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DESIGN CONTROL DOCUMENT TIER 2

CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

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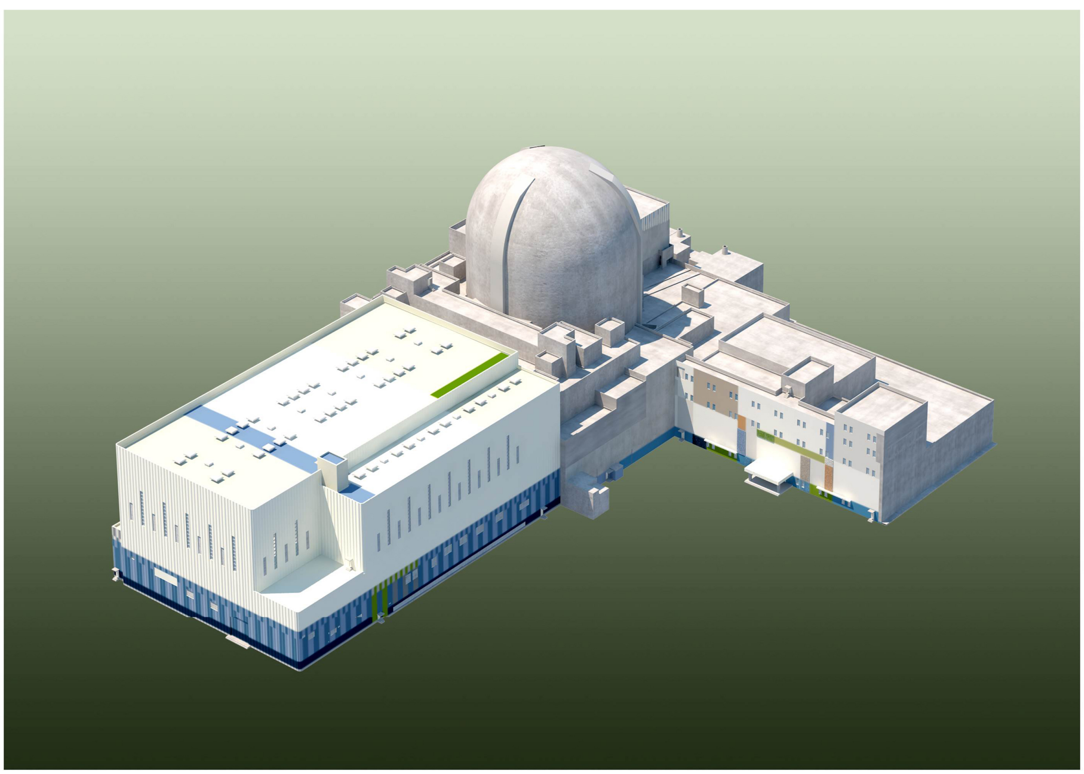

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CHAPTER 5 – REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

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ACRONYM AND ABBREVIATION LIST

ACP	auxiliary charging pump
ADV	atmospheric dump valve
ALMS	acoustic leak monitoring system
ANS	American Nuclear Society
ANSI	American National Standards Institute
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATM	atmosphere
BABT	boric acid bating tank
BAC	boric acid corrosion
BLPB	branch line pipe break
BTP	Branch Technical Position
CCW	component cooling water
CCWS	component cooling water system
CEDM	control element drive mechanism
CFR	Code of Federal Regulations
CIV	containment isolation valve
CM	condition monitoring
CMTR	certified material test report
COL	combined license
CSAS	containment spray actuation signal
CSP	containment spray pump
CSS	containment spray system
CVCS	chemical and volume control system
CVN	Charpy V-notch
DBE	design basis event
DCD	Design Control Document
DO	dissolved oxygen

DVI	direct vessel injection
EDT	equipment drain tank
EOL	end of life
EPRI	Electric Power Research Institute
ERVC	external reactor vessel cooling
FCAW	flux cored arc welding
FEI	fluid elastic instability
GDC	general design criteria (of 10 CFR Part 50, Appendix A)
GL	Generic Letter
GTAW	gas tungsten arc weld
GWMS	gaseous waste management system
HAZ	heat-affected zone
HJTC	heated junction thermocouple
HT	holdup tank
HX	heat exchanger
ICI	in-core instrumentation
IEEE	Institute of Electrical and Electronics Engineers
IHA	integrated head assembly
IPS	information processing system
IRWST	in-containment refueling water storage tank
ISI	inservice inspection
ISLOCA	intersystem loss of coolant accident
IST	inservice testing
ITAAC	inspections, tests, analyses, and acceptance criteria
IWSS	in-containment water storage system
LBB	leak before break
LOCA	loss of coolant accident
LOCV	loss of condenser vacuum
LST	lowest service temperature
LTOP	low temperature overpressure protection

LWMS	liquid waste management system
MCR	main control room
MFHX	miniflow heat exchanger
MOV	motor-operated valve
MRP	material reliability program
MSSV	main steam safety valve
NCC	natural circulation cooldown
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturers Association
NPS	nominal pipe size
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
OA	operational assessment
OD	outside diameter
OM	operator’s module
PCPS	pool cooling purification system
PIV	pressure isolation valves
POSRV	pilot-operated safety relief valve
PSA	probabilistic safety assessment
PSI	preservice inspection
P-T Limit	pressure-temperature limitation
PTLR	pressure and temperature limits report
PTS	pressurized thermal shock
PVRC	Pressure Vessel Research Committee
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
PZR	pressurizer
QIAS	qualified indication and alarm system
RCGVS	reactor coolant gas vent system

RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RDT	reactor drain tank
RG	Regulatory Guide
RMWT	reactor makeup water tank
RPS	reactor protection system
RSR	remote shutdown room
RTD	resistance temperature detector
RTE	random turbulent excitation
RT_{NDT}	reference temperature
RV	reactor vessel
RVI	reactor vessel internal
RVUH	reactor vessel upper head
SAFDL	specified acceptable fuel design limit
SAW	submerged arc welding
SCC	stress corrosion cracking
SCP	shutdown cooling pump
SCS	shutdown cooling system
SDCHX	shutdown cooling heat exchanger
SFP	spent fuel pool
SG	steam generator
SIAS	safety injection actuation signal
SIP	shutdown injection pump
SIS	safety injection system
SIT	safety injection tank
SMAW	shielded metal arc weld
SRP	Standard Review Plan
SS	sampling system
SSC	structure, system, and component

SSE	safe shutdown earthquake
SWMS	solid waste management system
TEMA	tubular exchanger manufacturers association
TMI	Three Mile Island
T_{NDT}	nil-ductility transition temperature
TT	thermally treated
USE	upper-shelf energy
VCT	volume control tank
WPS	welding procedure specification
WRC	Welding Research Council

CHAPTER 5 – REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

5.1 Summary Description

The reactor is a pressurized water reactor (PWR) with two coolant loops. The reactor coolant system (RCS) circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary system. The reactor vessel, steam generators, reactor coolant pumps, pressurizer, and associated piping are the major components of the RCS. Two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps, are connected to the reactor vessel, and one pressurizer is connected to one of the reactor vessel hot legs. All RCS components are located inside the containment building. Table 5.1.1-1 shows the principal parameters of the RCS. The reactor coolant pressure boundary (RCPB) components are consistent with 10 CFR 50.2 (Reference 1) and 10 CFR 50.55a (Reference 2). Applicable codes and standards of RCS components are listed in Table 3.2-1. The reactor vessel (RV) is equipped with suitable provision for connecting the head vent that meets the requirements of 10 CFR 50.34(f)(2)(vi) (Reference 3) (Three Mile Island [TMI] Action Item II.B.1).

The functions of the RCS are as follows:

a. Energy transfer from the reactor core to the steam generator (SG) where steam is produced for use in the turbine generator

b. Secondary barrier to the release of fission products from the reactor core to the environment

c. Sufficient cooling during all normal plant operations and expected transients to preclude significant fuel damage

d. Reactor coolant circulation with the required chemistry and boron concentration

e. System pressure control to a moderate extent through sprays

f. Rapid depressurization of the RCS by manual operation of the pressurizer pilot operated safety relief valves (POSRVs)

g. Venting of steam and noncondensable gases from RV closure head and pressurizer steam space

The major components of the RCS are as follows:

a. Reactor vessel: The reactor vessel contains fuel bundles, control rods, and other internals necessary for core support and flow direction. The six reactor coolant nozzles, two outlet and four inlet, are located with their centerlines on a common horizontal plane. The reactor vessel is a vertical right cylinder with two hemispherical heads. The lower head is welded to the reactor vessel shell and contains 61 in-core instrumentation penetrations. The upper closure head can be removed to provide access to the reactor vessel internals. The head is penetrated by 103 control element drive mechanism (CEDM) nozzles: eight of the nozzles are capped-off CEDM spares, and two are used for heated junction thermocouples (HJTCs). The reactor vessel is described in Section 5.3.

b. Steam generators: The steam generators provide the interface between the RCS and the main steam system. The steam generators are vertical U-tube heat exchangers with an integral economizer in which heat is transferred from the reactor coolant to the main steam system. Reactor coolant is prevented from mixing with the secondary steam by the steam generator tubes and the steam generator tube sheet, making the RCS a closed system that serves as a barrier to the release of radioactive materials from the core of the reactor to the secondary system and the containment building. The steam generators are described in Subsection 5.4.2.

c. Reactor coolant pumps: The four identical reactor coolant pumps (RCPs) are vertical single-stage, bottom suction, horizontal discharge, motor-driven centrifugal pumps designed to overcome the system flow resistances and circulate the reactor coolant at the flow rate required for design power operation. A flywheel on the shaft above the motor provides rotating inertia to increase the RCP coastdown time and reduce the rate of decay of reactor coolant flow if electrical power to the RCP motors is lost. The RCPs are described in Subsection 5.4.1.

d. Reactor coolant piping: Each of the heat transfer loops contains five pipe assemblies: one 106.7 cm (42 in) internal diameter pipe assembly between the reactor vessel outlet nozzle and SG inlet nozzle, two 76.2 cm (30 in) internal diameter pipe assemblies from the SG's two outlet nozzles to the RCP suction nozzles, and two 76.2 cm (30 in) internal diameter pipe assemblies from the RCP discharge nozzles to the reactor vessel inlet nozzles. The pipe assemblies are referred to as the hot leg, suction legs, and pump discharge legs. The pressurizer surge line is a 30 cm (12 in) schedule 160 pipe assembly located between the pressurizer and the hot leg in Loop 2. The reactor coolant piping is described in Subsection 5.4.3, and a typical arrangement of the piping is shown in Figures 5.1.3-1 and 5.1.3-2.

e. Pressurizer: The pressurizer is a vertical cylindrical vessel with a hemispherical top and bottom head. Electric heaters are installed vertically through the bottom head, and the spray nozzle and POSRV nozzles are mounted on the top head. The pressurizer is connected to the hot leg in RCS Loop 2 by the surge line. RCS pressure is controlled by the pressurizer where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer or is condensed by the pressurizer spray to limit pressure variations caused by the contraction or expansion of the reactor coolant. The average temperature of the reactor coolant varies with power level changing the pressurizer water level as the fluid expands or contracts. The charging control valves and letdown orifice isolation valves in the chemical and volume control system (CVCS) are used to maintain a programmed pressurizer water level. The pressurizer is described in Subsection 5.4.10.

f. Pressurizer POSRVs: Overpressure protection for the reactor coolant pressure boundary is provided by four POSRVs connected to the top of the pressurizer. These valves discharge to the in-containment refueling water storage tank where the steam is released under water to be condensed and cooled. Pressurizer POSRVs are described in Subsection 5.4.14.

5.1.1 Schematic Flow Diagram

The schematic flow diagram of the RCS is shown in Figure 5.1.1-1. The principal pressures, temperatures, and design minimum flow rates of the RCS under normal steady-state, full-power operating conditions are provided in Table 5.1.1-2. The RCS volumes at cold conditions are shown in Table 5.1.1-3.

5.1.2 Flow Diagram

The flow diagram of the RCS is shown in Figure 5.1.2-1. The entire RCS is located within the containment. Fluid systems that are connected to the RCS and within the limits of the reactor coolant pressure boundary, as defined in ANSI/ANS 51.1 (Reference 4), are identified and the relevant piping and instrument diagrams are referenced in other sections. The flow diagrams for the reactor coolant pumps and the pressurizer are provided in Figures 5.1.2-2 and 5.1.2-3, respectively.

The major components of the system are an RV, two parallel heat transfer loops, each containing one SG and two RCPs, and a pressurizer connected to the hot leg in RCS Loop 2. All of these major system components are located inside the containment building.

During normal operation, the reactor coolant is circulated through the RV and SGs by the RCPs. The reactor coolant is heated by fission energy produced in the core as it passes through the RV and is cooled in the SGs as it gives up heat to the secondary system.

The reactor coolant also serves as a neutron absorber (boron) for reactivity control. Except for some local boiling in the hottest channels in the core, the reactor coolant remains in a subcooled condition by maintaining a high system pressure. The RCS provides a barrier against the uncontrolled release of reactor coolant and radioactive materials to the containment.

RCS pressure is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer, or is condensed by the pressurizer spray to limit pressure variations caused by contraction or expansion of the reactor coolant.

Overpressure protection for the RCS is provided by four POSRVs connected to the top of the pressurizer. These valves discharge to the in-containment refueling water storage tank, where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the in-containment refueling water storage tank, it is vented to the containment atmosphere.

The RCS is interfaced with a number of auxiliary systems, including the main steam supply system, CVCS, SCS, SIS, CCWS, and sampling system.

The isolation valves are connected in series to the RCS and are a part of the RCPB. The isolation valves are designed to meet the exclusion requirements of 10 CFR 50.55a(c).

5.1.3 Elevation Drawings

The RCS arrangement plan and elevation drawings are provided as Figures 5.1.3-1 and 5.1.3-2, respectively.

Major components of the RCS are surrounded by concrete structures that provide support plus shielding and missile protection as needed. General arrangement drawings illustrating principal dimensions of the RCS in relationship to the surrounding building structures are presented in Figures 1.2-2 through 1.2-8 in Section 1.2.

5.1.4 Combined License Information

No COL information is required with regard to Section 5.1.

5.1.5 References

1. 10 CFR 50.2, “Definitions,” U.S. Nuclear Regulatory Commission.

2. 10 CFR 50.55a, “Codes and Standards,” U.S. Nuclear Regulatory Commission.

3. 10 CFR 50.34, “Contents of Applications; Technical Information,” U.S. Nuclear Regulatory Commission.

4. ANSI/ANS 51.1-1983, “American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plant,” American Nuclear Society, 1983.

