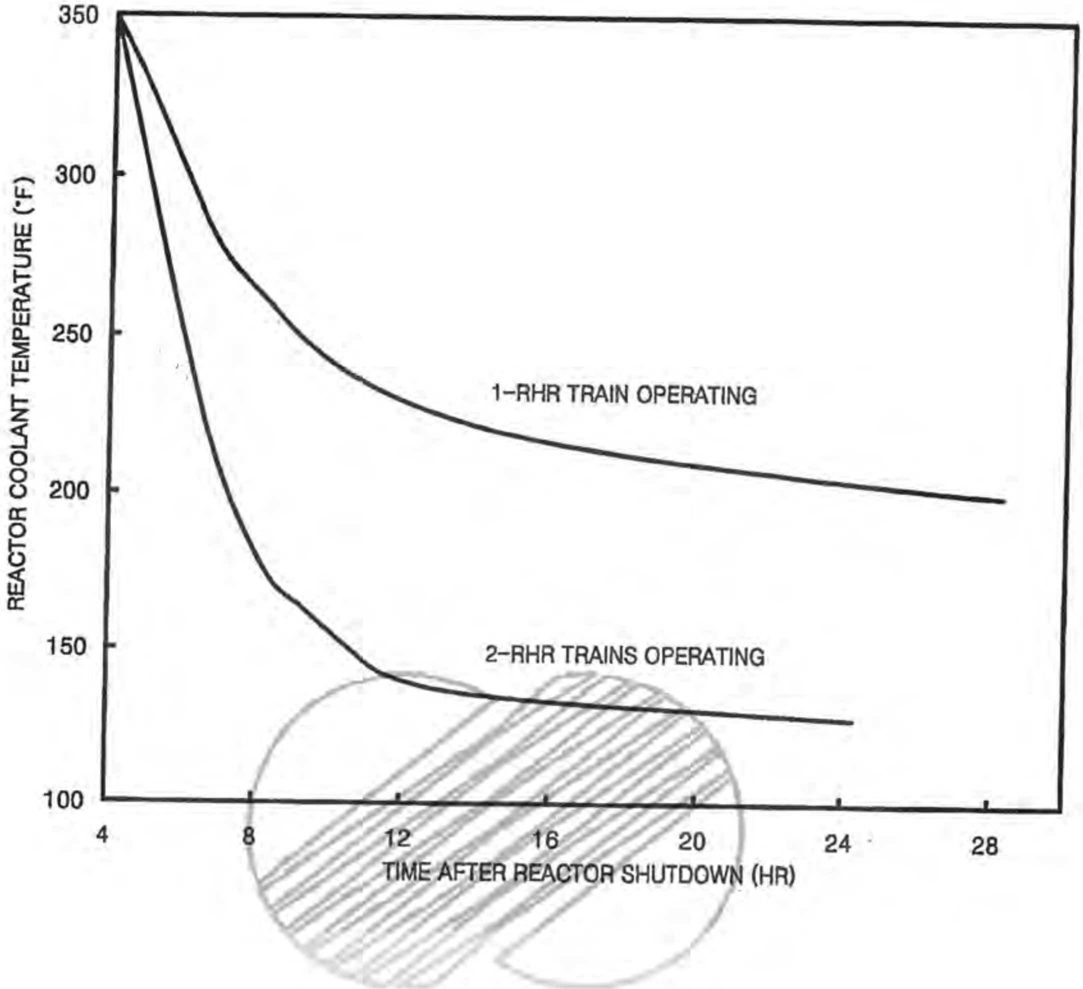
RESIDUAL HEAT REMOVAL SYSTEM
PROCESS FLOW DIAGRAM
Figure 5.4-6



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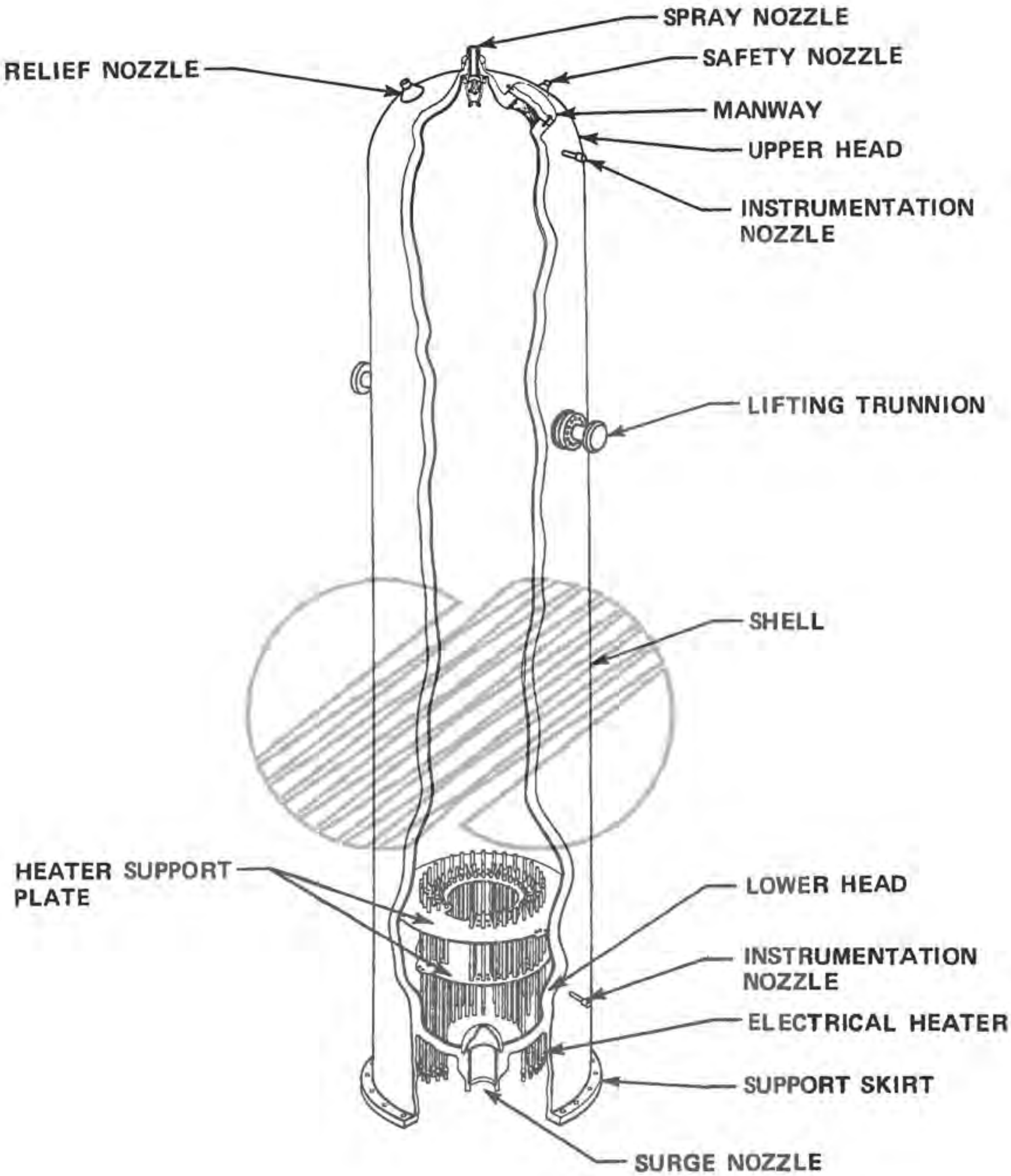
RESIDUAL HEAT REMOVAL COOLDOWN CURVES (1- TRAIN AND 2-TRAIN) Figure 5.4-7



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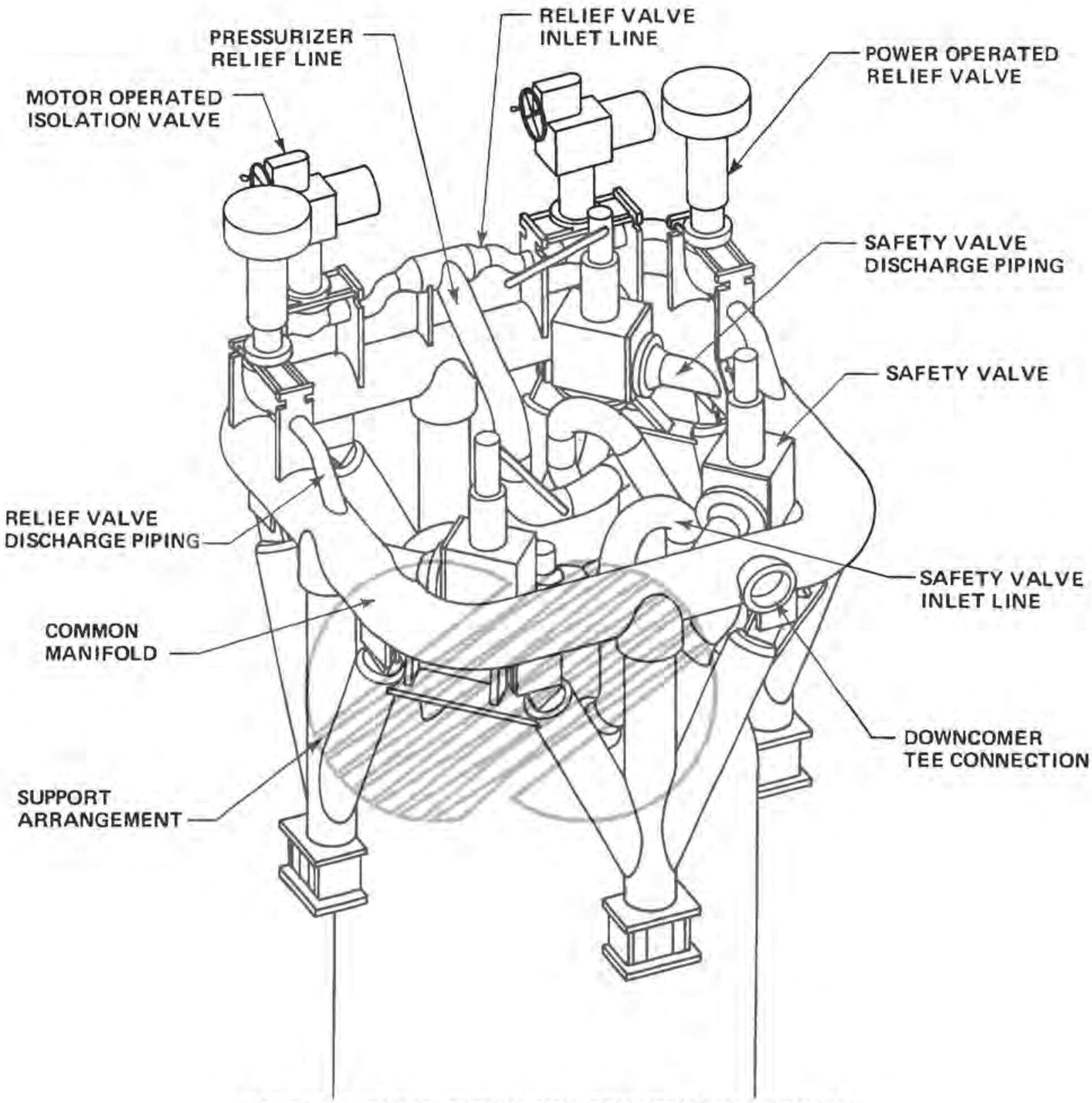
PRESSURIZER

Figure 5.4-8

1880B



NOTE: THE THIRD PORV IS NOT DEPICTED FOR CLARITY

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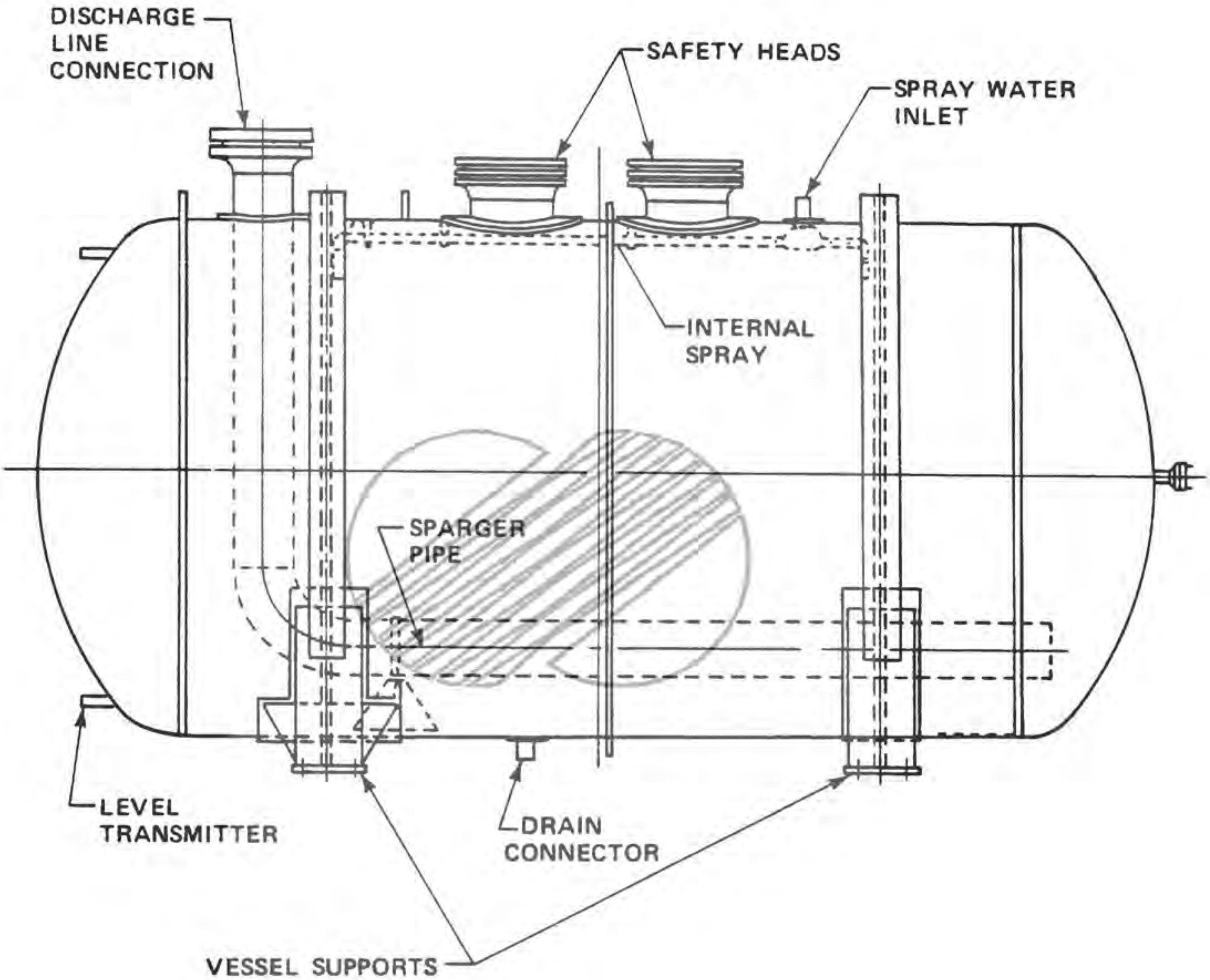
PRESSURIZER SAFETY AND RELIEF VALVE PIPING AND SUPPORT ARRANGEMENT

Figure 5.4-9



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PRESSURIZER RELIEF TANK

Figure 5.4-10

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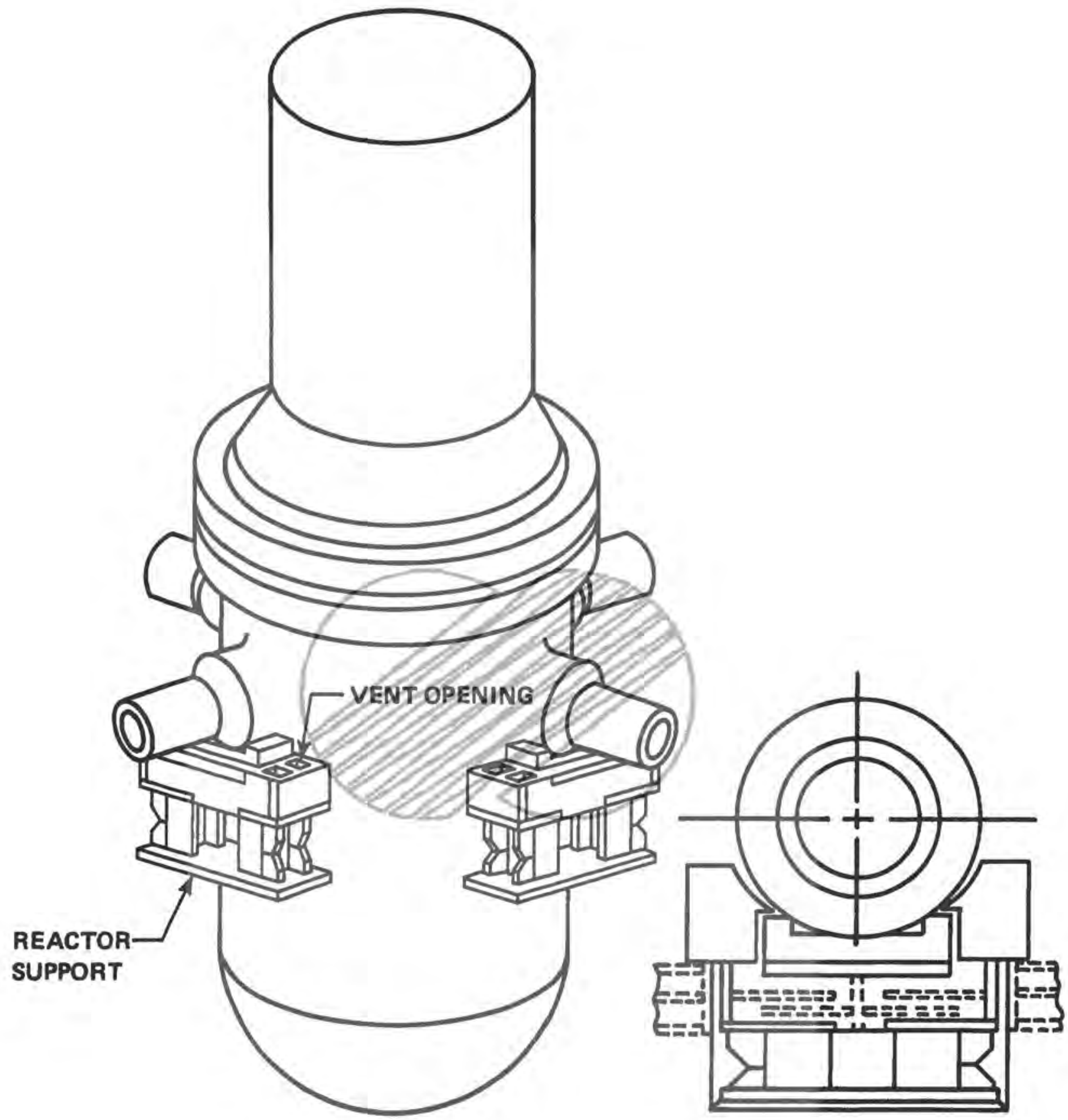

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REACTOR VESSEL SUPPORTS

Figure 5.4-11

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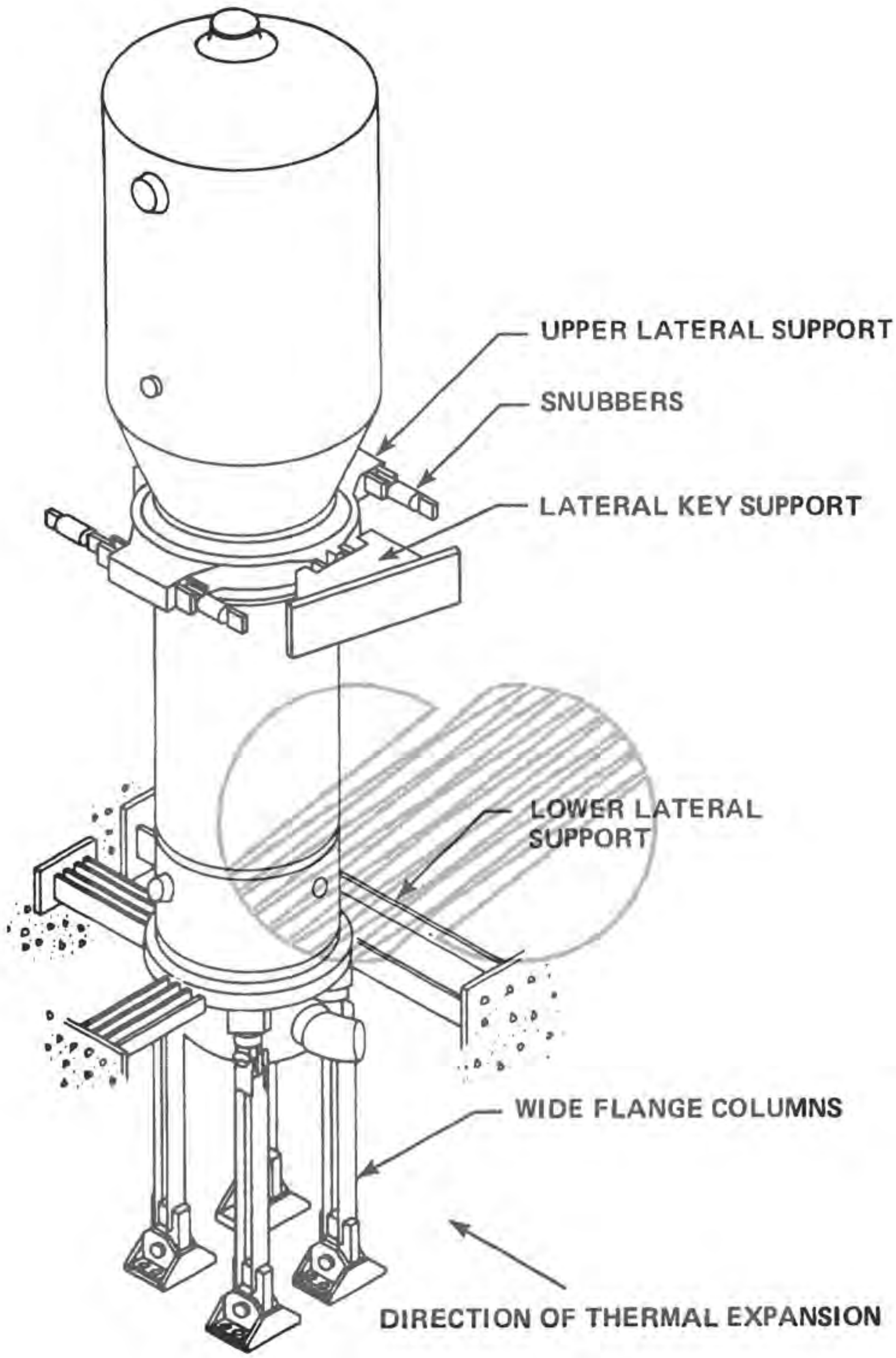

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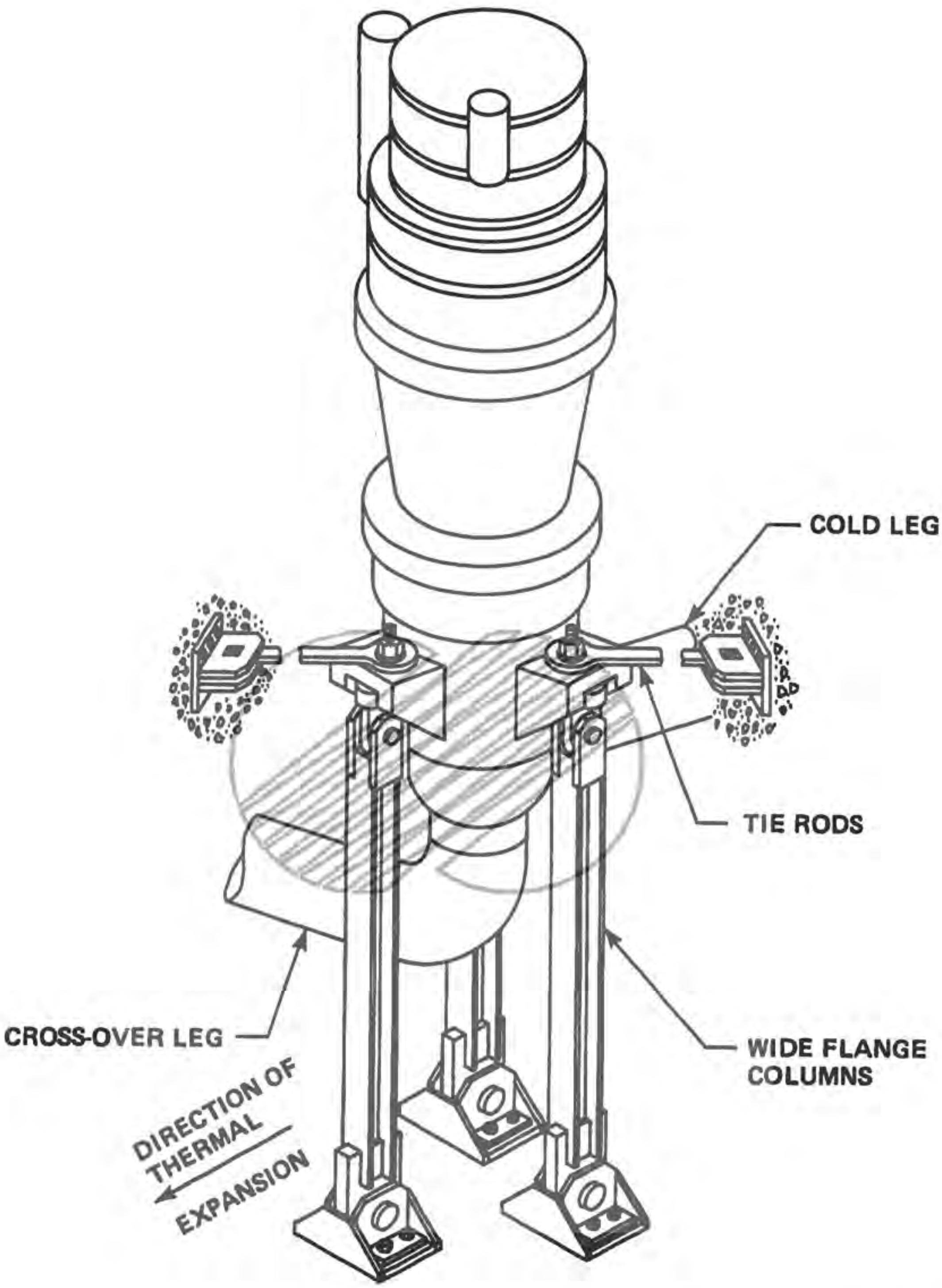
STEAM GENERATOR SUPPORTS

Figure 5.4-12

1880P



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REACTOR COOLANT PUMP SUPPORTS