Table 5.1.1-1

Reactor Coolant System Design Parameters

Parameter	Value
Design thermal power, MWt (including net heat addition from pumps)	4,000
Design pressure, kg/cm^2A (psia)	175.8 (2,500)
Design temperature (except pressurizer), °C (°F)	343.3 (650)
Pressurizer design temperature, °C (°F)	371.1 (700)
Coolant flow rate, kg/hr (lbm/hr)	75.6×10^6 (166.6×10^6)
Cold leg temperature, operating, °C (°F)	290.6 (555)
Average temperature, operating, °C (°F)	307.2 (585)
Hot leg temperature, operating, °C (°F)	323.9 (615)
Normal operating pressure, kg/cm^2A (psia)	158.2 (2,250)
System water volume, m^3 (ft^3) (with pressurizer)	455.3 (16,079)
Pressurizer water volume, m^3 (ft^3) (full power)	33.2 (1,171)
Pressurizer steam volume, m^3 (ft^3) (full power)	35.7 (1,260)

Table 5.1.1-2

Process Data Point Tabulation [1]

Parameter	Pressurizer ①[2]	Steam Generator 1 Midpoint ②[2]	Pump Suction ③[2]	Pump Outlet ④[2]	Reactor Vessel Midpoint ⑤[2]	Hot Leg ⑥[2]	Steam Generator 2 Midpoint ⑦[2]
Pressure, kg/cm^2A (psia)	158.2 (2,250)	157.5 (2,240)	156.1 (2,220)	163.8 (2,330)	161.7 (2,300)	158.9 (2,260)	157.5 (2,240)
Temperature, °C (°F)	344.8 (652.7)	307.2 (585.0)	290.6 (555.0)	290.6 (555.0)	308.8 (587.8)	323.9 (615.0)	307.2 (585.0)
Mass flow rate, kg/hr (lbm/hr)	N/A	37.8×10^6 (83.3×10^6)	18.9×10^6 (41.65×10^6)	18.9×10^6 (41.65×10^6)	75.6×10^6 (166.6×10^6)	37.8×10^6 (83.3×10^6)	37.8×10^6 (83.3×10^6)
Volumetric flow rate, L/min (gpm)	N/A	885,707 (233,979)	421,854 (111,442)	421,854 (111,442)	1,778,538 (469,840)	940,076 (284,342)	885,707 (233,979)

(1) For steady state, 100 % power conditions

(2) See Figure 5.1.1-1 for data points

Table 5.1.1-3

Reactor Coolant System Volumes

Component	Volume[1]
RV, m^3 (ft^3)	166.3 (5,872.2)
Steam generators, m^3 (ft^3)	86.8 (3,067.0) (each), without tube plugging
RCPs, m^3 (ft^3)	3.26 (115) (each)
Pressurizer, m^3 (ft^3)	68.0 (2,400)
Piping: Hot leg, m^3 (ft^3) Cold leg, m^3 (ft^3)	 3.8 (135.6) (each) 6.2 (219.9) (each)
Surge line, m^3 (ft^3) (nominal)	1.3 (44.9)
Spray line, m^3 (ft^3)	0.5 (17.1)