Figure 5.4-13

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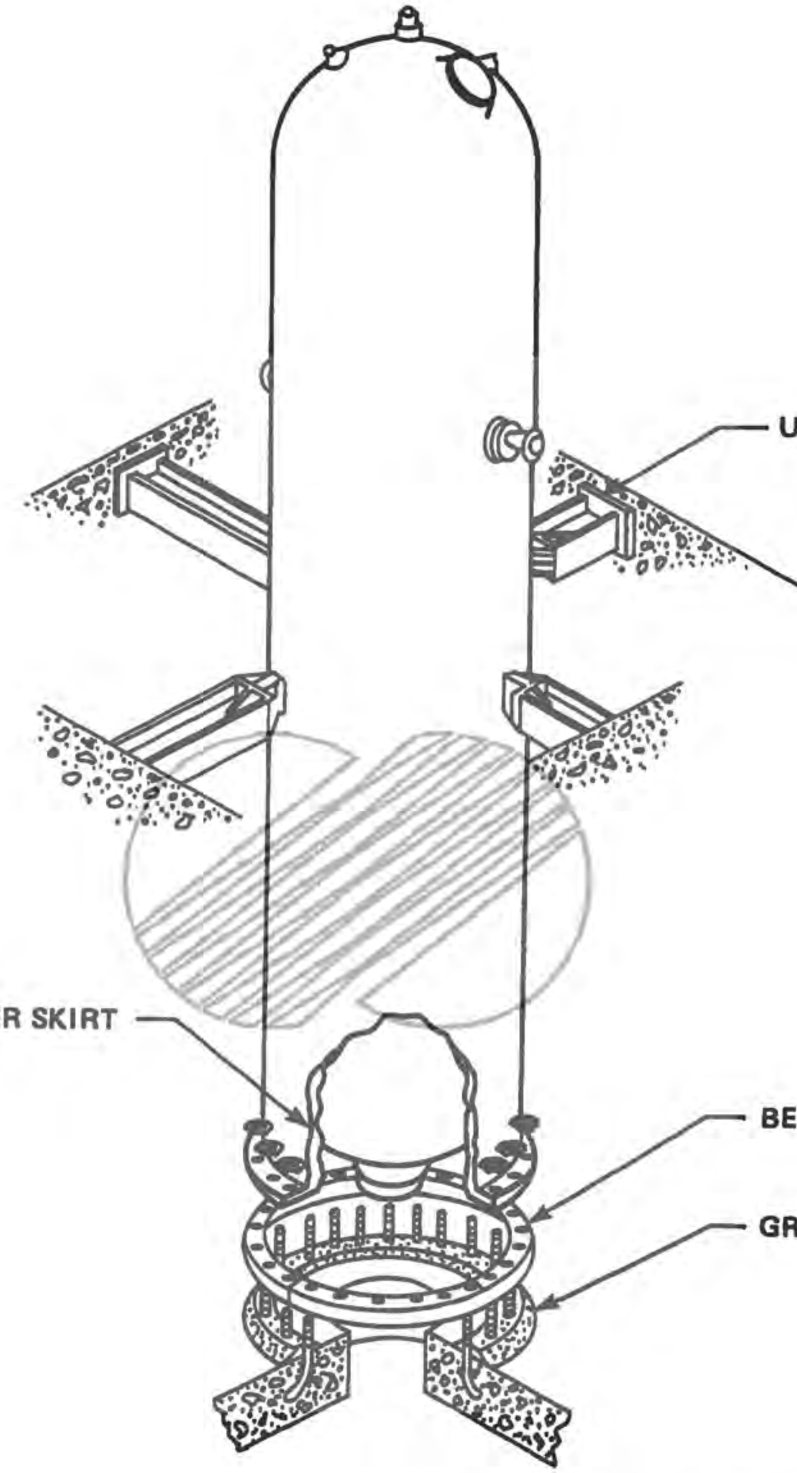

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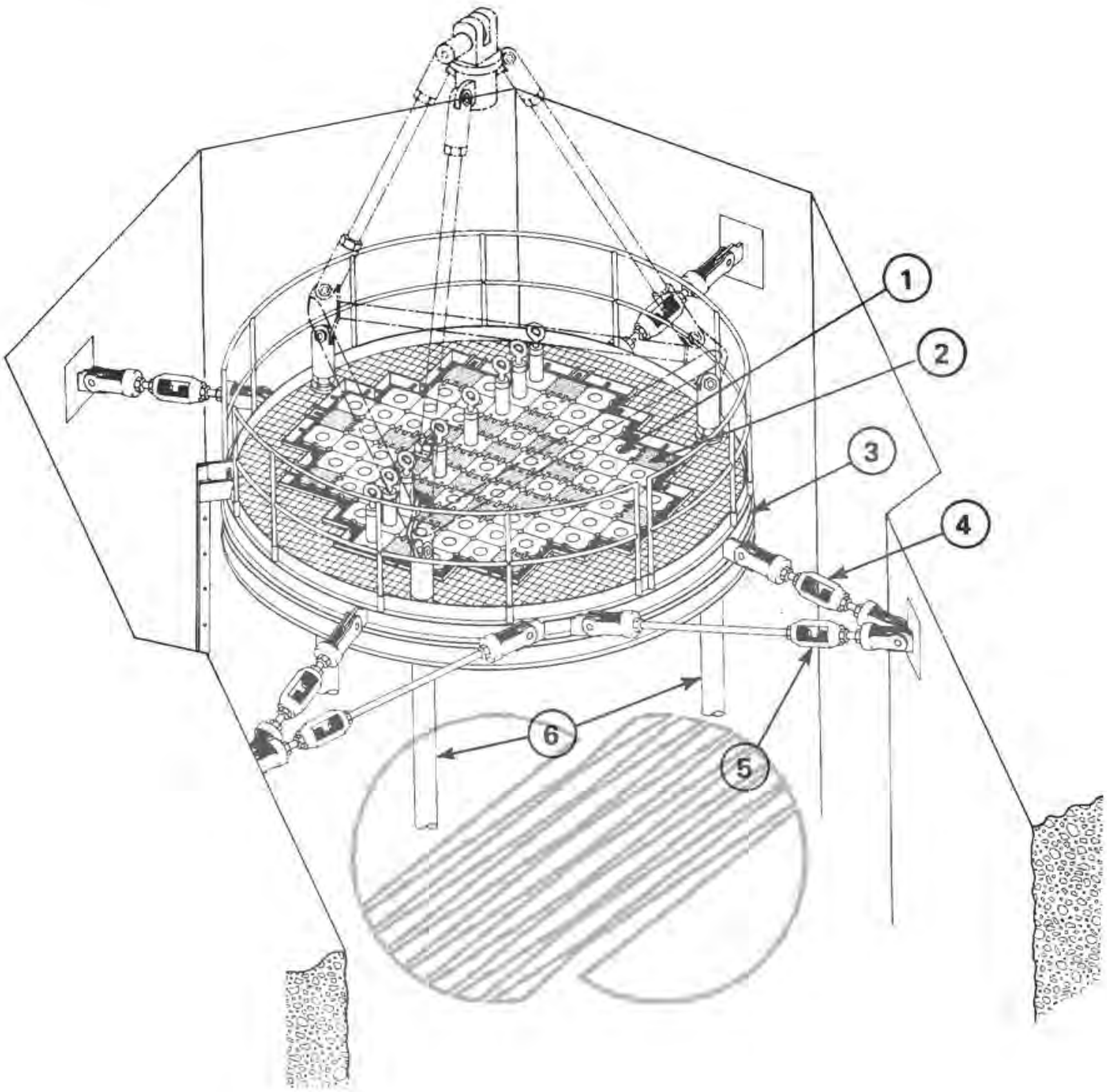
PRESSURIZER SUPPORTS

Figure 5.4-14

1880B



ITEM 1: ROD POSITION INDICATOR SQUARE PLATE
ITEM 2: SPACER PLATE
ITEM 3: SEISMIC SUPPORT PLATFORM
ITEM 4: RADIAL TIE ROD ASSEMBLY
ITEM 5: TANGENTIAL TIE ROD ASSEMBLY
ITEM 6: REACTOR VESSEL HEAD LIFTING LEGS

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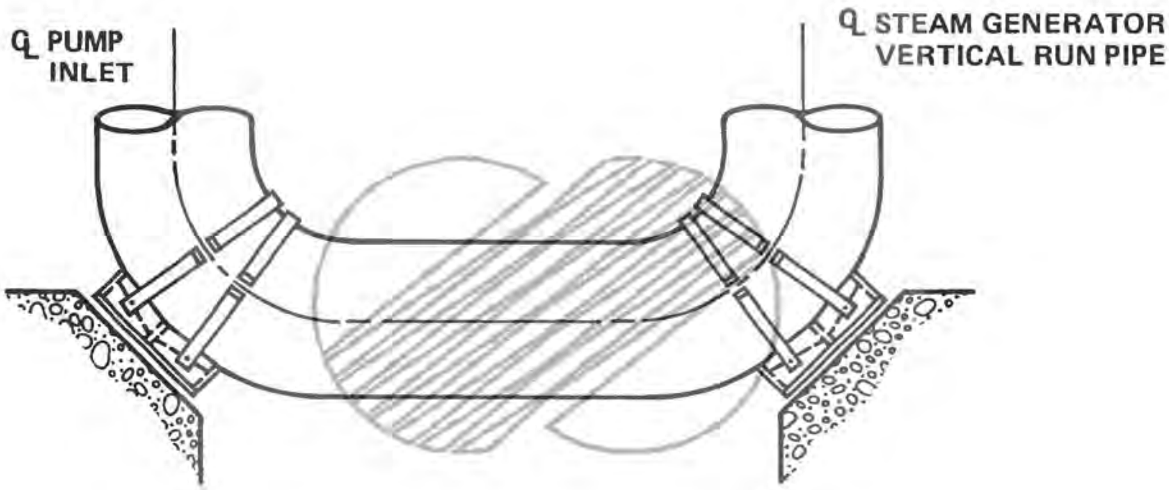
CRDM SUPPORT ASSEMBLY

Figure 5.4-15



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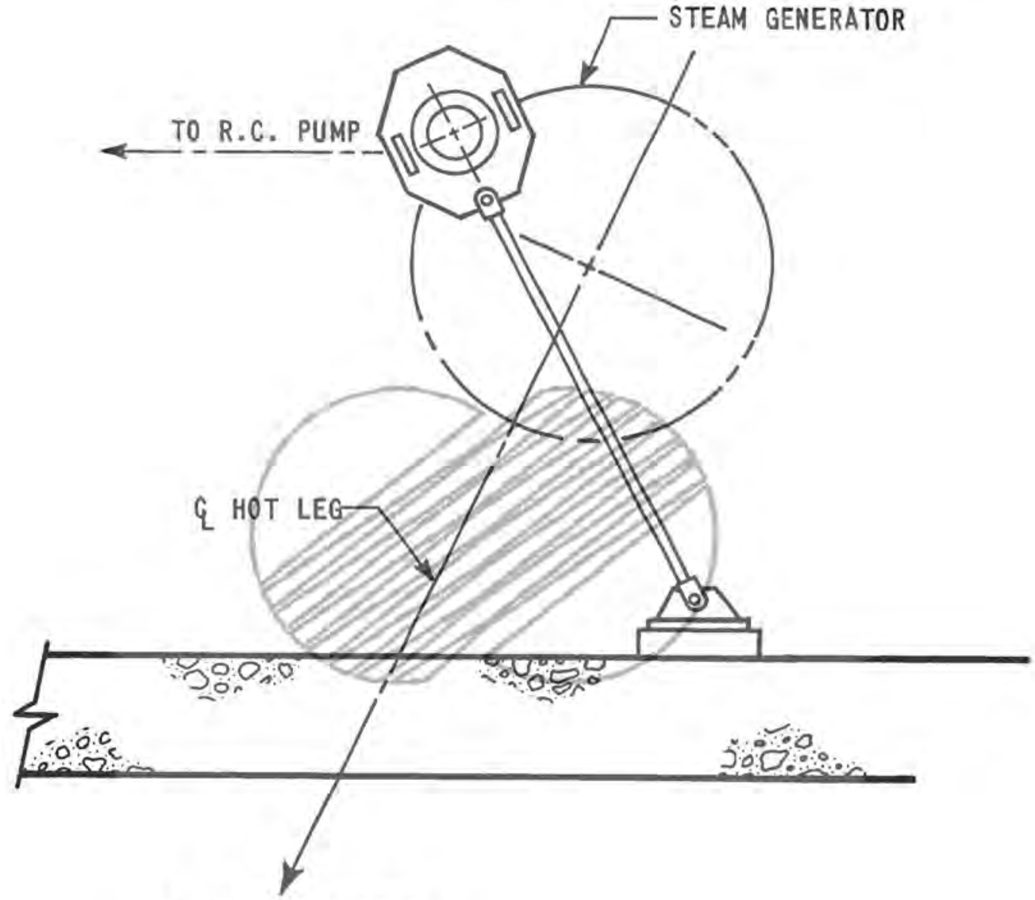
CROSSOVER LEG ELBOW RESTRAINT

Figure 5.4-16

18808



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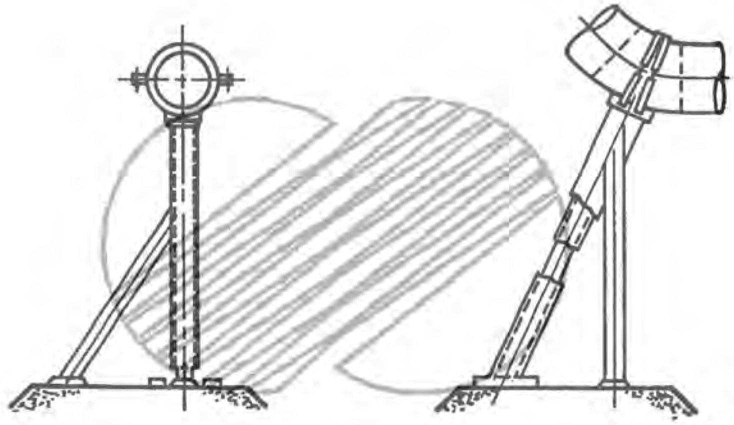
CROSSOVER LEG VERTICAL RUN RESTRAINT

Figure 5.4-17

1880B



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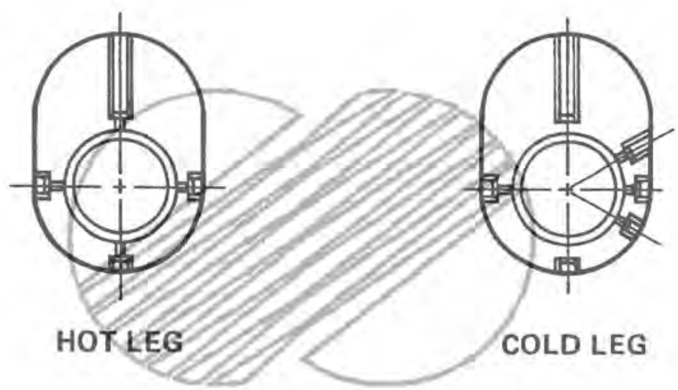
STEAM GENERATOR INLET ELBOW
(HOT LEG) RESTRAINT

Figure 5.4-18



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PRIMARY SHIELD WALL RESTRAINTS

Figure 5.4-19

APPENDIX 5A

NATURAL CIRCULATION COOLDOWN ANALYSIS

ABSTRACT

The result of a computer simulation for a natural circulation cooldown (NCC) of the KNU 5 & 6 Nuclear Steam Supply System (NSSS) from hot standby to RHR entry conditions is presented in this document. The simulation utilizes the methods and assumptions approved by the US NRC and was performed assuming the restrictions of US NRC Branch Technical Position (BTP) RSB 5-1. 321
The restrictions include the use of only safety-grade equipment, the concurrent loss of offsite power (LOOP), and a single failure. The study concludes that the KNU 5 & 6 NSSS can be cooled and depressurized to Residual Heat Removal (RHR) entry conditions in conformance with US NRC BTP RSB 5-1 requirements.

APPENDIX 5A

NATURAL CIRCULATION COOLDOWN ANALYSIS

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APPENDIX 5A

NATURAL CIRCULATION COOLDOWN ANALYSIS

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APPENDIX 5A

NATURAL CIRCULATION COOLDOWN ANALYSIS

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1.0 Introduction

1.1 Purpose

The results of a computer simulation for an NCC of the KNU 5 & 6 NSSS from normal operation to the conditions which permit the initiation of the RHR are presented in this appendix. The simulation utilizes a node and flow path network to model the Reactor Coolant System (RCS). The simulation was performed assuming the restrictions of US NRC Branch Technical Position (BTP) Reactor Systems Branch (RSB) 5-1 (Reference 1) using the methods and assumptions previously approved by the US NRC. These restrictions include the use of only safety-grade equipment, the concurrent LOOP and a single failure.

1.2 Scope

A computer simulation for a NCC of the KNU 5 & 6 NSSS has been performed using the CENTS computer code (Reference 4). A detailed node and flow path methodology is used to model the RCS. The NCC analysis from normal operation to RHR entry conditions was performed within the requirements of US NRC BTP RSB 5-1.

2.0 Analysis

2.1 General Description

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The US NRC BTP RSB 5-1 requires that the nuclear power plant demonstrate that it can be brought from the normal operation to cold shutdown under the NCC condition using only safety-grade systems with only onsite or offsite power available (not both) and assuming a single failure. In response to these requirements, an NCC analysis for KNU 5 & 6 is performed from the normal operation to RHR entry conditions.

Natural circulation is governed by decay heat, component elevations, primary to secondary heat transfer, loop flow resistance, and void formation. Component elevations in the KNU 5 & 6 are such that a satisfactory natural circulation flow for decay heat removal is obtained by fluid density differences between the core region and the Steam Generator (SG) tube region.

SG PORVs are used as a safety-grade means of the RCS cooldown and stabilization by controlling the SG secondary steam release rate for the KNU 5 & 6 NCC analysis. Each SG PORV line has two parallel flow paths with an air-operated valve in each path.

PZR PORVs are credited as a safety-grade means of the RCS depressurization because the main and auxiliary PZR spray systems are not designed as a safety-grade system for KNU 5 & 6. PZR PORV lines have three parallel flow paths with a solenoid valve in each path.

Auxiliary Feedwater System (AFWS) is also used as a safety-grade means of the SG level control to maintain the normal SG water level for NCC analysis. AFWS consists of two motor-driven pumps and one turbine-driven pump.

A safety-grade charging pump is used to control the PZR level and inject the boric acid through the normal charging bypass line or high pressure safety injection line. Class 1E motor-operated letdown isolation valve is also considered to control the PZR level as a safety-grade means for NCC analysis.

A more detailed description of the plant system status for the NCC analysis is shown in Table 5A-1.

2.2 Computer Code Modeling

The analysis results are obtained using CENTS computer code. The CENTS code is an interactive computer code for simulation of the NSSS and related systems. It calculates the behavior of a PWR for normal and abnormal conditions including accidents. It is a flexible tool for PWR analysis which gives the user complete control over the simulation through convenient input and output options. The CENTS code models most of the NSSS and related systems. Core power is computed using a point kinetics model. Boiling curves for forced convection and pool boiling are used in the multi-node core heat transfer model. Primary and secondary thermal-hydraulic behavior is calculated with detailed multi-node and flowpath models. The main control systems for reactivity, level, pressure and steam flow are simulated. A multi-node and flowpath representation of the feedwater system is provided. Related balance of plant systems for single-phase fluid are represented. 321

2.3 Assumptions and Initial Conditions

The assumptions and initial conditions for the NCC analysis are as follows:

a. In accordance with BTP RSB 5-1 requirements, the plant must be capable of completing a NCC assuming a LOOP and a concurrent single failure using only safety-grade equipment, including maintenance of the plant in hot standby condition for 4 hours. The single failure assumed is one emergency diesel generator failing to start. This disables one entire emergency power train (See Table 5A-1).

b. Main and auxiliary PZR sprays, PZR backup and proportional heaters, and control systems are assumed to be non-safety-grade and, therefore, unavailable.

c. RCS temperature and SG tube plugging are conservatively considered for the analysis.

d. The initiating event is assumed to be a LOOP at time zero resulting in an assumed loss of power to the Reactor Coolant Pumps (RCPs) thus nearly instantaneous reactor trip .

e. Following reactor trip, SG PORVs are automatically operated according to the setpoint, and then the operator manually controls SG PORVs to stabilize secondary pressure at formal no-load hot standby conditions.

f. Charging flow control valve fails open on loss of instrument air pressure, so operator action to isolate normal charging is assumed to occur at about 12 minutes. Before isolation, full charging flow from one pump is assumed.

g. Letdown isolation valve fails lock due to LOOP, and is manually isolated as the PZR level decreases after the normal charging flow isolation.

h. After 4 hour hot standby period, a cooldown is initiated with a normal cooldown rate (which is assumed to be 40 °F/hr for the NCC analysis). This cooldown rate is slower than the maximum cooldown rate of 50 °F/hr and, hence, conservatively increases the auxiliary feedwater usage.

i. The pressurizer level is controlled between 25 and 70% to avoid a solid condition or a drained condition. The operator throttles the flow of charging and letdown during the NCC as necessary to control pressurizer level and pressure (See Figure 5A-4).

j. The SG water level is conservatively assumed to be maintained at the normal operating water level. This will maximize the auxiliary feed water usage since it accounts for steady state decay heat requirements and compensates for the SG level shrink after reactor trip. 321

k. The operator manually controls the safety-grade PZR PORV instead of the auxiliary pressurizer spray system to depressurize the RCS.

l. Decay heat values used in this analysis are based on the 1971-ANS decay heat model (Reference 2). It is assumed that the uncertainties of 20% and 10% of the decay heat fractions used in the code are additionally considered from 0 to 1,000 seconds and from 1,000 to 100,000 seconds, respectively.

m. To prevent the Pressurized Thermal Shock (PTS) during NCC process, the cooldown and the depressurization are performed in maintaining the RCS subcooling margin (Reference 3).

n. It is conservatively assumed that no heat transfer occur from the Reactor Vessel Upper Head (RVUH) to the containment atmosphere via the Control Rod Drive Mechanism (CRDM), although heat loss from the RVUH to the containment via the CRDMs will exist even with the CRDM cooling fans unavailable as listed in Table 5A -1. The pressurizer heat loss to the containment is also conservatively considered for this analysis.

o. The moderator temperature coefficient, doppler reactivity feedback coefficient, scram rod worth, and boron worth are conservatively assumed for shutdown margin calculation.

p. The cooldown and depressurization to RHR entry conditions must be completed within the limitation of the available safety-grade Condensate Storage Tank (CST) capacity.

q. Once the plant reaches RHR entry conditions, the plant will subsequently be cooled to the cold shutdown using the RHR System. Natural circulation flow is not required to cool the RCS once RHR entry conditions are achieved.

2.4 Analysis Method

The NCC analysis process which is performed in accordance with the US NRC BTP RSB 5-1 requirements is divided into the following 3 phases:

a. Hot standby
b. Cooldown
c. Depressurization

Each phase listed above is controlled by the following constraints and operator action strategies:

First, during hot standby period,
a. 4 hours hot standby after reactor trip
b. SG water level at normal operating condition
c. SG pressure at no-load hot standby condition 321
d. PZR water level between 25 and 70%

Second, during a cooldown period,
a. Cooldown with a normal cooldown rate (40 "F/hr) using SG PORVs
b. PZR water level between 25 and 70%
c. RCS subcooling margin within P-T limit
d. RCS hot leg temperature less than 350 OF at the end of cooldown

Third, during a depressurization period,
a. Depressurization using PZR PORV
b. PZR water level between 25 and 70%
c. RCS pressure less than 415 psia at the end of depressurization
d. RCS subcooling margin within P-T limit

The case analyzed in this calculation consists of a series of above mentioned three phases of NCC to RHR entry conditions. The main purpose of this analysis is to maximize the auxiliary feedwater usage in order to confirm the designed CST source is enough for use. The operational status of key equipment and the key parameters pertinent to the KNU 5 & 6 NCC analysis are listed in Tables 5A-1 and 2, respectively.

2.5 Results

Immediately following the LOOP, RCPs trip and begin to coast down. The flow through the core decreases and results in reactor trip (Figure 5A-1).

Shortly after the reactor trip, SG pressure increases to the SG PORVs setpoint, and then the operator manually controls the SG PORVs to stabilize the NSSS at hot standby conditions.

Initially RCS pressure (Figure 5A-3) and PZR level (Figure 5A-4) decrease as a result of shrink following the reactor trip. As a result of the charging flow control valve failing open, the charging pump operates at full flow (Figure 5A-6). During this period, RCS pressure reaches to the PZR PORV setpoint. The operator is assumed to isolate the normal charging line 12 minutes after the event by closing one of the charging containment isolation valves from the control room. After the normal charging flow isolation, the operator isolates the letdown isolation valve as the PZR level decreases. Afterward, charging and letdown flows are manually controlled as necessary.

The initiating transient did not affect the RCP seal injection flow path. The net 15 gpm total flow for three RCPs is served as a source of RCS makeup during the entire scenario. After the normal charging line is isolated, RCS pressure and PZR level continue to rise slowly due to the seal injection flow until the operator opens the letdown isolation valve (Figure 5A-7).

The auxiliary feedwater flow to the steam generators is manually controlled to slowly refill the steam generators without overcooling the RCS. 321

The plant is maintained at hot standby for 4 hours consistent with the BTP RSB 5-1 requirements. The operator controls SG PORVs to maintain the steam generator pressure (Figure 5A-5) at 1,106 psia and thus the RCS cold leg temperature (Figure 5A-2) at 557 °F. During this hot standby period, the pressurizer pressure decreases slowly and reaches 2,113 psia at the end of 4 hours hot standby operation (Figure 5A-3).

At 4 hours, the operator begins a 40 °F/hr cooldown by increasing steam flow through SG PORVs. This increased steam flow causes a decrease in the steam generator pressure and RCS temperatures. The cooldown process also causes the pressurizer level to decrease as the RCS volume decreases. As the RCS starts to cooldown, the pressurizer level and pressure decrease, and the operator controls charging and letdown flow (Figures 5A-6 and 7) as necessary to maintain pressurizer level between 25 and 70%. The charging flow provides inventory control during the cooldown, adds negative reactivity, and increases the subcooling of the RCS. The pressurizer pressure remains relatively constant during this period because the RCS pressurization by the charging mass addition is compensated by RCS cooldown due to the secondary steam release and the injection of the cold fluid.

At 4.8 hours, the operator stops the initial cooldown as hot leg temperature reaches to less than 550 °F in order to block SI signal. The operator resumes RCS cooldown with a cooldown rate of

40 °F/hr by increasing steam flow and auxiliary feedwater flow (Figure 5A-2) at 5.8 hours, and continues the cooldown until the next hold point at 10.5 hours. During this cooldown, the operator maintains the pressurizer level within 25 - 70% by controlling charging and letdown flow (Figures 5A-6 and 7), and maintains RCS subcooling margin greater than 100 °F since the CRDM cooling fans for cooling of the upper head fluid are not credited in the analysis.

At 10.5 hours, the operator stops cooldown to prevent RCS overcooling, opens the PZR PORV to depressurize RCS to the point where the RCS subcooling margin decreases to 100 °F (Figure 5A-9) and isolates the SI accumulators to prevent discharge into the RCS

At 13.3 hours, the operator closes the PZR PORV to maintain RCS subcooling greater than 100 °F and resumes RCS cooldown with a cooldown rate of 40 °F/hr by increasing steam flow and auxiliary feedwater flow (Figure 5A-13). During the cooldown period, the operator maintains the pressurizer level within 25 - 70% by controlling charging and letdown flow (Figures 5A-6 and 7).