(1) Volumes determined at cold 21 °C (70 °F) conditions

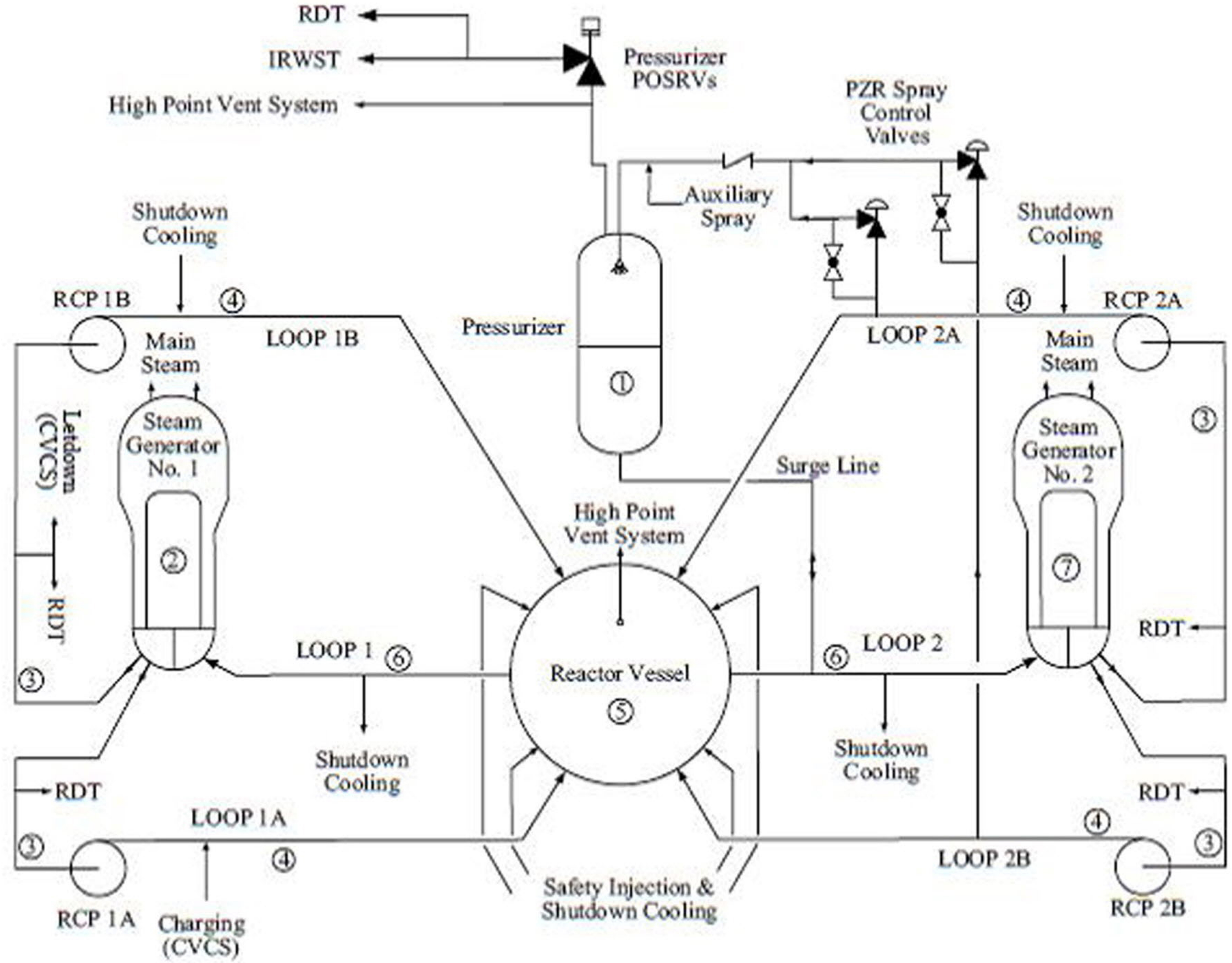

Figure 5.1.1-1   Reactor Coolant System Schematic Flow Diagram

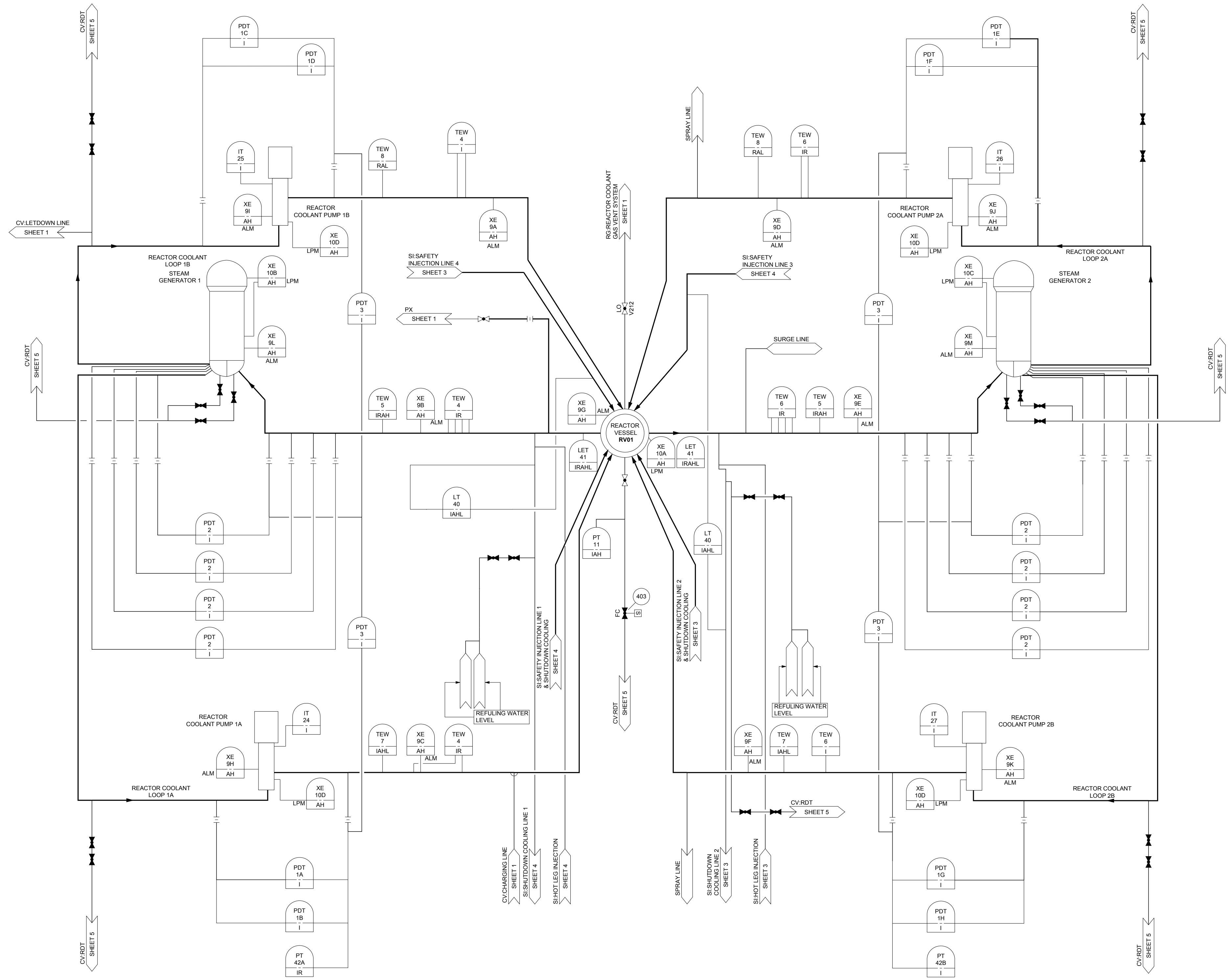

Figure 5.1.2-1 Reactor Coolant System Flow Diagram

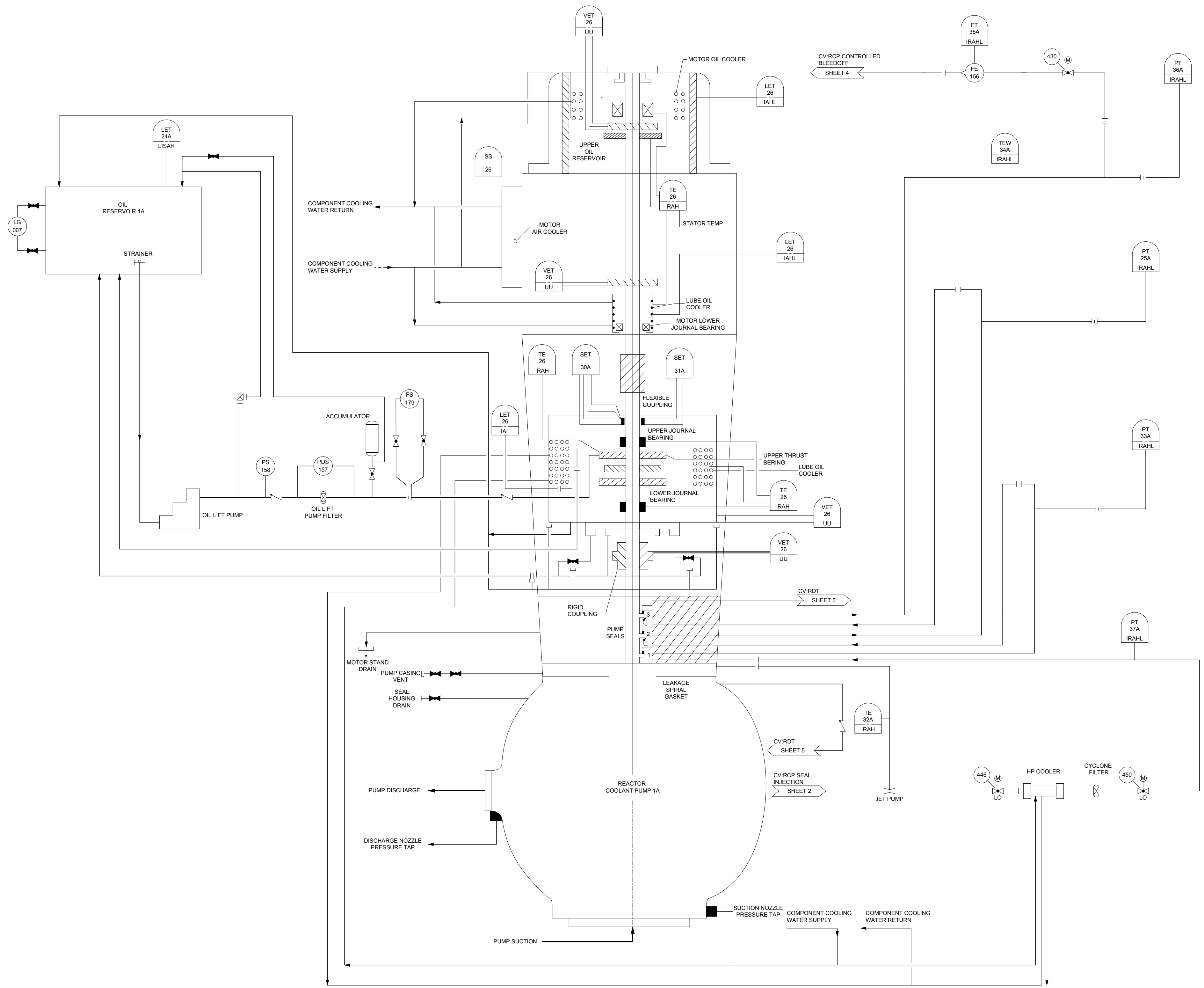

Figure 5.1.2-2   Reactor Coolant Pump Flow Diagram

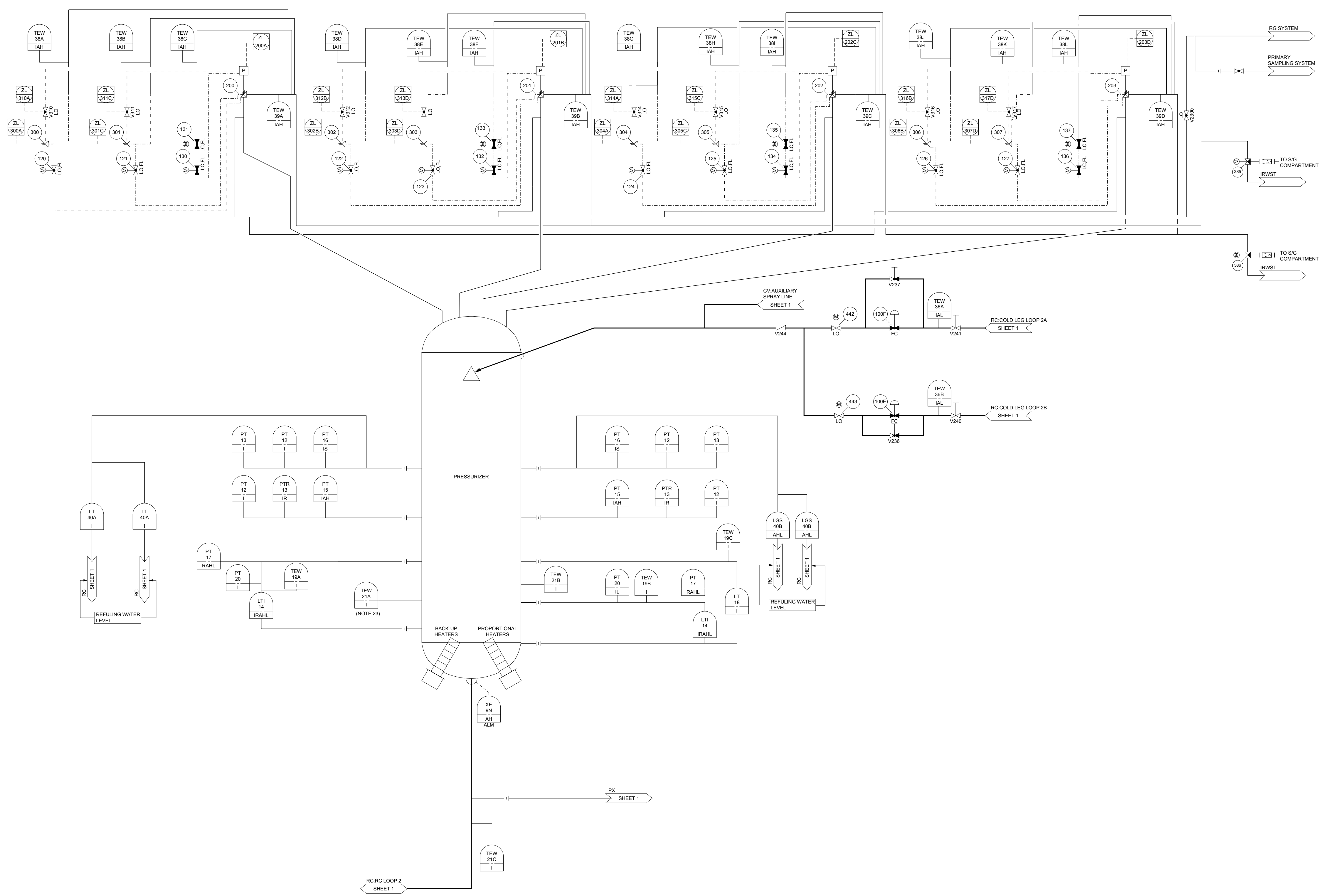

Figure 5.1.2-3 Pressurizer and POSRV Flow Diagram

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 5.1.3-1 Reactor Coolant System Arrangement Plan

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 5.1.3-2 Reactor Coolant System Arrangement Elevation

5.2 Integrity of the Reactor Coolant Pressure Boundary

This section describes the measures that provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) throughout the facility's design life. The RCPB is defined in accordance with ANSI/ANS 51.1 (Reference 1). The RCPB includes all pressure-containing components such as pressure vessels, piping, pumps, and valves that are:

a. Part of the reactor coolant system (RCS)

b. Connected to the RCS, up to and including the following:

   1) The outermost containment isolation valve in piping that penetrates the containment

   2) The second of two valves normally closed during reactor operation in piping that does not penetrate the containment

5.2.1 Conformance with Codes and Code Cases

5.2.1.1 Conformance with 10 CFR 50.55a

RCPB components are designed and fabricated as Class 1 components in accordance with ASME Section III (Reference 2), except for the components that meet the exclusion requirements of 10 CFR 50.55a(c) (Reference 3). RCPB components that meet the exclusion requirements are classified as Quality Group B in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.26 (Reference 4) and are fabricated as Class 2 components in accordance with ASME Section III. The classification of RCPB components conforms with the requirements of 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 1 (Reference 5). The remaining safety-related components are classified as Quality Group C in accordance with NRC RG 1.26 and are fabricated as Class 3 components in accordance with ASME Section III.

Subsection 5.2.4 and Section 6.6 provide a description of the application of ASME Section XI (Reference 6) to the RCPB. Subsection 3.9.6 provides a description of the application of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) (Reference 7) to the RCPB.

The components and code classes that are listed in Table 5.2-1 are in accordance with the provisions of 10 CFR 50.55a with this exception: the applicable ASME Code edition for the APR1400 is the 2007 Edition with 2008 Addenda. Table 3.2-1 provides the component classifications of pressure vessels, piping, pumps, valves, and storage tanks, along with the applicable component codes. The proposed inspections, tests, analyses, and acceptance criteria (ITAAC), as required by 10 CFR 52.47(b)(1) (Reference 8), are addressed in Tier 1 of the APR1400 DCD based on the selection criteria in Section 14.3.

5.2.1.2 Compliance with Applicable Code Cases

RCPB components are designed and fabricated in accordance with ASME Section III.

The applicable ASME Code Cases that are in conformance with the requirements of GDC 1 and 10 CFR 50.55a and that are used in the plant design and manufacturing are listed in Table 5.2-4. NRC RGs 1.84 (Reference 9), 1.147 (Reference 10), and 1.192 (Reference 11) are used in determining the applicable ASME Code Cases. The COL applicant is to address the addition of ASME Code Cases that are approved in NRC RG 1.84 (COL 5.2(1)). The COL applicant is to address the ASME Code Cases invoked for the ISI program of a specific plant (COL 5.2(2)). The COL applicant is to address the ASME Code Cases invoked for operation and maintenance activities (COL 5.2 (3)).

5.2.2 Overpressure Protection

Overpressure protection systems include all pressure-relieving devices for the following systems:

a. Reactor coolant system (RCS)

b. Primary side of auxiliary or emergency systems connected to the RCS

c. Secondary side of steam generators (SGs)

5.2.2.1 Design Bases

5.2.2.1.1 Design Bases for Overpressure Protection of the Reactor Coolant System

The functional design of the overpressure protection is in conformance with the requirements of GDC 15 and GDC 31.

Overpressurization of the RCS and steam generators is precluded by operation of the pressurizer POSRVs and main steam safety valves (MSSVs) and by the reactor protection system (RPS). Pressure relief capacity for the RCS and steam generators is conservatively sized to satisfy the overpressure requirements of ASME Section III, Division 1, NB 7000. The pressurizer POSRVs, MSSVs, and RPS are designed to maintain the RCS pressure below 110 percent of design pressure during the worst-case loss-of-load event with a delayed reactor trip. The MSSVs are sized conservatively to release steam flow equal to the full power level. Steam generator pressure is limited to less than 110 percent of steam generator design pressure during the worst-case transient.

In order to determine the appropriate pressurizer POSRV capacity, a sensitivity study was performed with the worst-case initial condition and nuclear parameters to conservatively evaluate the effect of valve capacity on the maximum RCS pressure during the design basis event. As shown in Figure 5.2.2-1, the design POSRV capacity is determined at the point where an additional increase in the capacity has a negligible effect on reducing the maximum RCS pressure during the loss-of-load transient.

At the onset of a loss-of-load transient, the reactor coolant and main steam systems are at maximum rated output plus a 2 percent uncertainty margin. No credit is taken for plant control systems such as letdown, charging, pressurizer spray, turbine bypass, reactor power cutback, and feedwater addition (main and auxiliary) after turbine trip in the loss-of-load analysis. A reactor scram is assumed to be initiated by the second safety grade signal from the RPS.

Peak reactor coolant and steam generator pressures are limited to less than 110 percent of design pressures during the worst-case transient (Figures 5.2.2-2, 5.2.2-3, and 5.2.2-4). Reasonable assurance of reliable overpressure protection is provided by the pressurizer POSRVs, the MSSVs, and the RPS.

Pressurizer sizing is described in Subsection 5.4.10.

5.2.2.1.2 Design Bases for Low Temperature Overpressure Protection

Each shutdown cooling system (SCS) suction line relief valve (SI-179 or SI-189) is designed to protect the RCS in a failure that initiates a pressure transient. The use of either SCS suction line relief valve provides sufficient pressure relief capacity to mitigate the most limiting low temperature overpressure protection (LTOP) events during low-temperature conditions.

The maximum pressure for low temperature overpressure protection is limited to 43.9 kg/cm^2A (625 psia), 20 percent of RCS hydraulic test pressure (219.7 kg/cm^2A [3,125 psia]), which is the maximum RCS pressure allowed under the minimum operating temperature required in Appendix G of ASME Section III. The SCS suction line relief valves are designed in accordance with LTOP requirements.

The LTOP is designed in accordance with Branch Technical Position (BTP) 5-2 (Reference 12). Overpressure protection of the RCS during low-temperature conditions is provided by the relief valves, SI-179 and SI-189, located in the SCS suction lines. Subsection 5.4.7 provides a description of the SCS. The SCS is schematically shown on the SCS flow diagram (Figure 5.4.7-3), RCS flow diagram (Figure 5.1.2-1) and safety injection system (SIS) flow diagram (Figure 6.3.2-1). The SCS suction line relief valves are shown in Figure 6.3.2-1 and described in Subsection 5.4.7.2.2.

Alignment of the SCS suction line relief valve to the RCS is specified by plant procedures to provide reasonable assurance of RCS overpressure protection for all temperatures below the temperature for which LTOP is required, called the LTOP temperature and designated as T_{LTOP}. Pressure-temperature (P-T) Limit curves are provided in the Pressure and Temperature Limits Report (PTLR) (Reference 13). For temperatures above the LTOP temperature, overpressure protection is provided by the pressurizer POSRVs, which is described in Subsection 5.2.2.4.1.

5.2.2.1.2.1 Design Criteria

The criteria considered in the design of the overpressure mitigating system to provide LTOP for the RCS are provided in Subsections 5.2.2.1.2.2 through 5.2.2.1.2.5.

5.2.2.1.2.2 Credit for Operator Action

No credit is taken for operator action in the limiting transient analyses described in Subsection 5.2.2.2.2.1. In the analysis, overpressure protection is provided by the SCS suction line relief valves.

5.2.2.1.2.3 Single Failure

In the LTOP mode, each SCS suction line relief valve is designed to protect the reactor vessel given a single failure in addition to the event that initiates the pressure transient. The event initiating the pressure transient is considered to result from either an operator error or equipment malfunction. The SCS suction line relief valve is independent of a loss of offsite power. Each SCS suction line relief valve is a self-actuating, spring-loaded liquid relief valve, which does not require control circuitry. The relief valve opens when the RCS pressure exceeds its setpoint.

The redundant SCS suction lines between the RCS and the SCS suction line relief valves meet the single failure criteria as described in Table 5.4.7-2. No single failure of an isolation valve or its associated interlock will prevent one relief valve from performing its intended function.

5.2.2.1.2.4 Testability

Periodic testing of the SCS LTOP relief valves and suction line isolation valves is defined in Subsections 3.4.11 and 3.4.13 in Technical Specifications (Chapter 16).

5.2.2.1.2.5 Seismic Design and IEEE Standards 308 and 603 Criteria

The SCS suction line relief valves, isolation valves, associated interlocks, and instrumentation are designed to seismic Category I requirements and are addressed in

Subsections 3.2.1 and 5.4.7.2.4 and Table 3.2-1. The interlocks and instrumentation associated with the SCS suction line isolation valves are described in Subsections 5.4.7, 7.6.1, and 7.6.1.1 and Table 7.6-1. The interlocks satisfy the applicable portions of IEEE Standards 279, 308, and 603 criteria.

5.2.2.2 Design Evaluation

5.2.2.2.1 Design Evaluation for Overpressure Protection of the Reactor Coolant System

Section 15.2 provides the functional design evaluation of the overpressurization protection system. The analysis demonstrates the adequacy of the overpressure protection system to maintain secondary and primary operating pressures within their respective pressure limits. The analytical model used in the analysis is addressed in Chapter 15.

The assumptions that are used in the analysis are provided in Chapter 15. The assumptions were chosen to maximize the required pressure-relieving capacity of the primary and secondary sides. The analysis for the most severe anticipated transient demonstrates that sufficient relieving capacity is provided to prevent the pressure from exceeding 110 percent of the design pressure when acting in conjunction with the reactor protective system.

5.2.2.2.2 Design Evaluation for Low Temperature Overpressure Protection

The information in the following subsections is provided to demonstrate that the SCS suction line relief valves meet the criteria provided in Subsection 5.2.2.1.2.

5.2.2.2.2.1 Limiting Transients

Transients during the low-temperature operating mode are more severe when the RCS is operated in the water-solid condition. Addition of mass or energy to an isolated water-solid system produces increased system pressure. The severity of the pressure transients depends upon the rate and total quantity of mass or energy addition. The choice of the limiting LTOP transients is based on evaluations of the potential transients. The most limiting transients initiated by a single operator error or equipment failure are:

a. An inadvertent safety injection actuation and one charging pump operation (mass addition)

b. A start of one reactor coolant pump (RCP) when a positive temperature difference (ΔT) between the steam generator secondary-side water and the reactor coolant exists (energy addition)

The most limiting transients are determined by the conservative analyses, which maximize mass and energy additions to the RCS. In addition, the RCS is assumed to be in a water-solid condition at the time of the transient; such a condition has been noticed to exist infrequently during plant operation because the operator is instructed to avoid water-solid conditions whenever possible.

The maximum ΔT for energy addition transient is assumed to be 139 °C (250 °F), which is greater than the value allowed by Subsections 3.4.6 and 3.4.7 in Technical Specifications (Chapter 16) during the LTOP mode. However, the operational procedures direct the operator to maintain the ΔT below approximately 11.1 °C (20 °F).

The analyses demonstrate that the SCS suction line relief valve (SI-179 or SI-189) provides sufficient pressure relief capacity to mitigate the most limiting LTOP events identified above.

5.2.2.2.2.2 Provision for Overpressure Protection

The LTOP pressure is defined to be the SCS suction line relief valve setpoint pressure adjusted to provide a margin to avoid lifting and to compensate for pressure measuring inaccuracies during plant normal operation. During heatup, the RCS pressure is maintained below the LTOP pressure until the RCS cold leg temperature exceeds the LTOP disable temperature. During cooldown, the RCS pressure is maintained below the LTOP pressure once the RCS cold leg temperature reaches the LTOP enable temperature.