At 15.5 hours, the operator stops cooldown as the RCS temperature is below the RHR entry conditions with an adequate RCS subcooling margin to prevent the steam void formation in the RVUH region, and opens the PZR PORV to depressurize until the RCS pressure reaches the RHR entry condition.

At 16.2 hours, the RCS pressure and temperature reach the RHR entry conditions of 415 psia and 350 °F, respectively. 321

The RCS boron concentration increases due to the charging flow including RCP seal injection flow, which has 2,450 ppm boron concentration from RWST and 8,000 ppm boron concentration from BAT. This boron concentration increase provides additional negative reactivity into the core (Figure 5A-11) and provides additional shutdown margin. The positive reactivity insertion due to the RCS cooldown with the most negative moderator temperature coefficient is compensated by the boron concentration increase, resulting in greater than required shutdown margin throughout the transient (Figure 5A-12).

The amount of safety-grade auxiliary feedwater (Figure 5A-13) used is about 260,000 gallons. This demonstrates that the NCC to RHR entry conditions, per the BTP RSB 5-1 requirements, can be performed well within the limit of CST capacity (i.e., minimum capacity of 470,000 gallons). During the entire transient, a sufficient RCS subcooling is maintained and the steam void in the RVUH region is not formed (Figure 5A-10). This also demonstrates that a single phase subcooled natural circulation flow can be maintained with appropriate operator actions.

The analyzed results in the form of plots are shown in Figures 5A-1 through 5A-13.

3.0 Conclusions

The KNU 5 & 6 NSSS natural circulation cooldown analysis results demonstrate that cooldown and depressurization to RHR entry conditions is achievable within the BTP RSB 5-1 requirements. These requirements include the use of only safety-grade equipments, a LOOP, a single failure, and auxiliary feedwater usage within the minimum available capacity.

As an NCC analysis result of KNU 5 & 6, the amount of safety-grade auxiliary feedwater used is estimated to be about 260,000 gallons. This demonstrates that the natural circulation cooldown from normal operation to RHR entry conditions, per the BTP RSB 5-1 requirements, can be performed well within the minimum available capacity of 470,000 gallons even though the uncertainty of the decay heat is considered in this analysis. 321

It is concluded that KNU 5 & 6 can be cooled and depressurized to RHR entry conditions in conformance with the restrictive assumptions of US NRC BTP RSB 5-1.

4.0 References

1. US NRC Branch Technical Position (BTP) RSB 5-1, "Design Requirements of the Residual Heat Removal System," Rev. 2, July 1981.

2. "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," American Nuclear Society, October 1971.

3. KHNP, "Kori Units 3 & 4 Natural Circulation Cooldown Emergency Operating Procedure," Rev. 5, June 2003. 321

4. "Technical Description Manual for the CENTS Code," Rev. 0, December 2002.

5. WCAP-11810, "Shearon Harris Nuclear Plant Natural Circulation Cooldown Evaluation Program Report," Rev. 1, May 2000.

Table 5A-1 Operational Status of Key Systems and Equipment

System/Component	Status
Offsite power	Unavailable
Reactor coolant pumps	Unavailable
Pressurizer pressure control system	Unavailable
Pressurizer level control system	Unavailable
Steam dump control system	Unavailable
Main feedwater control system	Unavailable
Letdown isolation valve	Available
Charging pumps	Only one available
Main/Auxiliary spray	Unavailable
Pressurizer heaters	Unavailable
CRDM cooling fan	Unavailable
SG PORVs	Only one available per SG
PZR PORVs	Only one available
Auxiliary feedwater pumps	Only one (motor-operated) available
Seismic category 1 Condensate Storage Tank (CST)	Available at the minimum capacity (470,000 gallons)
Refueling Water Storage Tank (RWST)	Available
Boric Acid Tank (BAT)	Available
Normal charging bypass line (or High pressure safety injection line)	Available

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Table 5A-2 Key Parameters and Initial Plant Conditions

Parameters	Units	Values
Reactor power	MWt	2,912
Pressurizer pressure	psia	2,250
Cold leg temperature	°F	553.0
Hot leg temperature	°F	621.0
Steam generator pressure	psia	912
Steam generator water level	% WR	85
Steam generator tube plugging rate	%	7
No-load steam generator pressure	psia	1,106
No-load reactor inlet temperature	°F	557.0
Pressurizer level control		
Maximum control limit	%	70
Minimum control limit	%	25
Cooldown rate control		
Max. cooldown rate	°F/hr	50
Normal cooldown rate (assumed)	°F/hr	40
RHR entry conditions		
RCS pressure	psia	415
RCS maximum temperature	°F	350
Minimum CST capacity	gallons	470,000
RWST Boron concentration	ppm	2,450
BAT Boron concentration	ppm	8,000
AFW Temperature	°F	120

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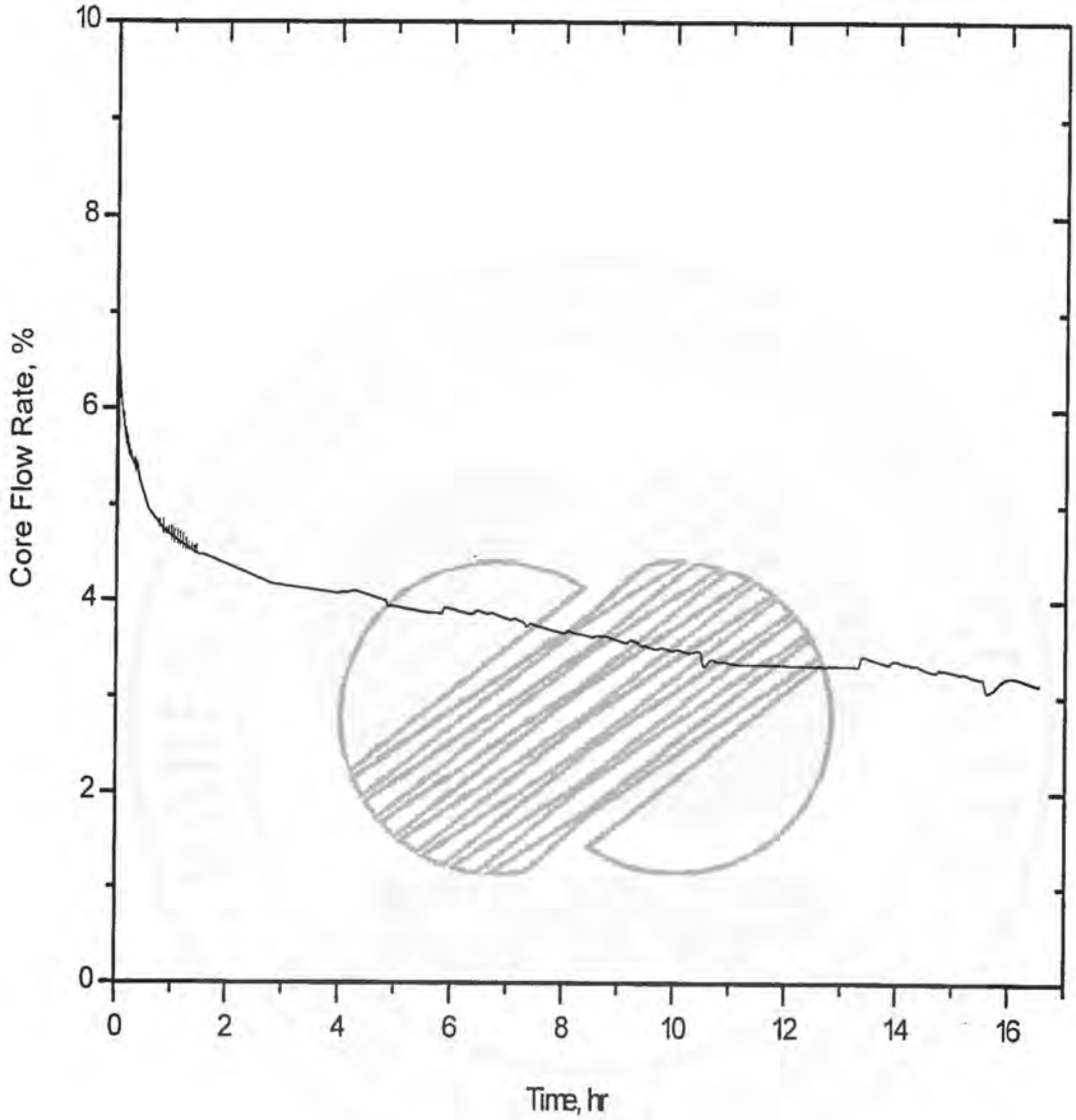

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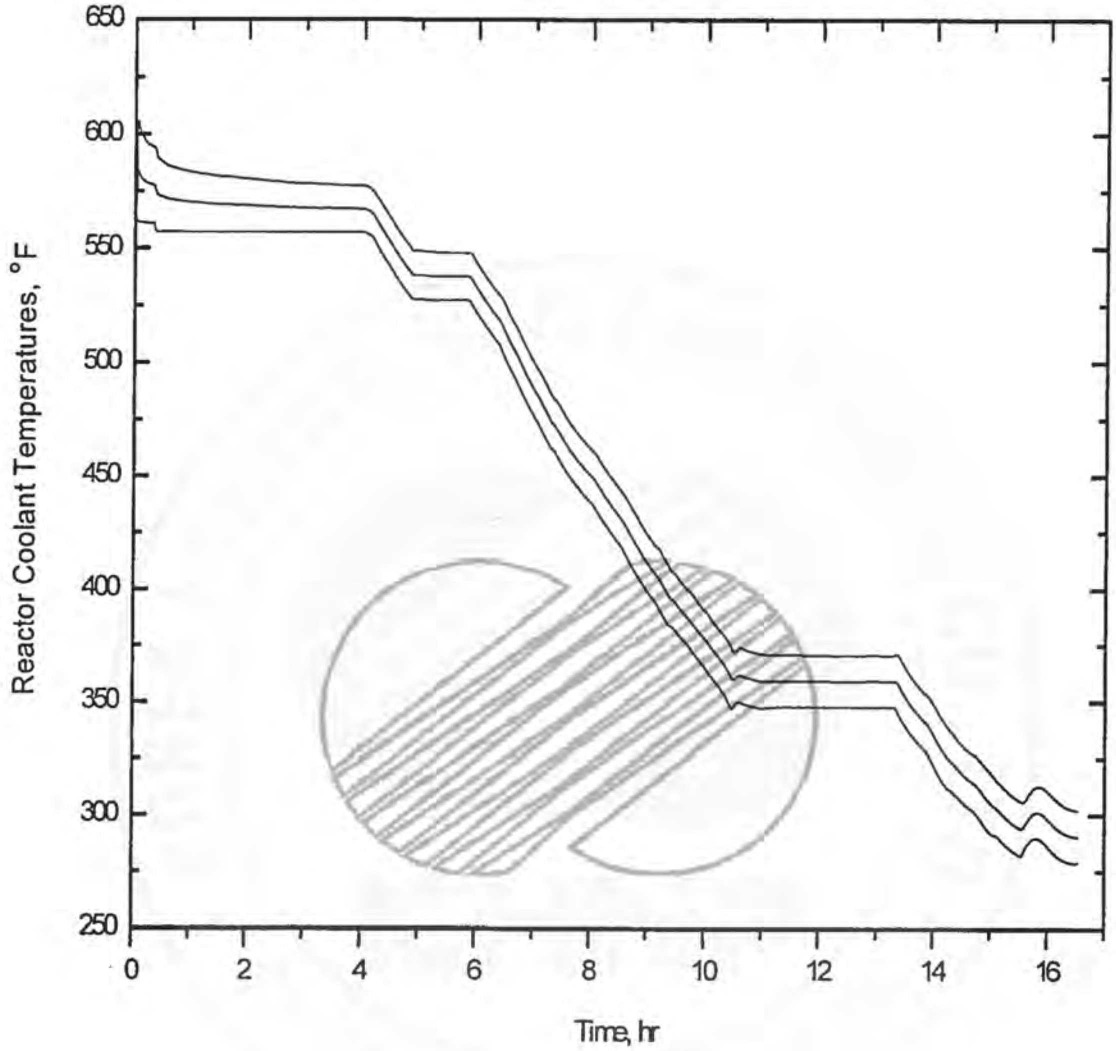
Core Flow Rate vs. Time for NCC Transient
Figure 5A-1



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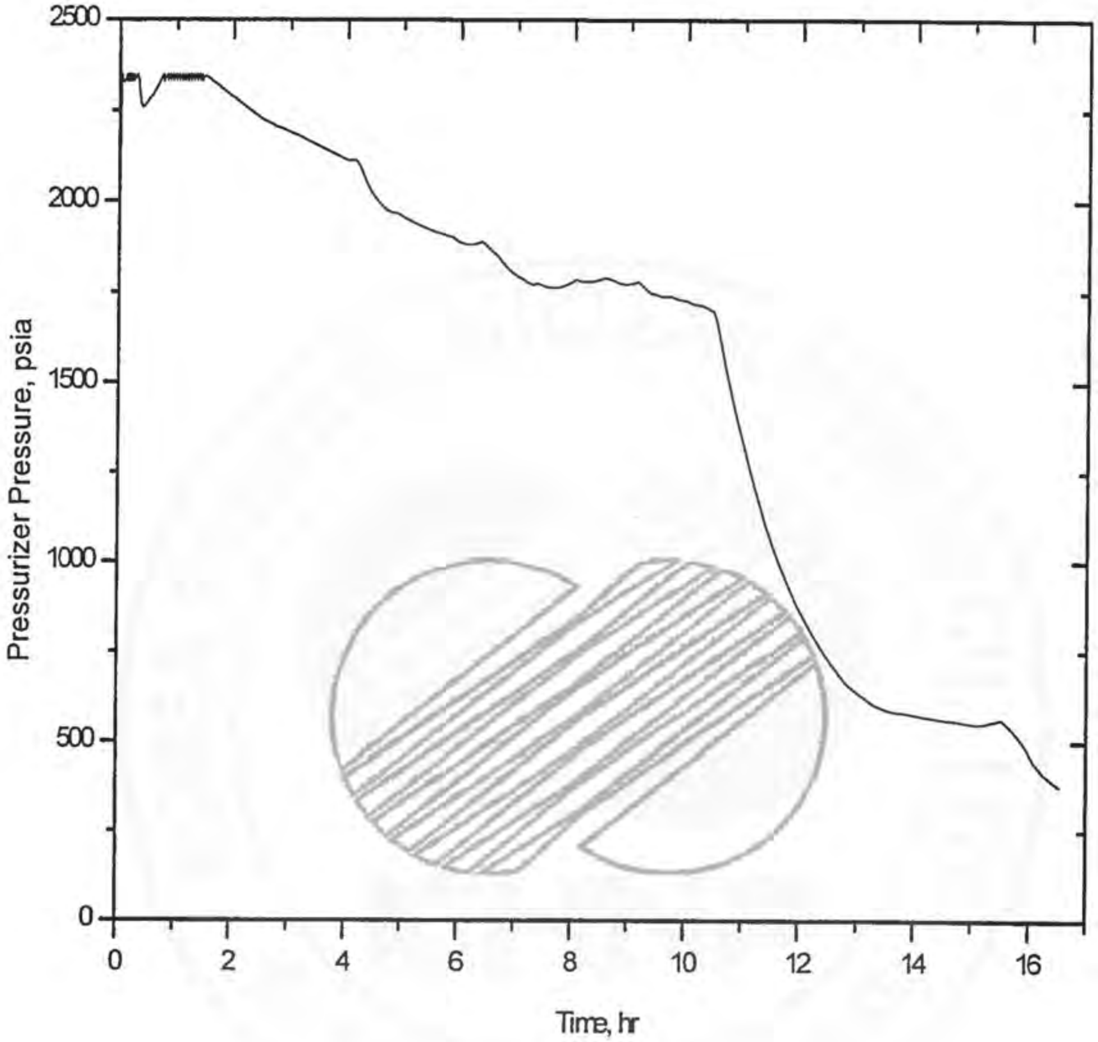
Reactor Coolant Temperatures vs. Time for NCC Transient

Figure 5A-2



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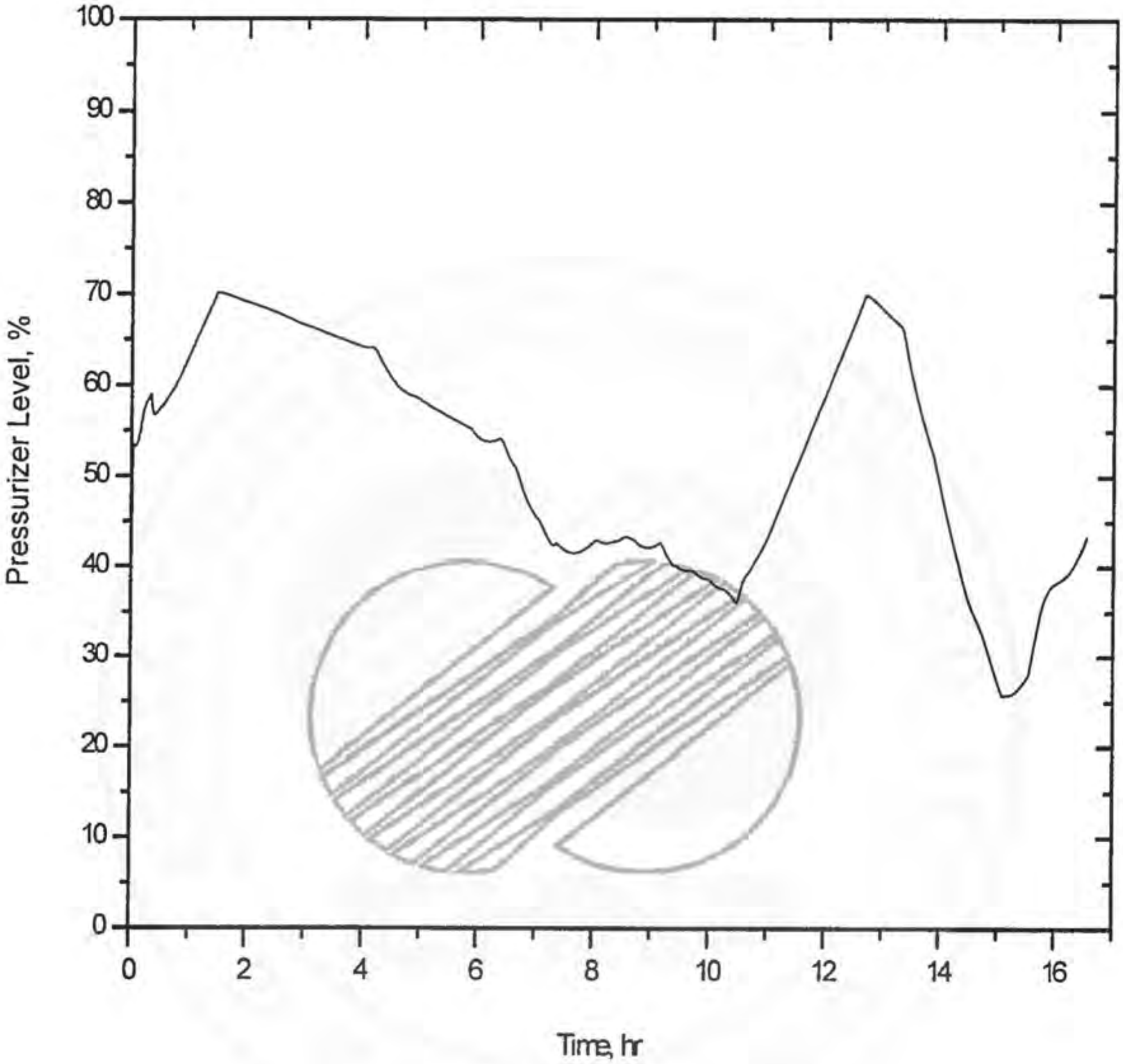
Pressurizer Pressure vs. Time for NCC Transient Figure 5A-3



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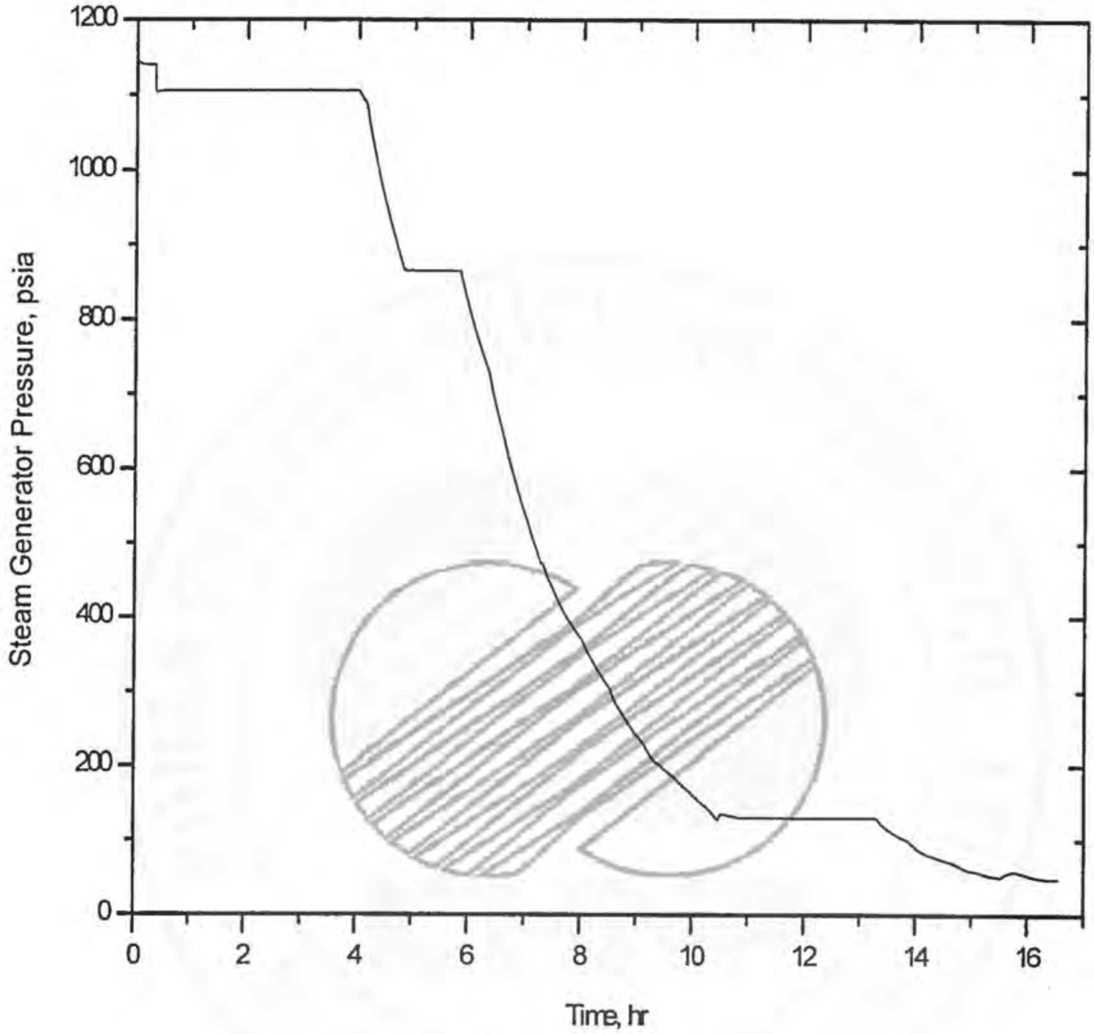
Pressurizer Level vs. Time for NCC Transient
Figure 5A-4



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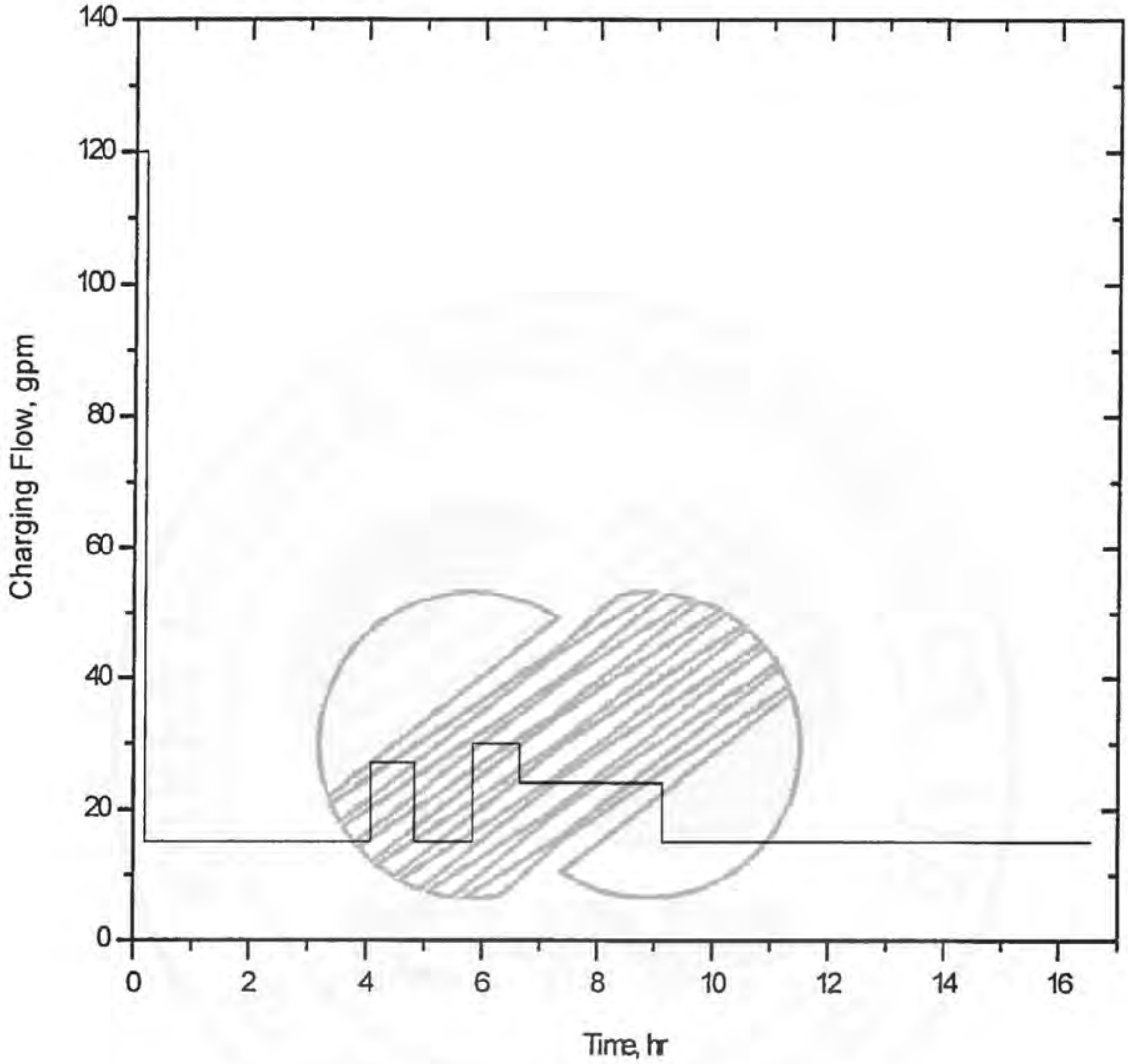
Steam Generator Pressure vs. Time for NCC Transient