An LTOP enable temperature is defined in BTP 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures." The definition is based on measuring the degree of protection provided by the LTOP system against violations of the pressure-temperature limitations (P-T Limits) in terms of the RT_{NDT} of the reactor vessel beltline material at either the 1/4t or 3/4t location, depending upon which P-T Limit curve is most limiting.

Given the implicit assumptions in the P-T Limit analysis described in Subsection 5.3.2.1.2, the 3/4t location is the controlling beltline P-T Limit for heatup, and the 1/4t location is controlling for cooldown. LTOP disable temperature for heatup and LTOP enable temperature for the controlling cooldown (isothermal) is specified in the PTLR.

The RCS is protected at elevated temperatures by the pressurizer POSRVs, which have a set pressure of 173.7 kg/cm^{2}A (2,470 psia) as indicated in Table 5.4.14-1.

Whenever the SCS suction line relief valves are aligned to the RCS to provide LTOP, an increase in RCS pressure above the maximum SCS alignment pressure of 31.6 kg/cm^{2}A (450 psia) will cause an LTOP transient alarm in the main control room to alert the operator that a pressure transient is occurring. Operator actions taken in response to an LTOP transient alarm are described in Subsection 5.4.7.2.6. Either SCS suction line relief valve will terminate inadvertent pressure transients occurring while RCS temperature is below the LTOP disable temperature. For temperatures above the LTOP disable temperature specified in the PTLR for heatup, overpressure protection is provided by the pressurizer POSRVs.

During cooldown, whenever the RCS cold leg temperature is below the LTOP enable temperature specified in the PTLR, the SCS suction line relief valves provide the necessary overpressure protection. If the SCS is not aligned to the RCS before the cold leg temperature is decreased below the LTOP enable temperature, an alarm will notify the operator to open the SCS suction line isolation valves. However, the SCS cannot be aligned to the RCS until the RCS pressure is below the SCS entry pressure.

The LTOP conditions described above are within the SCS operating range. Subsection 3.4.11 in Technical Specifications (Chapter 16) requires the SCS suction line isolation valves to be open when operating in the LTOP mode. This Technical Specification also provides reasonable assurance that appropriate action is taken if one or more SCS suction line relief valves are out of service during the LTOP mode of operation.

Either SCS suction line relief valve provides sufficient relief capacity to prevent any pressure transient from exceeding the controlling P-T Limit.

5.2.2.2.2.3 Administrative Controls

Administrative controls necessary to implement the LTOP provisions are limited to controls that are necessary to open the SCS suction line isolation valves.

During cooldown, when the temperature of the RCS is above that corresponding to the intersection of the controlling P-T Limit and the pressurizer POSRV setpoint, overpressure protection is provided by the pressurizer POSRVs, and no administrative procedural controls are necessary. RCS pressure is decreased to below the LTOP pressure before entering the low temperature region for which LTOP is necessary. The LTOP pressure is equal to the maximum allowable pressure for SCS operation. Once the SCS is aligned, no further specific administrative procedural controls are needed to provide reasonable assurance of proper overpressure protection. The SCS remains aligned whenever the RCS is at low temperatures and the reactor vessel head is secured or until an adequate vent has been established. The SCS suction line isolation valve position is indicated.

During heatup, the SCS suction line isolation valves remain open at least until the LTOP disable temperature is reached. Once the RCS temperature reaches the temperature corresponding to the intersection of the controlling P-T Limit and the pressurizer POSRV setpoint, overpressure protection is provided by the pressurizer POSRVs. The SCS can be isolated, and no further actions are necessary.

5.2.2.3 Flow Diagrams

The flow diagram, showing the pressurizer POSRVs and their discharge lines, is given in Figure 5.1.2-3. The flow diagram showing the in-containment refueling water storage tank (IRWST) is given in Figure 6.8-3. The main steam safety valves are discussed in Section 10.3.

The SCS suction line relief valves (SI-179 and SI-189), used to provide LTOP are addressed in Subsection 5.4.7 and shown in the SIS/SCS flow diagrams in Figure 6.3.2-1 and Table 5.2-3.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Pressurizer Pilot-Operated Safety Relief Valves

The pressurizer POSRVs are pilot-operated, forged-steel valves. These valves are mounted on the top of the pressurizer. Further description of these valves is provided in Subsection 5.4.14. A schematic drawing of the pressurizer POSRVs is given in Figure 5.4.14-1. Valve parameters are given in Table 5.4.14-1.

Open and closed indications of each POSRV are provided in accordance with the recommendations of TMI Action Plan Item II.D.3 in 10 CFR 50.34(f) (2)(xi).

5.2.2.4.1.1 Pressurizer Pilot-Operated Safety Relief Valves Operation

Four pressurizer POSRVs are connected to the top of the pressurizer by separate inlet lines. There are two main discharge lines to the IRWST. The steam from two POSRVs is discharged through one common discharge line. Each pressurizer POSRV provides the overpressure protection function with a main valve and two spring-loaded pilot valve assemblies.

Each spring-loaded pilot valve in the assembly consists of a motor-operated isolation valve, a spring-loaded pilot valve, a check valve, and a manual isolation valve. The spring-loaded pilot valve of each POSRV acts as a safety valve in the closed position during normal operation and opens automatically if the system pressure increases to the POSRV set pressure, thus opening the check valve and the main valve.

Motor-operated isolation valves are normally open but are manually closed by an operator to isolate the discharge from the POSRVs when the spring-loaded pilot valves fail to close. Manual isolation valves are normally open but are closed manually by an operator during maintenance or a setpoint test.

Each pressurizer POSRV inlet nozzle is designed to pass a maximum steam flow in consideration of the maximum pressure drop from the pressurizer to the POSRV.

5.2.2.4.1.2 Transients

The pressurizer POSRVs are designed to withstand the consequences of the design transients of Table 3.9-1 without failure or malfunction.

5.2.2.4.1.3 Environment

The pressurizer POSRVs are designed to operate in normal and accident conditions. Normal and accident environmental conditions refer to environmental categories B and A, respectively, and are addressed in Appendix 3.11A.

5.2.2.4.2 Main Steam Safety Valves

The main steam safety valves are direct-acting, spring-loaded, carbon steel valves. The valves are mounted on each of the main steam lines upstream of the main steam isolation valves outside the containment. The valve parameters are given in Table 5.4.14-2. Additional description of overpressure protection equipment and components for the main steam system is provided in Subsection 10.3.2.

5.2.2.4.2.1 Main Steam Safety Valve Operation

Main steam safety valve operation is characterized by a sharp opening at the set pressure. The sharp opening is produced by two stages of reactions working together to produce a continuous action. The initial lift is produced when the steam pressure under the disc exceeds the spring force. The escaping steam reacts against the upper guide ring and pushes the disc up to a high lift. The reaction of the deflected steam against the underside of the disc lifts it higher on an accumulation of pressure. The valve reaches a lift in excess of full bore lift within an accumulation of 3 percent above the set pressure. As the system pressure drops, the valve disc settles to a moderate lift and closes quickly after blowing down to a pressure within 5 percent of set pressure.

5.2.2.4.2.2 Transients

The main steam safety valves are designed to withstand the consequences of the design transients of Table 3.9-1 without failure or malfunction.

5.2.2.4.2.3 Environment

The main steam safety valves are designed to operate in the normal and accident conditions. Normal and main steam line break conditions refer to environmental categories L and M in Table 3.11-2, respectively.

5.2.2.4.3 Shutdown Cooling System Suction Line Relief Valves (SI-179 and SI-189)

The SCS suction line relief valves that are used to provide LTOP are self-actuating, spring-loaded liquid relief valves with sufficient capacity to mitigate the most limiting overpressurization event. Control circuitry is not required because the valves open when RCS pressure exceeds the setpoint.

5.2.2.4.3.1 Shutdown Cooling System Suction Line Relief Valves (SI-179 and SI-189) Operation

The relief valves installed in the two SCS suction lines (one per train) provide LTOP for RCS when the SCS is aligned to the RCS during plant cooldown and heatup operations. The valves open up to 100 percent at 110 percent pressure of the setpoint.

5.2.2.4.3.2 Transients

Transients are described in Subsection 5.2.2.2.2.1.

5.2.2.4.3.3 Environment

The SCS suction line relief valves are designed to operate in normal and accident conditions. Normal and accident environmental conditions refer to environmental categories B and A, respectively, and are addressed in Table 3.11-2.

5.2.2.5 Mounting of Pressure-Relief Devices

5.2.2.5.1 Configuration of Pressure Relief Devices

The POSRVs and MSSVs are mounted to provide overpressure protection for the primary side and secondary side, respectively.

The design basis for the assumed loads for the primary and secondary side pressure relief devices are described in Subsection 3.9.3.

The POSRVs are shown in Figure 5.4.14-1. They discharge to the IRWST.

5.2.2.5.2 Design Bases for Mounting of Reactor Coolant Pressure Boundary Pressure Relief Devices

The RCPB pressure relief devices are mounted and installed as follows:

a. Each discharge pipe is supported to transfer transient discharge load to the pressurizer or adjacent structure in order to stay within the allowable pressurizer nozzle loads.

b. There is no rigid restraint in the vertical directions. The piping system moves up or down with the pressurizer during heat up or cool down. The weight of the piping system is carried by spring supports.

c. Each POSRV discharge piping is supported so that forces and moments during operating plant conditions (normal, upset, emergency, and faulted) will not jeopardize the integrity of the valves, the inlet lines to the valves or the nozzles on the pressurizer.

d. Pipe breaks are postulated in the high-energy piping between the pressurizer nozzle and the valve flange joint. Pipe breaks are not postulated in the POSRV discharge piping (i.e., piping from the valve discharge flange to the IRWST). This piping is not considered high-energy piping because it operates for less than 2 percent of the normal plant operating time. Discharge piping is classified as moderate-energy piping. Through-wall cracks are not postulated because this piping is adequately separated from other safety-related active systems subject to impairment by such failures.

Dynamic analysis for seismic and valve discharge loadings are performed to verify the design of the support configuration.

5.2.2.5.3 Pressurizer Pilot-Operated Safety Relief Valve Analysis - Loading Criteria and Methods of Analysis

5.2.2.5.3.1 Loading Conditions

Loading combinations for upstream Class 1 piping is shown in Table 3.9-10.

5.2.2.5.3.2 Pressure

Pressure loading is in accordance with the applicable ASME Code. The design pressure is 175 kg/cm^{2}G (2,485 psig) for Class 1 piping and 49.21 kg/cm^{2}G (700 psig) for non-ASME Section III piping.

5.2.2.5.3.3 Weight

A weight analysis is performed on the complete piping system. Hydrotest conditions are also considered in the analysis.

5.2.2.5.3.4 Seismic

The structural response due to the loadings is analyzed as discussed in Subsection 3.7.2.

5.2.2.5.3.5 Thrust

5.2.2.5.3.5.1 Hydraulic Thrust Forces

The POSRV and connecting piping are subjected to hydraulic thrust forces that are the result of changes in direction of the fluid and change in flow rate and from different fluid states such as vapor, liquid, and mixed vapor and liquid. The piping and valve supporting system is designed to withstand these thrust loads.

5.2.2.5.3.5.2 Structural Analysis of Thrust

The dynamic structural solution for the thrust loading is obtained using a modified predict-corrector integration technique and normal mode theory. PIPESTRESS (Reference 14) is used in the piping design (refer to Subsection 3.9.1.2.1).

RELAP5/MOD3.3 (Reference 15) is used to predict the transient flows resulting from the actuation of a POSRV. It also predicts the resulting piping loads used as dynamic forcing functions for structural design of discharge piping and its supporting components.

The computation is based on finite difference solutions by the method of characteristics. The computed transient pressure, velocity, and density are then used to calculate loads on bends and pipe runs.

5.2.2.5.4 Main Steam Safety Valve Analysis

5.2.2.5.4.1 Valve Forces and Reaction Load Paths

Two conditions are considered in the stress analysis of the safety valve installations: the dynamic effects of the safety valve opening and the steady-state flow condition reached after the valve has opened and is exhausting into the stack. During the valve opening period, dynamic forces due to the hydraulic transients in the valve are compensated by the reaction forces of the header supports through the header. When the valve has opened and the steam is exhausting into the stack, the discharge thrust is compensated by the reaction forces of the stack support structure.

5.2.2.5.4.2 Loading Conditions

Refer to Table 3.9-10 for loading combinations for piping upstream of the main steam safety valves.

5.2.2.6 Applicable Codes and Classification

The applicable codes and classifications for the overpressurization protection equipment are contained in Table 3.2-1. The applicable codes and classifications for the main steam safety valves are identified in Subsection 10.3.2.

5.2.2.7 Material Specifications

RCPB material specifications are described in Subsection 5.2.3.

5.2.2.8 Process Instrumentation

Process instrumentation for the overpressure protection equipment associated with the RCS is shown on Figure 5.1.2-3 and described in Chapter 7. Instrumentation associated with the pressurizer relief discharge is described in Subsection 6.8.3.

Open and closed indications of each POSRV are provided in accordance with the recommendations of TMI Action Plan Item II.D.3 in 10 CFR 50.34(f) (2)(xi).

The SCS suction line relief valves, isolation valves, associated interlocks, and instrumentation are addressed in Subsections 3.2.1 and 5.4.7.2.4 and Table 3.2-1. The interlocks and instrumentation associated with the SCS suction line isolation valves are addressed in Subsections 5.4.7, 7.6.1, and 7.6.1.1 and Table 7.6-1.

5.2.2.9 System Reliability

The pressurizer POSRVs are pilot-operated mechanisms and cannot fail closed if the setpoint pressure is exceeded. The force of the spring on the main valve disc area is designed to resist the inward pressure force of 2.0 kg/cm^2 D (29 psid). The minimum RCS pressure must be higher than or equal to 2.0 kg/cm^2 D (29 psid) for the main valve to remain open. In addition, the minimum RCS pressure required to open the main valve from the closed position is 10.2 kg/cm^2 D (145 psid).

The spring-loaded pilot valves are designed in accordance with the requirements of ASME Section III. Two spring-loaded pilot valves in parallel provide single-failure protection against failure to open the pressurizer POSRVs. A motor-operated isolation valve installed upstream of each spring-loaded pilot valve and operated by electrical actuator with normal and safety grade powers provides protection against failure to close the pressurizer POSRVs. In addition, all pilot valves and main valves have temperature indicators in the discharge line to detect a leak in any valve. The indicators show inadvertent openings of the pressurizer POSRVs. The operator can close the valves by actuating the motor-operated isolation valves. Section 15.6.1 presents an analysis of the inadvertent opening of a POSRV.