Figure 5A-5



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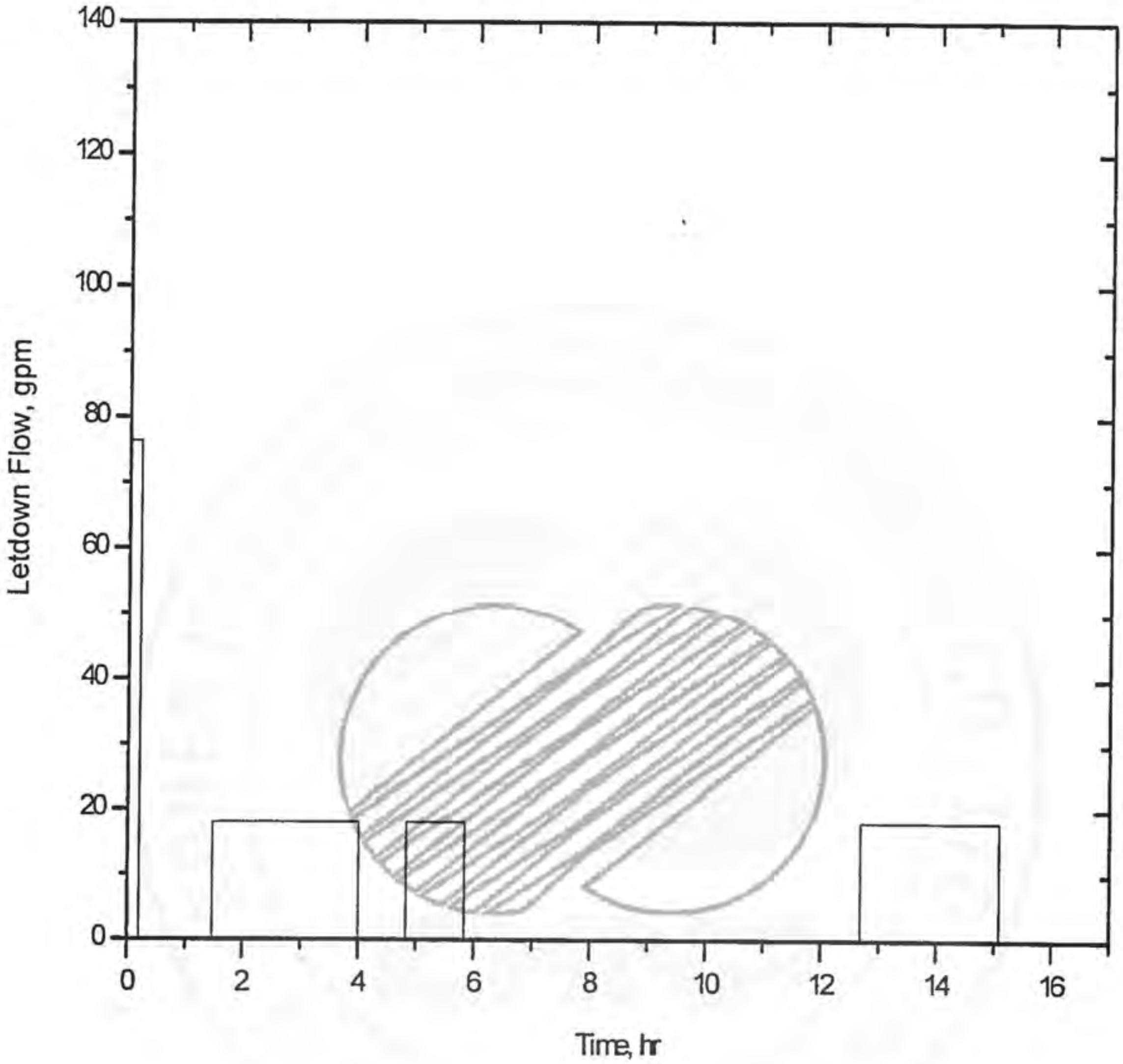
Charging Flow vs. Time for NCC Transient
Figure 5A-6



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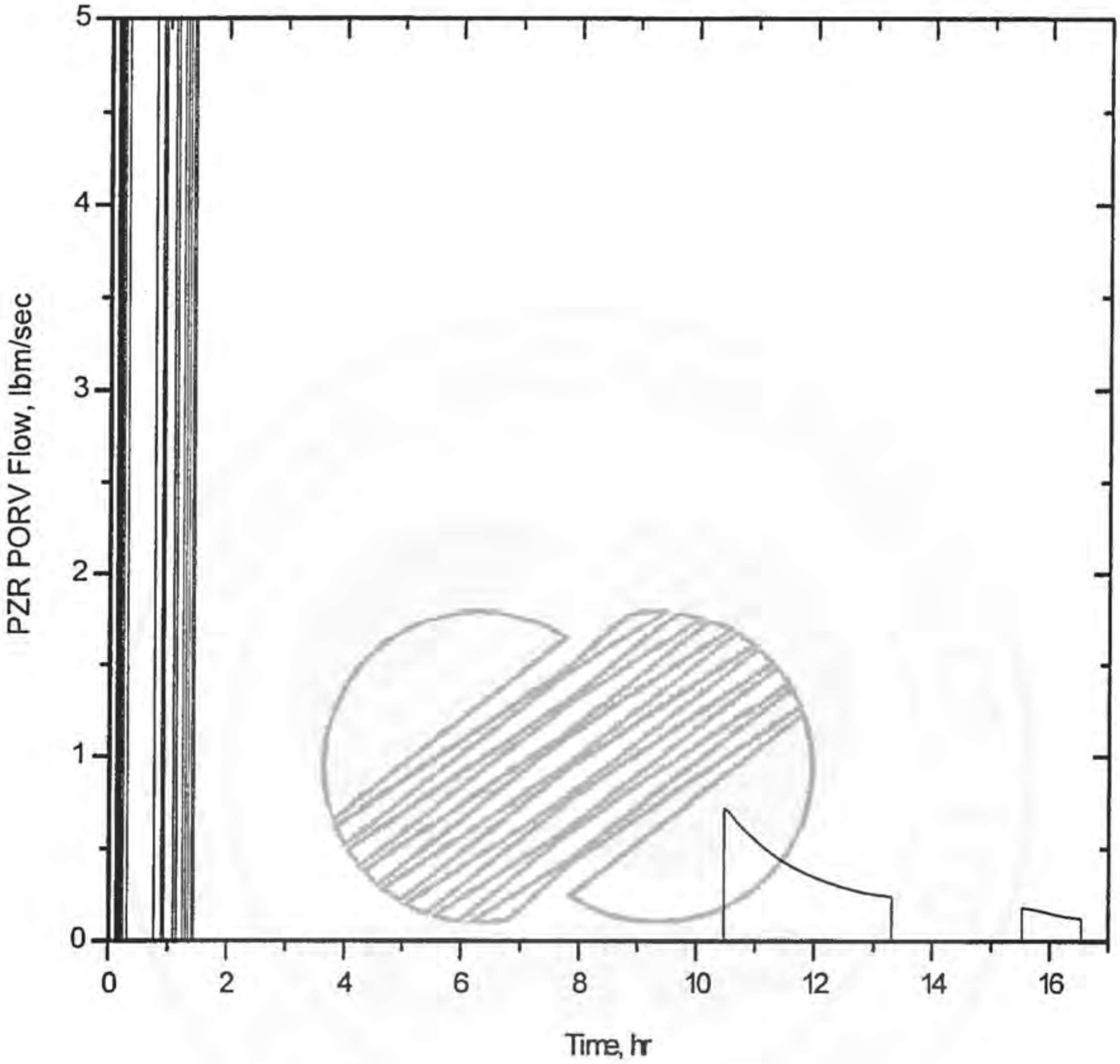
Letdown Flow vs. Time for NCC Transient
Figure 5A-7



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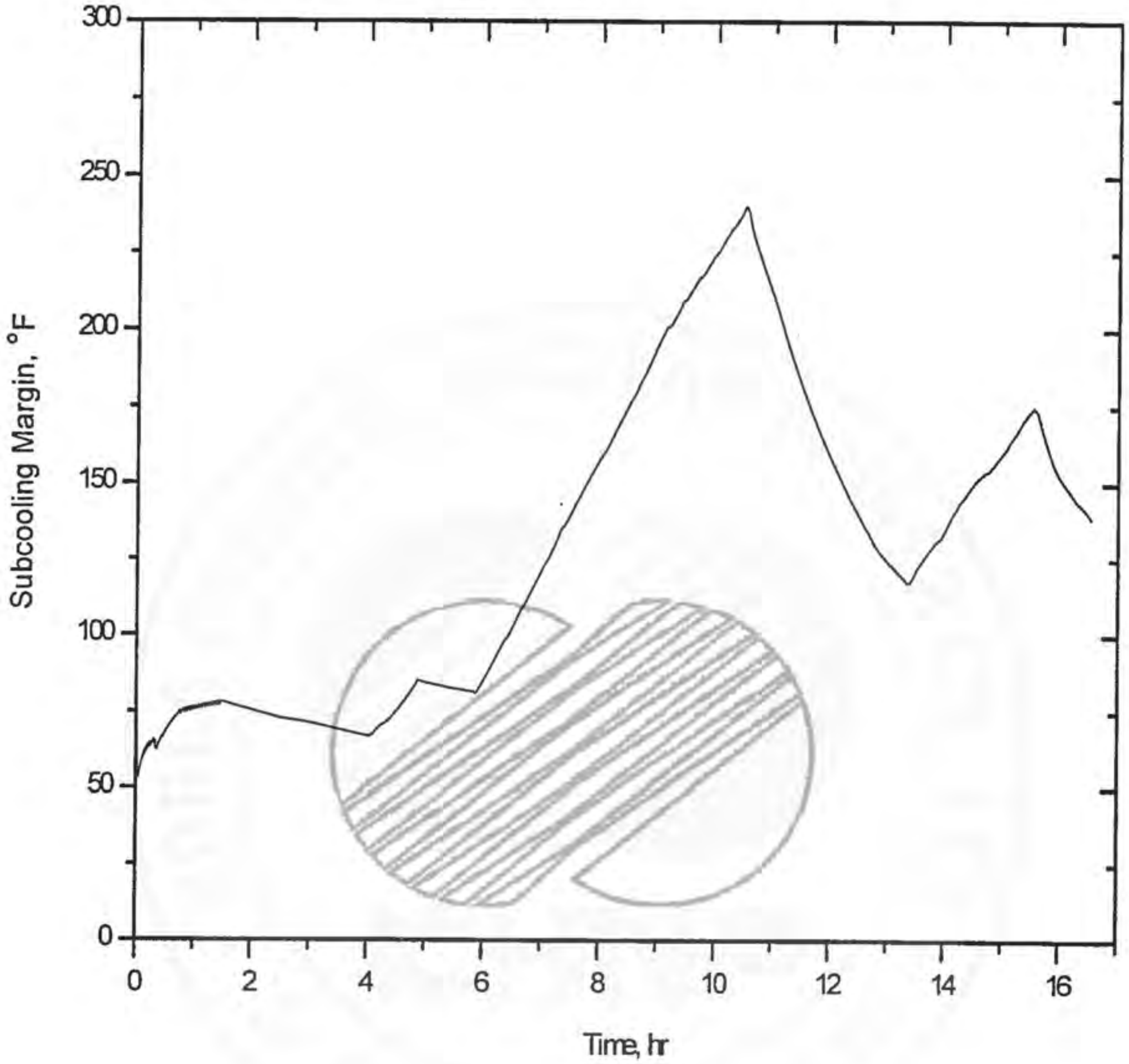
PZR PORV Flow vs. Time for NCC Transient
Figure 5A-8



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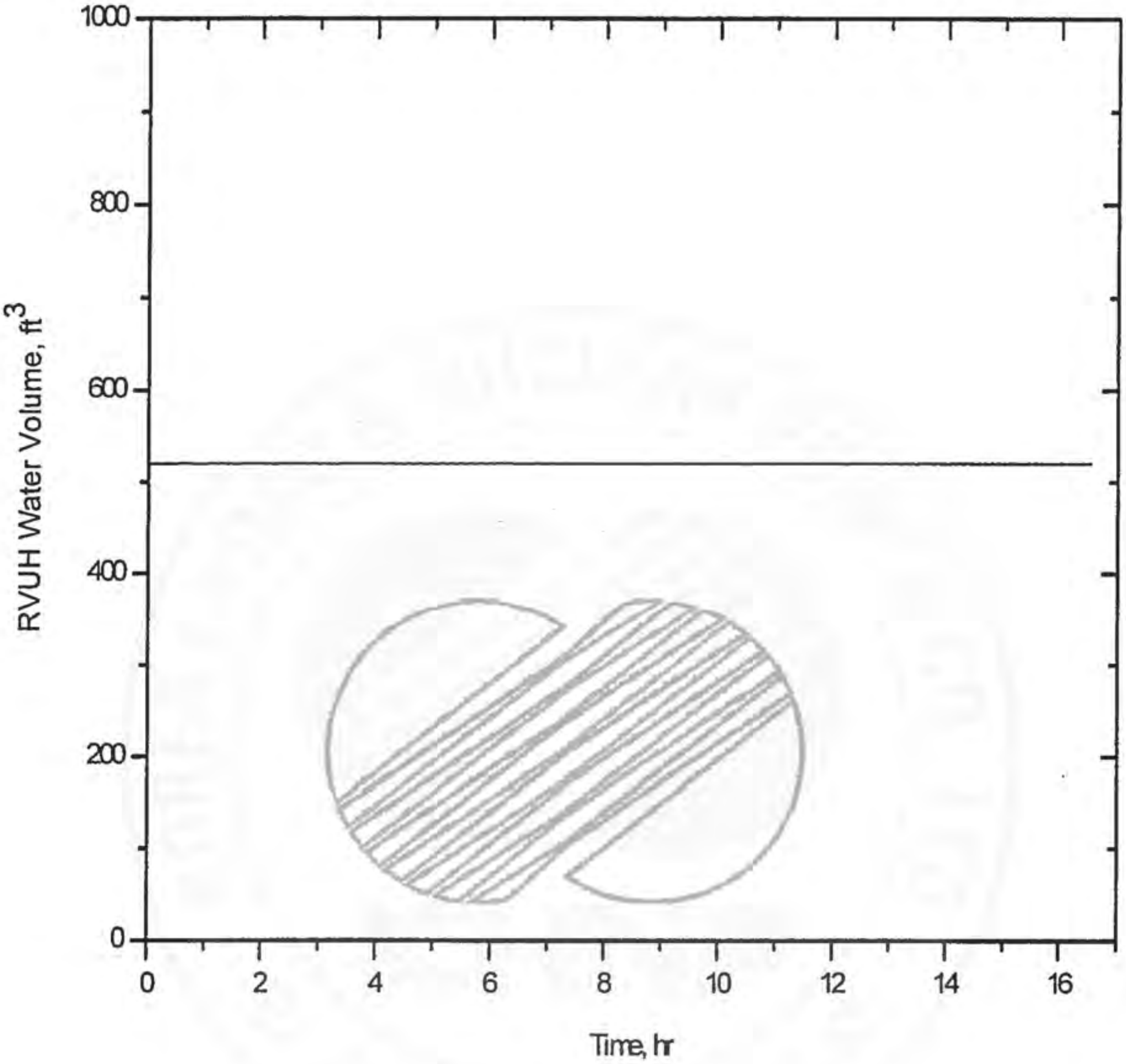
Subcooling Margin vs. Time for NCC Transient
Figure 5A-9



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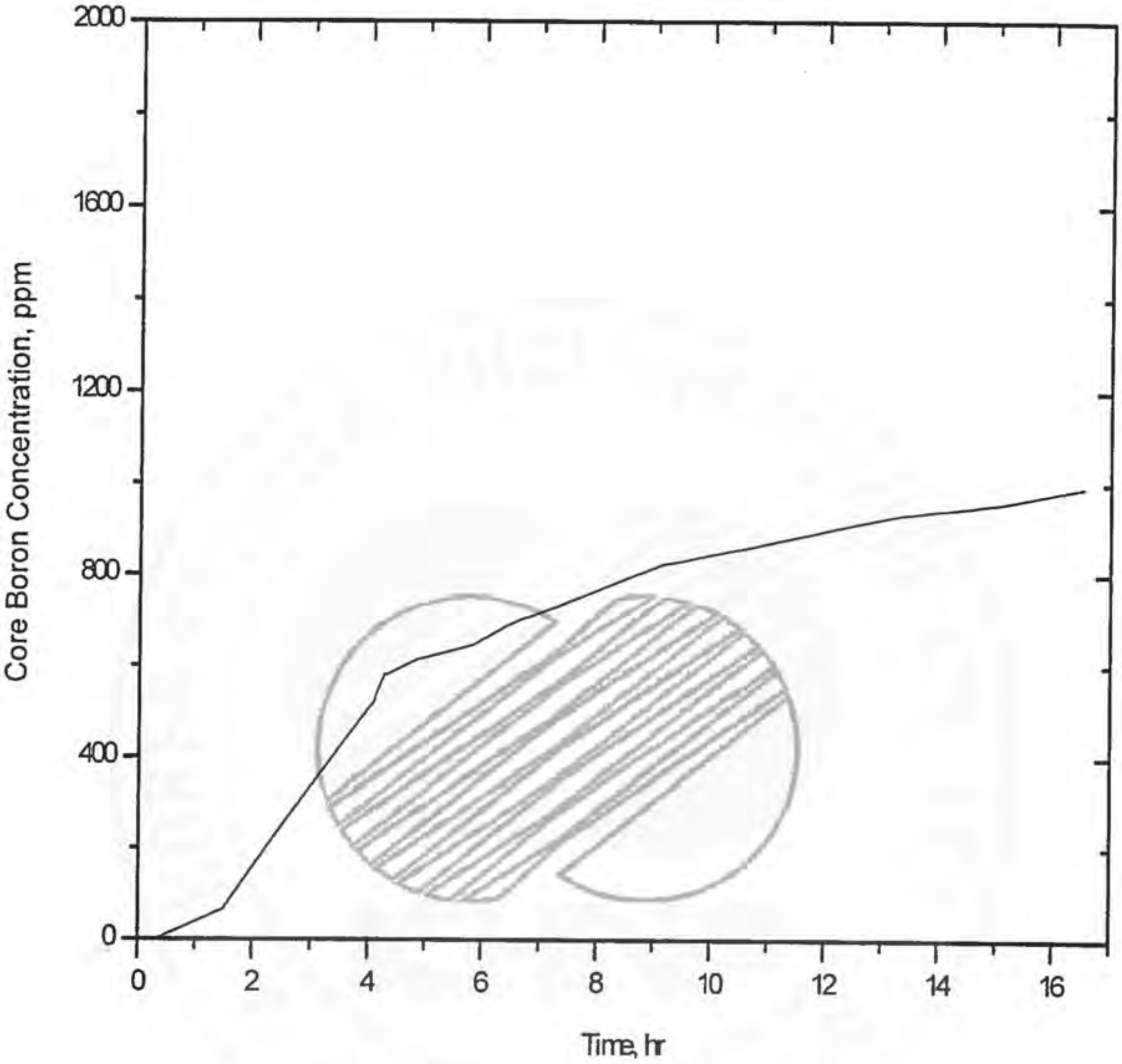
RVUH Water Volume vs. Time for NCC Transient Figure 5A-10



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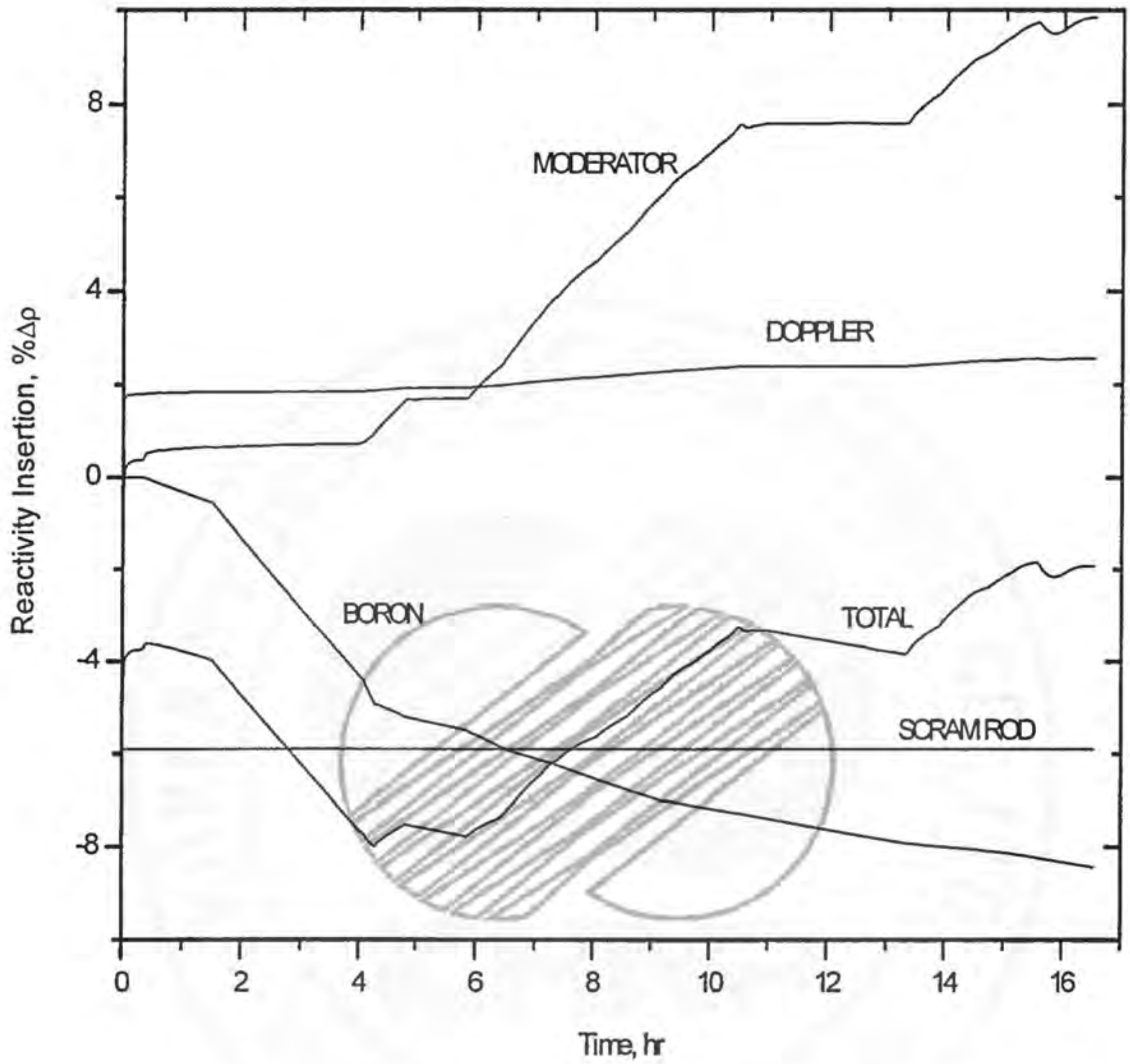
Core Boron Concentration vs. Time for NCC Transient

Figure 5A-11



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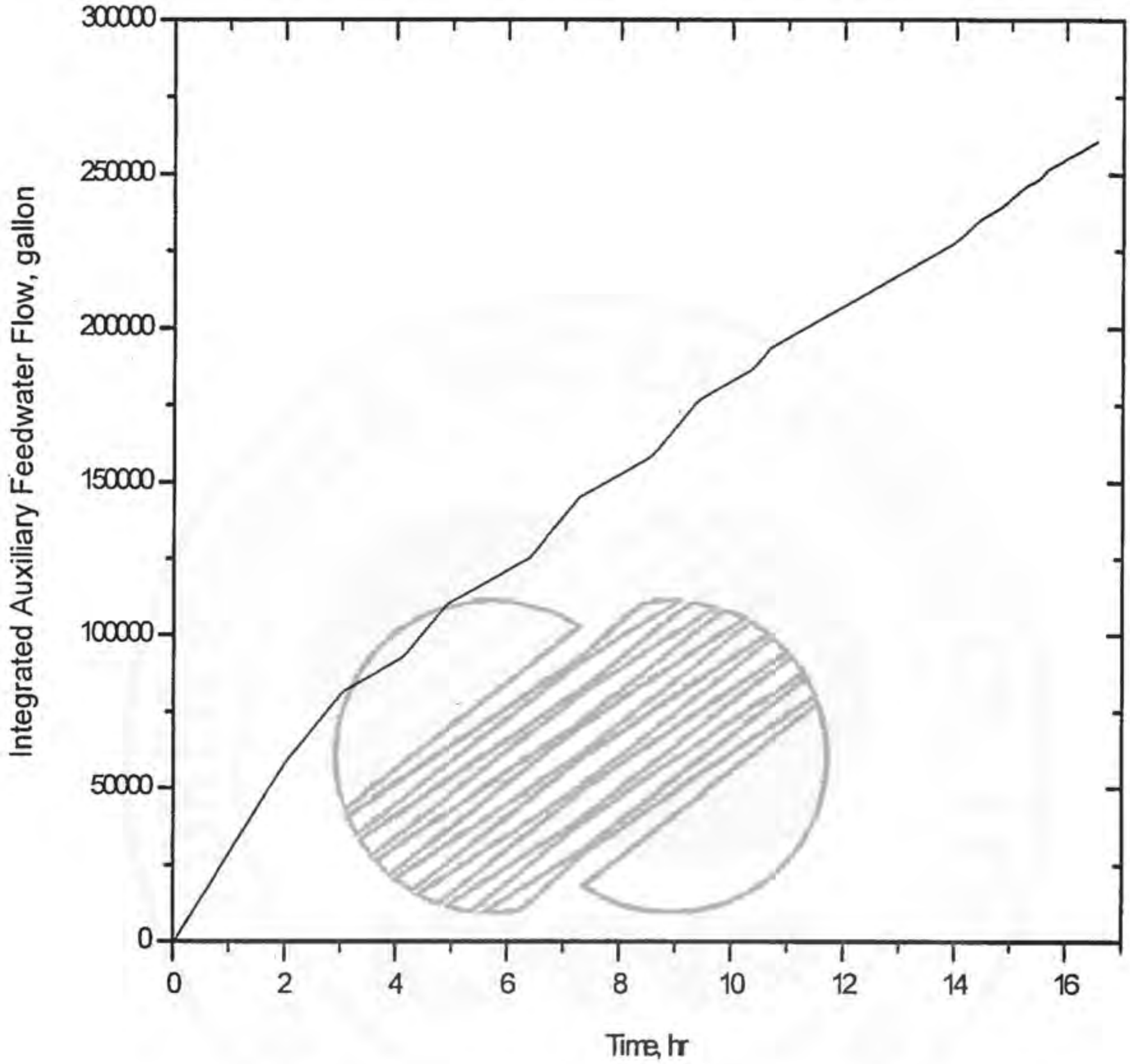
Reactivity Insertion vs. Time for NCC Transient

Figure 5A-12



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Integrated Auxiliary Feedwater Flow vs. Time for NCC Transient

Figure 5A-13