Reasonable assurance of the operational reliability of the pressurizer POSRVs is provided by:

a. Conformance with ASME OM Code for safety valves

b. Conservative design criteria (actual pressurizer POSRV steam flow exceeds the analysis flow rate)

c. Testing and surveillance prescribed by the Technical Specifications

The main steam safety valves have active and spring-actuated mechanisms that cannot maintain a closed position when the setpoint pressure is exceeded. Reasonable assurance of the operational reliability of the main steam safety valves is provided by the same methods as the pressurizer POSRVs.

The SCS suction line relief valves (SI-179 and SI-189) are independent of a loss of offsite power. No single failure of an isolation valve or its associated interlock prevents one relief valve from performing its intended function. If the SCS suction line relief valves are not aligned to the RCS before the cold leg temperature is reduced to below the maximum RCS cold leg temperature requiring LTOP, an alarm notifies the operator to open the SCS suction line isolation valves (SI-651, SI-652, SI-653, and SI-654). Each SCS suction line relief valve is a self-actuating, spring-loaded liquid relief valve that does not require control circuitry. Reasonable assurance of the SCS suction line relief valves is provided by design according to ASME Section III-NC, conservative design criteria, and periodic testing of the Technical Specifications.

5.2.2.10 Testing and Inspection

Preservice testing of the pressurizer POSRVs and main steam safety valves includes testing as specified in Chapter 14. The testing and inspection requirements are in conformance with ASME OM Code and ASME Section XI including the recommendations of TMI Action Plan Item II.D.1 in 10 CFR 50.34(f)(2)(x). The tests listed below are performed for preservice and inservice testing of the pressurizer POSRVs. No separate inservice testing requirements are to be applied to the operability of the POSRV check valve because it is actuated automatically by a spring-loaded pilot valve, and the POSRV check valve operability can be verified by testing the main valve as follows:

a. Opening setpoint verification test for the spring-loaded pilot valves

b. Operability (including opening time) test for the main valves by use of the spring-loaded pilot valves

c. Operability test for the motor-operated isolation valves

d. Seat leak tight test for each valve

e. Operability test for the position indication of each valve

f. Closing setpoint verification test for the spring-loaded pilot valves

g. Operability (including closing time) test for the main valves by the use of spring-loaded pilot valves

Inservice inspection and testing of the main steam safety valves are governed by ASME OM Code and ASME Section XI.

Preservice and inservice testing of the SCS suction line relief valves (SI-179 and SI-189) setpoint are performed according to the ASME OM Code.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specification

This subsection describes material issues common to the reactor coolant pressure boundary (RCPB) components. RCPB materials are fabricated in accordance with the requirements of GDC 1 and GDC 30 of 10 CFR Part 50, Appendix A; NRC RG 1.84; and 10 CFR 50.55a.

A list of specifications for the principal ferritic materials, austenitic stainless steels, bolting and weld materials, which are part of the RCPB, is given in Table 5.2-2. The COL applicant is to address the list of material specifications, which are not shown in Table 5.2-2, as necessary (COL 5.2(4)). The materials used in the RCPB meet the applicable material requirements of ASME Section III and conform to the applicable ASME Section II (Reference 16) material specifications or ASME Code Cases permitted or approved by the NRC. The COL applicant is to address the addition of ASME Code Cases that are approved in NRC RG 1.84 (COL 5.2(1)).

Austenitic stainless steel base materials for RCPB applications are solution-heat-treated to prevent sensitization and primary water stress corrosion cracking (SCC). Alloy 600 base metal and Alloy 82/182 weld metal are not used in RCPB applications. Only alloy 690 base metal and Alloy 52/52M/152 weld metals are used for RCPB applications. Alloy 690 base metals are thermally treated to enhance their resistance to primary water stress corrosion cracking (PWSCC).

All carbon and low alloy steel materials including weld materials used within the RCPB are limited to maximum sulfur (S) content of equal to or less than 0.010 wt%.

Studies referenced in NRC RG 1.99, Revision 02 (Reference 17), have shown that irradiation-induced mechanical property changes of SA-508 materials can depend significantly on the amount of residual elements present in the compositions (i.e., copper, nickel, phosphorous, and vanadium). Residual sulfur has also been found to affect the initial toughness of SA-508 materials. Controls are placed on the residual chemistry of RV materials and the as-deposited welds used to join these materials to limit the maximum predicted increase in the RT_{NDT}, which is described in Subsections 5.3.1.6 and 5.3.2.1.1, and to limit the extent of the RV beltline. The beltline is defined in Appendix G of 10 CFR Part 50 (Reference 18).

SA-508 Grade 3, Class 1 material, which is used in the RV beltline and the as-deposited welds, contains no greater than the following weight percent of residual elements:

Copper (in welds)	0.05	Phosphorous	0.012
Copper (in forgings)	0.03	Sulfur	0.010
Nickel (in forgings)	1.00	Vanadium	0.030
Nickel (in welds)	0.10		

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Reactor Coolant Chemistry

Controlled water chemistry is maintained within the RCS. RCS water chemistry is specified to minimize corrosion. RCS water chemistry specification is shown in Table 5.2-5. Water chemistry limits are determined at a level comparable to the guidelines in the Electric Power Research Institute's (EPRI's) "PWR Water Chemistry Guidelines." The COL applicant is to specify the version of EPRI's "Primary Water Chemistry Guidelines" (References 19, 20) that are to be implemented (COL 5.2(5)). Control of the reactor coolant chemistry is a function of the chemical and volume control system (CVCS), which is addressed in Subsection 9.3.4.

The CVCS is designed to perform the following functions:

a. Maintain the chemistry including pH and purity of the reactor coolant during prestartup testing, startups, normal operation, and during shutdowns

b. Scavenge oxygen from the coolant before heat-up

c. Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during normal operation

Two chemicals (hydrazine and hydrogen) are added to the reactor coolant to control dissolved oxygen (DO). Hydrazine is maintained in the reactor coolant at 1.5 times of DO concentration whenever the reactor coolant temperature is below 65.6 °C (150 °F). At power operation, DO concentration is limited by maintaining the excess dissolved hydrogen in the coolant. To minimize the effect of crud deposition on the reactor core heat transfer surfaces, lithium-7 hydroxide is added. Lithium-7 hydroxide produces pH conditions within the reactor coolant at operating temperatures that reduce the corrosion product solubility and hence the dissolved crud inventory in the circulating reactor coolant. The lithium concentration is maintained as shown in Tables 9.3.4-1B and 9.3.4-1C. Subsection 9.3.4 provides additional information on the water chemistry limits applicable to the RCS. For example, information on the control of suspended solid and demineralizer performance is described in Subsections 9.3.4.2.8.5 and 9.3.4.2.7, respectively.

A soluble zinc compound (Zn-64 < 1.0 wt%) may be added to the reactor coolant for the purpose of radiation field reduction and mitigation of PWSCC initiation.

5.2.3.2.2 Materials of Construction Compatibility with Reactor Coolant

The construction materials used in the RCPB that are in contact with the reactor coolant are designated in Table 5.2-2. These materials are selected to minimize corrosion and have demonstrated satisfactory performance in existing operating reactor plants. The materials used for the RCPB conform with the requirements of GDC 4 of 10 CFR Part 50, Appendix A. Conformance of the fabrication and processing of austenitic stainless steels with NRC RG 1.44 (Reference 21) is shown in Subsection 5.2.3.4.1. Stainless steel or nickel-chromium-iron cladding is applied for corrosion resistance to all ferritic low-alloy and carbon steel surfaces that come into contact with the reactor coolant.

The joints between the austenitic safe ends and low alloy or carbon steel nozzles are made by welding with Alloy 690 equivalent weld materials. Austenitic stainless steel and A690 base materials that are used for primary pressure-retaining applications are supplied in the solution-annealed and thermally treated condition, respectively.

Cobalt content is restricted to as low a level as practicable in metallic materials that are in contact with reactor coolant and that are in stainless steel or nickel-based alloy components with a large wetted surface area. Cobalt-based alloys are avoided except when no proven alternative exists.

Spiral wound gaskets, which contain nonmetallic material, are used for primary manways of steam generators and pressurizer. Generally, the spiral wound gasket is made of flexible graphite and Inconel X-750. Field experience shows that the gaskets have compatibility with reactor coolant.

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

All metallic insulation used in the plant is stainless steel reflective, which minimizes insulation contamination in the event of chemical solution spillage. All non-metallic insulation used in the plant is designed to meet the requirements of NRC RG 1.36 (Reference 22), "Non-metallic Thermal Insulation for Austenitic Stainless Steel." Conforming with the NRC RG provides reasonable assurance that non-metallic insulation is designed in a manner that minimizes the potential for stress corrosion of stainless steel due to leaching of chloride or fluoride ions onto the stainless steel surfaces.

5.2.3.3 Fabrication and Processing of Ferritic Materials

Fracture toughness requirements for reactor coolant pressure boundary components are established in accordance with ASME Section III and NRC SRP BTP 5-3 (Reference 23). Fracture toughness testing of base, weld and heat-affected zone materials is conducted in accordance with the ASME Code. Data from these tests are available after the required testing has been performed and are examined upon request at the appropriate manufacturing facility. The RCPB also conforms with the requirements of GDC 14 and 31 of 10 CFR Part 50, Appendix A, and 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

Consideration is given to the effects of irradiation on material toughness properties in the core beltline region of the reactor vessel to provide reasonable assurance of adequate fracture toughness for the service life of the vessel. Subsection 5.3.1.6 addresses the prediction of irradiation effects and the material surveillance program.

Testing and measuring equipment for fracture toughness tests for the reactor vessel, steam generators, pressurizer, piping, and reactor coolant pumps are calibrated in accordance with ASME Section III, NB-2360.

The COL applicant is to address the actual, as-procured, fracture toughness data of the RCPB materials to the staff at a predetermined time by an appropriate method (e.g., ITAAC) (COL 5.2(6)).

Welding of ferritic materials conforms with the recommendations of NRC RG 1.50 (Reference 24), "Control of Preheat Temperature for Welding of Low Alloy Steel," as described below.

Paragraph C.1. of NRC RG 1.50 implies that the qualification materials are an infinite heat sink that would instantaneously dissipate the heat input from the welding process. The qualification procedure consists of starting the welding at the minimum preheat temperature. Welding is continued until the maximum interpass temperature is reached. The test material is then permitted to cool to the minimum preheat temperature and the welding is restarted. Preheat temperatures utilized for low alloy steel are in accordance with ASME Section III Appendix D (Reference 25). Generally, the maximum interpass temperature utilized is 260 °C (500 °F). The minimum preheat temperature and maximum interpass temperature to be specified in the welding procedure specification are equal to the temperatures used during procedure qualification.

Hydrogen is removed either by maintaining preheat until post-weld heat treatment is performed or by post heating at a temperature and time sufficient to preclude the effects of hydrogen assisted cracking if post-weld heat treatment is required by ASME Code.

In the event that Paragraphs C.1., C.2., and C.3. of NRC RG 1.50 are not met, the soundness of the weld is demonstrated by an examination that meets the acceptance criteria specified in ASME Section III.

With regard to NRC RG 1.43 (Reference 26), major components are fabricated with corrosion resistant cladding on internal surfaces exposed to reactor coolant. The major portion of the material protected by cladding from exposure to reactor coolant is SA-508 Grade 3, Class 1, which is resistant to underclad cracking.

NRC RG 1.34 (Reference 27) recommends that control be applied during welding using the electroslag process. The electroslag process is not used in the fabrication of any RCPB components. Therefore, the recommendations in NRC RG 1.34 are not applicable.

The APR1400 conforms with the recommendations of NRC RG 1.71 (Reference 28) for the performance qualifications of personnel welding under conditions of limited accessibility.

The nondestructive examination requirements for tubular products are specified by ASME Section III. For major components, additional testing and inspection are explained in Subsections 5.3.1.3, 5.4.1.4, 5.4.2.3, 5.4.3.4, and 5.4.10.4.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

5.2.3.4.1 Avoidance of Stress Corrosion Cracking of Nuclear Steam Supply System Components

Fabrication of RCPB components is consistent with the recommendations of NRC RG 1.44, as described below, except for the criterion used to demonstrate freedom from sensitization. ASTM A262 (Reference 29) Practice A or E is used to demonstrate freedom from sensitization in fabricated, unstabilized stainless steel.

a. Solution heat treatment requirements

   All raw austenitic stainless steel material, both wrought and cast, used in the fabrication of the major NSSS components in the RCPB is supplied in the annealed condition as specified by the pertinent ASME Code.

   Solution heat treatment is not performed on completed or partially fabricated components. Rather, the extent of chromium carbide precipitation is controlled during all stages of fabrication as described in Items b, c, and d below.

b. Material inspection program

   Extensive testing on stainless steel mockups, fabricated using production techniques, has been conducted to determine the effect of various welding procedures on the susceptibility of unstabilized Type 300 series stainless steels to sensitization induced intergranular corrosion. Only the procedures and/or practices demonstrated not to produce a sensitized structure are used in the fabrication of RCPB components. ASTM A262 Practices A or E is the criterion used to determine susceptibility to intergranular corrosion. The test has shown excellent correlation with a form of localized corrosion peculiar to sensitized stainless steels. As such, ASTM A262 Practice A or E is used as a go/no-go standard for acceptability.

As a result of the above test, a relationship was established between the carbon content of Type 304 stainless steel and weld heat input. This relationship is used to avoid weld-heat-affected-zone sensitization as described in Item d below.

c. Unstabilized austenitic stainless steel

The unstabilized grades of austenitic stainless steels with carbon content of more than 0.03 percent used for components of the RCPB are Type 304 and Type 316. These materials are furnished in the solution-annealed condition. Completed or partially fabricated components are not exposed to temperatures from 427 °C (800 °F) to 816 °C (1,500 °F).

Duplex, austenitic stainless steels containing a certain quantity of delta ferrite (weld metal, cast metal, weld deposit overlay) are not considered unstabilized because these alloys do not sensitize, meaning they do not form a continuous network of chromium-iron carbides. Alloys in this category are:

CF8M, CF8	Cast stainless steel: delta ferrite 8 percent to 30 percent, 8 percent to 20 percent for normal operating temperature above 260 °C (500 °F), 14 percent maximum for static cast stainless steel of CF8M
Type 308, 309, 312, 316	Singly and combined stainless steel weld filler metals: delta ferrite controlled to 8FN-15FN (8FN-16FN for Type 309 (L)) with no reading below 5FN as deposited

In duplex, austenitic/ferritic alloys, chromium-iron carbides are precipitated preferentially at the ferrite/austenitic interfaces during exposure to temperatures ranging from 427 to 816 °C (800 to 1,500 °F). This precipitate morphology precludes intergranular penetrations associated with sensitized Type 300 series stainless steels exposed to oxygenated or fluoride environments.

d. Avoidance of sensitization

Exposure of unstabilized austenitic Type 300 series stainless steels to temperatures ranging from 427 to 816 °C (800 to 1,500 °F) results in carbide precipitation. The degree of carbide precipitation, or sensitization, depends on the temperature, the amount of time at that temperature, and the carbon content. Severe sensitization is defined as a continuous grain boundary chromium-iron carbide network. This condition induces susceptibility to intergranular corrosion in oxygenated aqueous environments, as well as those containing fluorides. Such a metallurgical structure rapidly fails the ASTM A262 Practice A or E Test. Discontinuous precipitates (i.e., an intermittent grain boundary carbide network) are not susceptible to intergranular corrosion in a PWR environment.

Weld-heat-affected-zone sensitized austenitic stainless steels are avoided by carefully controlling:

1) Weld heat input to less than 23.6 kJ/cm (60 kJ/in)

2) Interpass temperature to 176.7 °C (350 °F) maximum

3) Carbon content to 0.065 percent maximum

Homogeneous or localized heat treatment in the temperature range from 427 to 816 °C (800 to 1,500 °F) is prohibited for unstabilized austenitic stainless steel with a carbon content greater than 0.03 percent used in components of the RCPB. When stainless steel safe ends are required on component nozzles, fabrication techniques and sequencing require that the stainless steel piece be welded to the component after final stress relief. This is accomplished by welding a NiCrFe overlay on the end of the nozzle. Following final stress relief of the component, the stainless steel safe end is welded to the NiCrFe overlay, using NiCrFe weld filler metal.

5.2.3.4.2 Cleaning and Contamination Protection

Specific requirements for cleanliness and contamination protection are included in the equipment specifications for components fabricated with austenitic stainless steel. The provisions described below indicate the type of procedures used for NSSS components to provide contamination control during fabrication, shipment, and storage as required by NRC RG 1.28 (Reference 30).

Contamination of austenitic stainless steels of Type 300 series by compounds that can alter the physical or metallurgical structure and/or properties of the material is avoided during all stages of fabrication. Type 300 series stainless steels are not painted. Grinding is accomplished with resin or rubber-bounded aluminum oxide or silicon carbide wheels that were not previously used on materials other than austenitic alloys that could contribute to intergranular corrosion or SCC.

Outside storage of partially fabricated components is avoided and in most cases prohibited. Exceptions are made for certain components provided they are dry, completely covered with a waterproof material, and kept above ground.

Internal surfaces of completed components are cleaned to produce an item that is clean to the extent that grit, scale, corrosion products, grease oil, wax, gum, adhered or embedded dust, or extraneous materials are not visible to the unaided eye. Substances used for cleaning include solvents (acetone or isopropyl alcohol) and inhibited water (hydrazine or tri-sodium phosphate). Water conforms to the following requirements:

Chloride (ppm)	< 0.60
Fluoride (ppm)	< 0.40
Conductivity (μS/cm)	< 5.0
pH	6.0-8.0
Visual clarity	No turbidity, oil, or sediment

To prevent halide-induced intergranular corrosion, which can occur in an aqueous environment with significant quantities of dissolved oxygen, flushing water is inhibited by adding hydrazine or trisodium phosphate. Tests have shown these inhibitors to be effective. Operational chemistry specifications restrict concentrations of halide and oxygen, both precursors of intergranular attacks (refer to Subsection 9.3.4).

5.2.3.4.3 Characteristics and Mechanical Properties of Cold-Worked Austenitic Stainless Steels for RCPB Components

Cold-worked austenitic stainless steel is not used for components of the RCPB. The COL applicant is to submit the actual, as-procured yield strength of the austenitic stainless steel materials used in RCPB to the staff at a predetermined time agreed-upon by the regulatory body (COL 5.2(7)).

5.2.3.4.4 Control of Welding

NSSS components are designed as follows:

a. NRC RG 1.31 (Reference 31)

   In order to preclude microfissuring in austenitic stainless steel welds, RCPB components are consistent with the recommendations of NRC RG 1.31 as follows:

   The delta ferrite content of each lot and/or heat of weld filler metal used for welding of austenitic stainless steel code components is determined for each process to be used in production. Delta ferrite of consumable inserts, electrodes, rod, or wire filler metal used with the gas tungsten arc welding process, and deposits made with the plasma arc welding process may be determined by either of the alternative methods of magnetic measurement or chemical analysis described in ASME Section III.

   Delta ferrite content is verified for all other processes by tests using the magnetic measurement method on undiluted weld deposits described by ASME Section III. The average ferrite content shall meet the acceptance limits of 8FN to 15FN (8FN to 16FN for Type 309 (L)) with no reading below 5FN.

b. NRC RG 1.34

NRC RG 1.34 is addressed in Subsection 5.2.3.3.

c. NRC RG 1.71

NRC RG 1.71 is addressed in Subsection 5.2.3.3.

5.2.3.4.5 Toughness of Cast Austenitic Stainless Steels or Welds

Reasonable assurance of the fracture toughness of cast stainless steels is provided by limiting the delta ferrite in the materials as follows:

a. For normal operating temperatures less than or equal to 260 °C (500 °F): 8 percent to 30 percent

b. For normal operating temperatures above 260 °C (500 °F): 8 percent to 20 percent

c. Static cast stainless steel of CF8: 14 percent maximum

Reasonable assurance of the fracture toughness of stainless steel welds is provided by limiting the delta ferrite in the weld materials as follows:

a. Singly and combined stainless steel weld filler metals: 8FN-15FN (8FN-16FN for Type 309 (L)) with no reading below 5FN as deposited.

5.2.3.4.6 Nondestructive Examination

Nondestructive examinations of austenitic stainless steel tubular products for components of RCPB are carried out in accordance with ASME Section III, Subsection NB-2500, under their construction, and Section XI during inservice inspections. Additional testing and inspection for major components are explained in Subsections 5.3.1.3, 5.4.1.4, 5.4.2.3, 5.4.3.4, and 5.4.10.4.

5.2.3.5 Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Base Alloys

Thermally treated Alloy 690 (690TT) and Alloys 52/52M and 152 weld metals are used for the APR1400 design, which have performed well against PWSCC in field operations and laboratory experiments. Alloy 600 and Alloys 82/182 are not used. The resistance of Alloy 690, 52/52M, and 152 to PWSCC in pressurized water reactors is described in EPRI Report MRP-111, "Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors" (Reference 32). At present, there were no reports of the cracking of Alloy 690 base and weld metals. A boric acid corrosion (BAC) prevention program or an ISI program is applied to provide reasonable assurance of the integrity of 690TT base and weld metals.

5.2.3.6 Threaded Fasteners

Pressure-retaining threaded fasteners used for RCPB components are fabricated in accordance with ASME Section III, Subsection NB. A description of the design of threaded fasteners except for the reactor vessel stud bolts and nuts is provided in Section 3.13.

The stud bolting material of the APR1400 reactor vessel satisfies NRC RG 1.65 (Reference 33). Nondestructive examination of the stud bolting material is performed according to NB-2580 of ASME Section III. Other mechanical tests such as tensile and Charpy V-notch (CVN) impact tests are performed for the threaded fastener materials. More information is described in Subsection 5.3.1.7.

Threaded fasteners for the other pressure retaining parts of Class 1 components are made of SA-540 Grade B24, SA-193 Grade B7, or SA-194 Grade 4 or 7 low-alloy steel bolting materials. SB-637 N07718 precipitation hardening nickel-based alloy bolting is also used for pressure retaining parts of Class 1 components. CVN tests are performed to confirm that the materials conform with ASME Section III NB-2330 fracture toughness requirements, if required by the Code. Actual fracture toughness test results are provided to the NRC staff at a predetermined time.

5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

5.2.4.1 Inservice Inspection and Testing Program

The inservice inspection (ISI) and testing program for quality Group A components of the RCPB (ASME Section III, Class 1 components) conforms with the guidelines of 10 CFR 50.55a and GDC 32 of 10 CFR Part 50, Appendix A. The program reflects the principles and intent of ASME Section XI and OM Code. The purpose of the inservice inspection program is to periodically monitor the systems or components requiring inservice inspection in order to identify the systems or components that do not meet acceptance standards and to make the necessary repairs.

The COL applicant is to prepare the inservice inspection and testing program. The COL applicant is to provide and develop the implementation milestones for the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a (COL 5.2(8)).

The ISI and inservice testing (IST) programs consist of the following three subprograms:

a. The component inspection program, which includes nondestructive inspection of major components, piping system and support system

b. The pump and valve inservice testing program, which requires operability testing of selected pumps and valves

c. The hydrostatic testing program

The NSSS design provides reasonable assurance that the reactor coolant pressure boundary has an accessibility to perform the preservice and inservice inspections. The pump and valve inservice testing is described in Subsections 3.9.6.2 and 3.9.6.3.

5.2.4.1.1 System Boundary Subject to Inspection

The reactor pressure vessel, pressurizer, primary side of the steam generator and associated piping, pumps, valves, bolting, and component supports are subjected to inspection.

The high-energy system piping between containment isolation valves receives an augmented ISI as described in Subsection 6.6.8.

5.2.4.1.2 Arrangement of Systems and Components to Provide Accessibility

Accessibility to equipment for maintenance, testing, and inspection is a basic element of the APR1400 design process. The layout and arrangement of the plant provide adequate working space and access for inspection and for repair and maintenance of specific areas of Class 1 components of the RCPB in accordance with ASME Section XI IWA 1500. All Class 1 components shall be designed for and provided with access to enable the performance of ASME Section XI inspections in the installed conditions. Systems and components are designed such that design, materials, and geometry do not restrict inspections required by ASME Section XI.

The COL applicant is to address the accessibility of Class 1 components for ISI if the design of the APR1400 Class 1 component is changed from the DCD design (COL 5.2 (9)).

The provisions for access for examination of the RCPB are as follows:

a. Reactor vessel and closure head

1) From inside the vessel:

All internals of the reactor vessel, which is an open structure offering insignificant impediment to access, are removable making the entire inner surface of the vessel, including the beltline welds and the weld zones of the internal load-carrying structure attachments, available for the required surface and volumetric inspections. Provisions are made in the plant design to allow for the removal and storage of all vessel internals (except the flow skirt) during inservice inspection. The surveillance capsules assemblies are located to allow an essentially 100 percent ultrasonic testing for reactor vessel circumferential welds. For interim inspections of the vessel, nozzle-to-shell welds and inner radii of the outlet nozzles are accessible from inside the reactor vessel by using remote automated equipment without removal of the vessel internals.

2) From outside the vessel:

The bottom head of the reactor vessel is manually examined from the outside surface and an access tunnel is provided to allow personnel into the area below the bottom head. Insulation is provided by removable panels over the bottom head weld seams.

3) Closure head

The closure head is available for inspection when it is removed and its removal makes available the vessel closure flange, upper shell-to-intermediate shell weld, closure stud holes and ligaments, and closure studs and nuts. Each control element drive mechanism is removable as a unit through a closure at the top of its housing. Because many of the reactor vessel closure head examinations are conducted from the underside of the head, the head laydown area provides access for personnel to work under this component.

b. Reactor coolant piping

Biological shielding around the reactor coolant piping in the area of the reactor vessel is designed to afford access to the circumferential and longitudinal welds, as well as the transition piece-to-nozzle welds. The volumetric examinations are performed using ultrasonic techniques.

All reactor coolant piping, as well as major components, excluding the reactor vessel, is provided with removable insulation in the areas of all welds and adjacent base metal requiring examination.

The primary coolant piping has access at each side of the welds to facilitate manual examination of the welds.

c. Steam generators

Sufficient space is provided within the stay cylinder to permit inspection of the welds. A 305 × 406 mm (12 × 16 in) access opening in the steam generator support skirt is provided. The insulation in this area is removable through the support skirt opening.

The steam generators have removable insulation and access at welds requiring examination. Manways are provided for internal steam generator inspections. Accessibility is considered to enable the ultrasonic examination for the vessel transition welds of the steam generator.

d. Pressurizer

The pressurizer has sufficient clearance around the shell weld seams for manual ultrasonic examination of these welds. The insulation is removable at each weld and access is provided for ultrasonic and visual examinations in the area of the bottom head and its nozzle penetrations of the pressurizer. A manway is provided for internal inspections of the pressurizer.

e. Reactor coolant pumps

The reactor coolant pumps require inside visual examination. To allow this, provisions are made in the design for the removal and storage of the RCP motors and the disassembly of the reactor coolant pumps. Access is provided to the motor flywheels for ultrasonic examination.

f. Other components

All other components, including portions of the steam generators, reactor coolant pumps, pressurizer, and primary piping, are accessible for manual examination from the outside surface.

General provisions are made for removable insulation, removable shielding, installation of handling machinery, adequate personnel, and equipment access space, and laydown space for all temporarily removed or serviced components. Storage space for the removable insulation panels is also provided. Working room for personnel is provided adjacent to each weld in order to examine all piping system welds manually.

5.2.4.1.3 Examination Categories and Methods

Examinations include liquid penetrant, magnetic particle, or eddy current techniques when surface examination is specified; ultrasonic or radiographic techniques when volumetric examination is specified; and visual inspection techniques that are used to determine surface condition of components and for evidence of leakage. Specific techniques, procedures and equipment, including any special techniques or equipment are in accordance with the requirements of ASME Section XI and are defined in the inservice inspection program. Preservice inspection (PSI) and subsequent inservice inspection are conducted with equivalent equipment and techniques.

The visual, surface, and volumetric examination techniques and procedures agree with the requirements of Subarticle IWA-2200, IWB-2000, and Table IWB-2500-1 of ASME Section XI. The methods, procedures, and requirements for qualification of personnel performing ultrasonic examination are in accordance with the requirements of ASME Section XI, Appendix VII. The performance demonstration for ultrasonic examination procedure, equipment, and personnel used to detect and size flaws is in accordance with the requirements of ASME Section XI, Appendix VIII.

The data from the various baseline examinations, collected in accordance with the related procedures, are entered into a report with tabulated results. The report describes the scope of the inspection, the procedures utilized, the equipment utilized, names and qualifications of personnel, and all the examination results including all instrument calibration criteria in sufficient detail to provide reasonable assurance of repeatability for each examination. The categories and requirements appropriate for each examination area follow the categories and requirements specified in Table IWB 2500-1 of ASME Section XI. An inservice inspection program that includes the examination categories is provided in accordance with ASME Section XI, IWB 2000.

The methods, procedures, and requirements for ultrasonic examination of RV welds are qualified by performance demonstration in accordance with the requirements of ASME Section XI, Appendix VIII.

5.2.4.1.4 Inspection Intervals

The examination program for the 10-year inspection interval is defined in the ISI plan. The inspection period may be reduced or extended by as much as 1 year to enable an inspection to coincide with a plant outage. The ISI plan for all Class 1 systems and components is in accordance with ASME Section XI, IWA 2400 and IWB 2400.

5.2.4.1.5 Evaluation of Examination Results

Evaluation of examination results for Class 1 components is conducted in accordance with IWA 3000 and IWB 3000 of ASME Section XI. Unacceptable indications are repaired in accordance with the requirements of IWA 4000 of ASME Section XI. Criteria for establishing need for repair or replacement are in accordance with IWB 3000 of ASME Section XI.

5.2.4.1.6 System Pressure Tests

The leakage and hydrostatic pressure tests of the RCPB Code Class 1 components are conducted in accordance with the requirements of IWA 5000 and IWB 5000 of ASME Section XI and the Technical Specification. The requirements in the Technical Specifications (Chapter 16) on operating limits during heatup, cooldown, and system hydrostatic pressure testing are used for these tests.

5.2.4.1.7 Code Exemptions

The COL applicant is to provide a list of ASME Section XI Code exemptions in the ISI program of the specific plants, if it exists (COL 5.2(10)).

5.2.4.1.8 Relief from ASME Code Requirements

The COL applicant is to prepare and provide any requests for relief from the ASME Code requirements that are impracticable as a result of limitations of component design, geometry, or materials of construction for specific plants, if necessary. The request is to contain the information on applicable ASME Code requirements, alternative ISI methods, and justification (COL 5.2(11)).

5.2.4.1.9 Code Cases

The COL applicant may invoke ASME Code Cases listed in NRC RG 1.147 for the ISI program (COL 5.2(12)).

5.2.4.1.10 Other Inspection Program

The COL applicant is to prepare and implement a boric acid corrosion (BAC) prevention program in conformance with Generic Letter 88-05 (Reference 34)(COL 5.2(13)). The BAC program includes the selection of locations of degradation caused by small leakage, identification of small leakage locations, implementation methods of inspection and evaluation, and corrective action procedures for preventing recurrences of leakage.

5.2.4.2 Preservice Inspection and Testing Program

The preservice examination program is in accordance with the requirements of Article NB-5280 of ASME Section III, Division I. The preservice inspection (PSI) program conforms with the edition and addenda of ASME Section XI, as required by 10 CFR 50.55a(b). ASME Code Cases listed in NRC RG 1.147 that are incorporated by reference in 10 CFR 50.55a(b) are incorporated into the program as necessary.

The PSI program provides detailed information on areas subject to examination as well as methods, acceptance criteria, and extent of preservice examinations.

The COL applicant is to prepare the preservice inspection and testing program (COL 5.2(14)).

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The reactor coolant pressure boundary (RCPB) leakage detection systems provide a means for detecting and, to the extent practical, identifying the source of reactor coolant leakage and monitoring leaks from the reactor coolant and associated systems.

The RCPB leakage detection systems are designed in accordance with NRC RG 1.29 (Reference 35), "Seismic Design Classification," to identify the seismic classifications for the leakage detection systems.

The RCPB leakage detection systems conform to the guidance of NRC RG 1.45 (Reference 36), "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," regarding the detection, monitoring, quantifying, and identification of reactor coolant leakage.

5.2.5.1 Leakage Detection Methods

5.2.5.1.1 Unidentified Leakage

Indications of unidentified coolant leakage into the containment are provided by a containment sump level and flow monitor, an airborne particulate radioactivity monitor and an atmosphere humidity monitoring system.

In normal operation, these monitors show a background level that is indicative of the normal level of unidentified leakage inside the containment. Variations in airborne radioactivity or specific humidity above the normal level signify an increase in unidentified leakage rates and signal to the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage.

The sensitivity and response time of leakage detection equipment for unidentified leakage is such that a change in leakage rate, or its equivalent, of 1.89 L/min (0.5 gpm) can be detected in less than 1 hour.

The methods used to detect leakage to the containment from unidentified sources are:

a. Containment sump level

b. Containment airborne particulate radioactivity

c. Containment atmosphere humidity

Additionally, temperature and pressure monitoring of the containment atmosphere are used for alarms and indirect indication of leakage to the containment. They do not quantify the reactor coolant leakage.

5.2.5.1.1.1 Inventory Methods

Total leakage from the RCS can be determined by net level changes in the pressurizer and volume control tank over a measured period since the RCS and the chemical and volume control system represent a closed-loop system. Since the letdown flow and the reactor coolant pump seal controlled bleed off flow are collected and recycled back into the RCS by the CVCS, the net inventory in the RCS and CVCS under normal operating conditions is constant. Transient changes in letdown flow rate or RCS inventory are accommodated by changes in the volume control tank level. By monitoring reactor drain tank and equipment drain tank level changes within a given test period, the identified RCS leakage value can be determined. Subtracting the identified leakage value from the total leakage value (corrected for any RCS contraction) results in the RCS unidentified leak rate.

Makeup flow rate also provides a means of detecting leakage from the RCS through measurement of the net amount of makeup flow to the system. The net makeup to the system under no-leakage steady-state conditions is zero. The net makeup flow rate and the total makeup flow rate from the CVCS are continuously monitored and recorded. Analysis of the total makeup flow rate over a period of steady-state operation can determine the abnormal leakage. An increasing trend in the amount of required makeup indicates that the abnormal leakage exists and is increasing in rate. Sudden leakage induces a step increase in the amount of makeup, which does not decrease again. Sudden leakage would be the case in a purely transient condition.

5.2.5.1.1.2 Sump Level and Flow Method

Unidentified liquid leakages are routed to the containment drain sump, or incore instrumentation (ICI) cavity sump. The containment drain sump collects unidentified leakage from the containment floor drain piping. The ICI cavity sump collects unidentified leakage in the reactor cavity. Sump levels are monitored in the MCR and alarmed on high level. Additionally, the amount of leakage is calculated using the frequency and length of time the sump pumps operate. This system is addressed in NRC RG 1.45.

A change in leak rate greater than or equal to 1.89 L/min (0.5 gpm) is detectable within 1 hour, with an alarm actuating in the MCR to alert the operators, consistent with Regulatory Positions 2.2 and 3.3 of NRC RG 1.45.

The sump level monitoring system is qualified for a safe shutdown earthquake.

5.2.5.1.1.3 Containment Air Particulate Radioactivity Monitoring

Two containment air radiation monitors are installed in parallel to measure the unidentified RCPB leakage, as described in Subsection 11.5.2.2 and listed in Table 11.5-1. One of the two containment air radiation monitors takes continuous containment air samples and measures the particulate. High radiation alarms are displayed in the MCR as shown in Figure 11.5-1. Leakage of reactor coolant has the effect of increasing radioactive particles in the containment, thus an increase of radiation level is an indication of leakage. Radiological monitoring is addressed in more detail in Section 11.5.

The airborne particulate radiation monitors can detect a 1.89 L/min (0.5 gpm) leakage rate within 1 hour at full power operation. The sensitivity of the containment atmosphere radiation monitors is sufficient for detection of the limiting leakage.

The containment air radiation monitor is capable of functioning when subjected to a safe shutdown earthquake (SSE).

5.2.5.1.1.4 Other Methods

Consistent with NRC RG 1.45 Position 2.3, containment pressure, temperature, humidity monitoring, and an acoustic leak monitoring system (ALMS) are used as indirect indications of RCS leakage to the containment. Containment air pressure is continuously monitored and is indicated and alarmed in the main control room. The pressure is measured by independent pressure transmitters located at widely separated points within the containment. Refer to Section 7.5 for information on the display instrumentation associated with containment pressure. Temperature sensors are positioned at appropriate locations throughout the containment. Containment temperature is displayed in the MCR, along with high-temperature alarms. Finally, containment humidity sensors are also provided in the MCR with displays and high alarms.

The ALMS is described in Subsection 7.7.1.5. The ALMS monitors changes in acoustic levels above a normal background with no leakage. There are 19 locations selected based on the criticality of the component or region and in some cases where leakage has occurred. The indication is qualitative; it alerts operators to investigate and compare with other leakage monitoring methods.

5.2.5.1.2 Identified Leakage

Identified leakage is defined in accordance with the guidance of NRC RG 1.45 as follows: (1) leakage (such as pump seal or valve packing leakage) that is captured, flow-metered, and conducted to a sump, collecting tank, or collection system and (2) leakage into the containment atmosphere from a known source, which does not interfere with the operation of unidentified leakage monitoring systems and is not attributable to leakage in the RCPB.

The amount of identified leakage from the RCS can be determined by adding up the amounts from all identified paths described below. Indication and alarms associated with all of the identified leakage paths are provided in the MCR.

5.2.5.1.2.1 Pressurizer Pilot-Operated Safety Relief Valves

The pressurizer POSRVs, located at the top of the pressurizer, are routed to the IRWST. Valve leakage is monitored by resistance temperature detectors (RTDs) located on the discharge lines of each pressurizer POSRV and pilot valves. An abnormally high temperature in the discharge lines of the pressurizer POSRV and the pilot valves is an indication of valve leakage. A high temperature due to leakage is alarmed in the MCR. Position indication for each pressurizer POSRV is also provided in the MCR.

5.2.5.1.2.2 Reactor Coolant Pump Seals

Instrumentation is provided to detect abnormal seal leakage. The reactor coolant pumps are equipped with two-stage seals plus a vapor or backup seal as described in Subsection 5.4.1.2. During normal operation, the RCS pressure is decreased through the two seals to controlled bleed-off pressure.

The vapor or backup seal prevents leakage to the containment atmosphere and operates at sufficient pressure to direct the controlled bleed off to the volume control tank (VCT). The vapor or backup seal is designed to withstand the full RCS pressure in the event the two primary seals fail. The vapor seal pressure indicator would show a decrease in pressure upon a significant leak into the containment across the vapor seal and an increased level in the reactor drain tank (RDT) would be indicated. Seal leakage through the tubes of the reactor coolant pump seal coolers to the component cooling water system (CCWS) would be indicated by an increase in temperature of the component cooling water return line from the reactor coolant pumps, by an increased water level in component cooling water surge tanks, and by increased radiation monitor readings in the CCWS.

5.2.5.1.2.3 Valves

Valves inside the containment that isolate the RCS from connecting systems during normal operation are equipped with stem leakoffs at the stem packing. These leakoffs are connected by piping to the RDT. Valves in the charging and letdown system inside the containment are equipped with stem leakoffs.

5.2.5.1.2.4 Reactor Vessel Head Flange Leakage

Leakage between the two metal O-rings that seal the reactor vessel head flange is routed through a leakoff line to the RDT. A normally closed, remotely activated isolation valve and a pressure indicator are installed in the leakoff line. Leakage from the reactor vessel head flange will cause pressure in this leakoff line to rise. The pressure in the leakoff line is continuously monitored to detect the presence of a leak. Any leakage is bled off to the RDT by opening the isolation valve.

5.2.5.1.2.5 Leakage through Steam Generator Tubes or Tubesheet

N-16 radiation monitors are installed on each main steam line to reveal reactor coolant leakage through the steam generator tubes to the secondary side. An increase in radioactivity, as indicated by the condenser vacuum vent effluent monitor, and steam generator blowdown monitors will reveal reactor coolant leakage through steam generator tubes to the secondary side. Routine analysis of steam generator secondary water samples will also indicate leakage of reactor coolant into the secondary system. Appendix 11B describes in detail the methods used for primary-to-secondary leakage detection.

5.2.5.1.2.6 Leakage to Auxiliary Systems

Chapter 11 describes the design basis for process monitors used in all potentially contaminated auxiliary systems and their sensitivity.

5.2.5.2 Leakage Instrumentation in the Main Control Room

5.2.5.2.1 Pilot-Operated Safety Relief Valve

Indication of POSRV leakage is provided by the temperature instruments mounted on the POSRV discharge pipes and drain lines. The signal from the temperature sensor is continuously monitored and provides an alarm in the main control room via the information processing system (IPS) and qualified indication and alarm system (QIAS) displays.

5.2.5.2.2 Primary Indicators of Reactor Coolant Unidentified Leakage

The MCR indication of leakage from the RCS to the containment building is provided by indications and alarms. The primary indications of reactor coolant leakage are:

a. Containment air radiation indications as well as alarms that are initiated by abnormally high air particulate and radioactive gas levels in the containment atmosphere

b. Pumped flow from containment drain sump tank and reactor cavity sump

c. Levels of containment drain sump and reactor cavity sump

d. Acoustic leakage monitoring

5.2.5.2.3 Other Indicators of Reactor Coolant Leakage

Other main control room instrumentation that may indicate significant reactor coolant leakage includes:

a. Pressurizer level

b. Containment pressure, temperature, and humidity

c. Condenser vacuum vent radiation

d. Steam generator blowdown radiation

e. CVCS makeup system flow

f. Component cooling water radiation

g. RDT level, temperature, and pressure

h. Equipment drain tank pressure, temperature and level

i. VCT level

j. Reactor vessel O-ring leak-off pressure

k. Safety injection tank pressure and level

l. Safety injection line pressure

m. Pressurizer POSRV discharge temperature

n. Pressurizer POSRV position

o. Reactor coolant gas vent system pressure and temperature

p. Main steam line N-16

5.2.5.2.4 Leakage Conversion to Equivalent

Procedures for converting the instrument output to a leakage rate, in accordance with NRC RG 1.45, are available to the operators as described in the following subsections.

5.2.5.2.4.1 Containment Radioactive Air Particulate Monitoring

Upon actuation of a high activity alarm, the operator or computer performs the appropriate action. It may not be appropriate to estimate the RCS activity level based on particulate activity because the radioactivity value could vary as a result of operating conditions. Therefore, when the high-radiation alarms are occurred or the indication values have increased, a RCS inventory balance is performed and the quantitative measurement of the coolant unidentified leakage is performed. The measurement methods are described in Subsections 5.2.5.1.1.3.

5.2.5.2.4.2 Leakage to Containment Sumps

A computer is programmed to calculate leakage rates from the rate of change of sump level. Digital readouts are available to the operator.

A computer is also programmed to calculate leakage rates utilizing the frequency and interval of time the sump pumps operate in combination with the known pumping rate of the pumps. Digital readouts are available to the operator.

5.2.5.3 Maximum Allowable Total Leakage

The maximum allowable identified and unidentified leakage and the required instrument availability to detect leakage are stated in the Technical Specifications (Chapter 16).

5.2.5.4 Intersystem Leakage

5.2.5.4.1 Safety Injection System

The SIS is isolated from the RCS during normal operation. Leakage from the RCS to the SIS under normal operation is detected by SIS pressure increases and alarmed. For example, leakage into the safety injection tanks (SITs) by the two check valves isolating the tanks from the RCS is detected by an increase in pressure between the check valves. This pressure is indicated and alarmed in the MCR. Secondarily, the SIT level increases. The leakage rate is computed from the rate of change of the level.

5.2.5.4.2 Steam Generator Leakage

The detection of leakage across the steam generator boundary between the primary to secondary side is addressed in Subsection 5.2.5.1.2.5. Leakage across this boundary would be quantified, after the indication of radioactivity in the N-16 radiation monitors and the condenser vacuum vent effluent radiation monitor, by performing an RCS inventory balance. If the amount of leakage is small, chemical and radioisotope analyses of both the primary and secondary sides may be necessary to determine the leakage rate. Appendix 11B describes the methods that are used to detect primary-to-secondary leakage.

5.2.5.4.3 Shutdown Cooling System

The SCS is a closed system. Leakage from the RCS to SCS under normal operation, when the system is isolated from the RCS, would be detected by relief valve discharges.

5.2.5.4.4 Component Cooling Water System

The CCWS cools the reactor coolant pumps (RCPs), the SCS heat exchanger, the letdown heat exchanger, and the containment spray pump and SCS pump miniflow heat exchangers.

Leakage from the RCS to the CCWS is detected by the CCW radiation monitors and/or the CCW surge tank level. The change in surge tank level is utilized to quantify any leakage.

5.2.5.5 Sensitivity and Response Time

For unidentified leakage, the leakage detecting system is designed to detect a minimum of 1.89 L/min (0.5 gpm) in 1 hour. The sensitivity and response times meet Regulatory Positions 2.1 and 2.2 of NRC RG 1.45.

5.2.5.6 Operability Testing and Calibration

Leakage monitoring systems have provisions to permit calibration and testing during plant operation, as appropriate. Periodic testing of leakage detection systems is conducted to verify the operability and sensitivity of detection equipment. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks.

Periodic inspection of the floor drainage system to the containment sump is conducted to check for blockage and provide reasonable assurance of unobstructed pathways.

The containment humidity monitoring is also tested periodically to provide reasonable assurance of proper operation and verify sensitivity.

An inservice inspection (ISI) program for the examination of RCPB components and supports to periodically monitor the systems or components is described in Subsection 5.2.4.

5.2.5.7 Limits for Reactor Coolant Leakage Rates within the RCPB

The limiting conditions for identified, unidentified, RCPB, and intersystem reactor coolant leakages are identified in the Technical Specifications (Chapter 16). Subsections 3.4.12 and 3.4.13 of the Technical Specifications address RCS operational leakage and RCS pressure isolation valve, respectively. Subsection 3.4.14 addresses RCS leakage detection instrument requirements. The COL applicant is to address and develop the milestones for the preparation and implementation of the procedure for operator responses to prolonged low-level leakage (COL 5.2(15)).

5.2.6 Combined License Information

COL 5.2(1) The COL applicant is to address the addition of ASME Code Cases that are approved in NRC RG 1.84.

COL 5.2(2) The COL applicant is to address the ASME Code Cases invoked for the ISI program of a specific plant.

COL 5.2(3) The COL applicant is to address the Code Cases invoked for operation and maintenance activities.

COL 5.2(4) The COL applicant is to address the material specifications, which are not shown in Table 5.2-2, as necessary.

COL 5.2(5) The COL applicant is to specify the version of EPRI's, "Primary Water Chemistry Guidelines," that will be implemented.

COL 5.2(6) The COL applicant is to address the actual, as-procured, fracture toughness data of the RCPB materials to the staff at a predetermined time by an appropriate method.

COL 5.2(7) The COL applicant is to submit the actual, as-procured yield strength of the austenitic stainless steel materials used in RCPB to the staff at a predetermined time agreed-upon by the regulatory body.

COL 5.2(8) The COL applicant is to provide and develop the implementation milestones for the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a.

COL 5.2(9) The COL applicant is to address the provisions to accessibility of Class 1 components for ISI if the design of the APR1400 Class 1 component is changed from the DCD design.

COL 5.2(10) The COL applicant is to provide the list of Code exemptions in the ISI program of the specific plants, if it exists.

COL 5.2(11) The COL applicant is to prepare and provide any requests for relief from the ASME Code requirements that are impracticable as a result of limitations of component design, geometry, or materials of construction for the specific plants, if necessary. The request will contain the information on applicable Code requirements, alternative ISI method, and justification.

COL 5.2(12) The COL applicant may invoke ASME Code Cases listed in NRC RG 1.147 for the ISI program.

COL 5.2(13) The COL applicant is to prepare and implement a boric acid corrosion (BAC) prevention program in conformance with Generic Letter 88-05.

COL 5.2(14) The COL applicant is to prepare the preservice inspection and testing program.

COL 5.2(15) The COL applicant is to address and develop the milestones for the preparation and implementation of the procedure for operator responses to prolonged low-level leakage.

5.2.7 References

1. ANSI/ANS 51.1-1983, “American Nation Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants,” American Nuclear Society, 1983.

2. ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Facility Components,” The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

3. 10 CFR 50.55a, “Codes and Standards,” U.S. Nuclear Regulatory Commission.

4. Regulatory Guide 1.26, “Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Rev. 4, U.S. Nuclear Regulatory Commission, March 2007.

5. 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission.

6. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

7. ASME Boiler and Pressure Vessel Code, OM Code, “Code for Operation and Maintenance of Nuclear Power Plants,” The American Society of Mechanical Engineers, the 2004 Edition with the 2005 and the 2006 Addenda.

8. 10 CFR 52.47, "Contents of Applications; technical information," U.S. Nuclear Regulatory Commission.

9. Regulatory Guide 1.84, "Design, Fabrication and Materials Code Case Acceptability, ASME Section III," Rev. 36, U.S. Nuclear Regulatory Commission, August 2014.

10. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1", Rev. 17, U.S. Nuclear Regulatory Commission, August 2014.

11. Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code", Rev. 1, U.S. Nuclear Regulatory Commission, August 2014.

12. NUREG-0800, Standard Review Plan, BTP 5-2, "Overpressurization Protection of Pressurized - Water while Operating at Low Temperatures," Rev. 3, U.S. Nuclear Regulatory Commission, March 2007.

13. APR1400-Z-M-NR-14008-P, "Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown," Rev. 0, KHNP, November 2014.

14. DST Computer Service SA, "A nuclear and non-nuclear piping analysis program," PIPESTRESS Version 3.7.0, Geneva, Switzerland, 2012.

15. NUREG/CR-5535, "RELAP5/MOD3.3 Code Manual," Rev. 3, U.S. Nuclear Regulatory Commission, March 2006.

16. ASME Boiler and Pressure Vessel Code, Section II, "Materials," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

17. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Rev. 2, U.S. Nuclear Regulatory Commission, May 1988.

18. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission.

19. EPRI TR-105714-R5, "PWR Primary Water Chemistry Guidelines," Rev. 5, Electric Power Research Institute, March 2003.

20. EPRI 1014986, "PWR Primary Water Chemistry Guidelines," Rev. 6, Electric Power Research Institute, December 2007.

21. Regulatory Guide 1.44, “Control of the Processing and Use of Stainless Steel,” Rev. 1, U.S. Nuclear Regulatory Commission, March 2011.

22. Regulatory Guide 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,” Rev. 0, U.S. Nuclear Regulatory Commission, February 1973.

23. NUREG-0800, Standard Review Plan, BTP 5-3, “Fracture Toughness Requirements,” Rev. 2, U.S. Nuclear Regulatory Commission, March 2007.

24. Regulatory Guide 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” Rev. 1, U.S. Nuclear Regulatory Commission, March 2011.

25. ASME Section III, Appendix D, “Nonmandatory Preheat Procedures,” The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

26. Regulatory Guide 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,” Rev. 1, U.S. Nuclear Regulatory Commission, March 2011.

27. Regulatory Guide 1.34, “Control of Electroslag Weld Properties,” Rev. 1, U.S. Nuclear Regulatory Commission, March 2011.

28. Regulatory Guide 1.71, “Welder Qualification for Area of Limited Accessibility,” Rev. 1, U.S. Nuclear Regulatory Commission, March 2007.

29. ASTM A 262, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” American Society for Testing and Materials, 2010.

30. Regulatory Guide 1.28, “Quality Assurance Program Criteria (Design and Construction),” Rev. 4, U.S. Nuclear Regulatory Commission, June 2010.

31. Regulatory Guide 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” Rev. 4, U.S. Nuclear Regulatory Commission, October 2013.

32. EPRI Report MRP-111, “Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors.”, March 2004.

33. Regulatory Guide 1.65, “Materials and Inspections for Reactor Vessel Closure Studs,” Rev. 1, U.S. Nuclear Regulatory Commission, April 2010.

34. Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” U.S. Nuclear Regulatory Commission, March 17, 1988.

35. Regulatory Guide 1.29, “Seismic Design Classification,” Rev. 4, U.S. Nuclear Regulatory Commission, March 2007.

36. Regulatory Guide 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” Rev. 1, U.S. Nuclear Regulatory Commission, May 2008.

Table 5.2-1

Reactor Coolant System Pressure Boundary Code Requirements [1]

Components	Codes and Classes
RV, steam generators (primary side), pressurizer	ASME Section III, Nuclear Power Plant Components, Class 1
RCP (structural portions necessary to provide reasonable assurance of the integrity of the RCPB)	ASME Section III, Nuclear Power Plant Components, Class 1
RCP auxiliaries	ASME Section III, Nuclear Power Plant Components, Class 3
Pipe and valves	ASME Section III, Nuclear Power Plant Components, Class 1
Pressurizer spray and pilot-operated safety relief valves	ASME Section III, Nuclear Power Plant Components, Class 1
Steam generators (secondary side)	ASME Section III, Nuclear Power Plant Components, Class 2
Control element drive mechanisms (CEDMs)	ASME Section III, Nuclear Power Plant Components, Class 1
Primary component supports	ASME Section III, Nuclear Plant Component Supports, Class 1

(1) The codes listed in this table are construction codes.
The ASME Code of 2007 Edition with 2008 Addenda is applicable to the APR1400 for construction. In addition, the components listed in this table are designed and constructed to meet the test and inspection requirements of the ASME OM and ASME Section XI, Rules for Inservice Inspection, 2007 Edition with 2008 Addenda.

Table 5.2-2 (1 of 5)

Reactor Coolant System Materials and Weld Materials

Component	Material Specification
Reactor Vessel	
Forgings	SA-508 Grade 3 Class 1
Cladding (1)	Weld deposited austenitic stainless steel with 5FN- 18FN delta ferrite or NiCrFe alloy
Direct vessel injection (DVI) nozzle safe ends (1)	SA-182 Grade F316 or F316LN
RV head (1), (3) CEDM nozzles	NiCrFe Alloy 690 (SB-166)
Vessel internals(1)	Austenitic stainless steel and NiCrFe alloy
Flow skirt (1), (3)	NiCrFe Alloy 690 (SB-168)
Fuel cladding (1)	ZIRLO
Instrument nozzles (1), (3)	NiCrFe Alloy 690 (SB-166)
Closure head studs	SA-540 Grade B24 Class 3
Control Element Drive Mechanism Housings	
Lower(1)	Type 403 stainless steel according to Code Case N-4-12 with end fittings to be SB-166 (N06690)(3) and SA-182 Grade F347/F348 stainless steel
Upper(1)	SA-479 and SA-213 Type 316 stainless steel with end fitting of SA-479 Type 316 and vent valve ball seal of Type 440C stainless steel
Pressurizer	
Shell	SA-533 Type B Class 1 or SA-508 Grade 3 Class 1
Cladding (1)	Weld-deposited austenitic stainless steel with 5 FN-18FN delta ferrite or NiCrFe alloy
Forged nozzles	SA-508 Grade 3 Class 1
Instrument nozzles and heater sleeves (1), (3)	NiCrFe Alloy 690 (SB-166, SB-167 or SB-168)
Nozzle safe ends (1)	SA-182 Grade F316, F316LN, F316N or F347
Studs and nuts	SB-637 N07718

Table 5.2-2 (2 of 5)

Component	Material Specification
Steam Generator	
Primary head	SA-533 Type B Class 1 or SA-508 Grade 3 Class 1
Primary nozzles	SA-508 Grade 3 Class 1
Primary head cladding (1)	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite or NiCrFe alloy
Tubesheet	SA-508 Grade 3 Class 1 or Class 2
Divider Plate (1)	SA-240 Type 410S
Tubesheet stay	SA-508 Grade 3 Class 1 or Class 2
Tubesheet cladding (1)	Weld-deposited NiCrFe alloy
Tube (1), (3)	NiCrFe Alloy 690 (SB-163)
Tube supports	ASTM A240, Type 409
Secondary shell (4)	SA-533 Type B Class 1 or SA-508 Grade 3 Class 1
Secondary head (4)	SA-508 Grade 1, Grade 1a or Grade 3 Class 1, or SA-533 Type B Class 1
Secondary nozzles (4)	SA-508 Grade 1, Grade 1a, Grade 3 Class 1 or Grade 3 Class 2
Secondary nozzle safe ends (4)	SA-508 Grade 1 or 1a
Secondary instrument nozzles (4)	SA-106 Grade B, SA-333 Grade 6
Secondary studs and nuts	SA-540 Grade B24, or SA-193 Grade B7
Primary studs and nuts	SB-637 N07718

Table 5.2-2 (3 of 5)

Component	Material Specification
Reactor Coolant Pumps	
Casing	SA-508 Grade 3 Class 1 and clad with austenitic stainless steel
Cladding (1)	Weld-deposited austenitic stainless steel with 5FN-18FN delta ferrite
Internals (1)	SA-487 CA6NM, SA-336 Type 304, 304LN, 347 or austenitic stainless steel
Shaft (1)	SA-182 Grade F6NM
Reactor Coolant Piping	
Pipe (30 in and 42 in ID)	SA-516 Grade 70 or SA-508 Grade 1 or Grade 1a
Cladding (1)	Weld-deposited austenitic stainless steel with 5FN-18FN delta ferrite
Piping Nozzles and Safe Ends	
Nozzle forgings	SA-508 Grade 1, Grade 1a, or Grade 3 Class 1
Instrument nozzles (1), (3)	NiCrFe Alloy 690 (SB-166)
Nozzle safe ends(1)	SA-182 Grade F316, F316N, F316LN, F347, or NiCrFe Alloy 690 (SB-166)
Valves (1)	SA-351 CF8M or SA-182 Grade F316 or NiCrFe Alloy 690 (SB-166) F316LN
Surge line (1)	SA-312 TP347 or TP316N (piping) SA-403 WP347 or WP316N (elbows)
DVI and shutdown lines inside containment	SA-312 TP316 or TP304

Table 5.2-2 (4 of 5)

Base Material Type (5)	Base Material Type (5)	Type of Weld Material	Example of Use
Weld Materials for Reactor Coolant Pressure Boundary Components			
P-1	P-1	a. SFA 5.1 E-7018, E-7016 b. SFA 5.18 ER70S-6 c. SFA 5.23, EA-3(N)	Primary piping straight to primary piping elbows
P-1	P-3	a. SFA 5.1 E-7018, E-7016 b. SFA 5.5 E-8018-C3, E-8018-G, E-8016-G c. MIL-E-18193 B-4 d. SFA 5.23 EA3 e. SFA 5.18 ER70S-6	Primary piping straight to the RV primary nozzle
P-1	P-8	a. NiCrFe filler metal b. SFA 5.4 E309L-16 c. SFA 5.9 ER309L	Primary piping surge nozzle to surge nozzle safe end
P-1	P-43	NiCrFe filler metal	Buttering (NiCrFe filler metal) of J-grooves in hot leg pipe
P-3	P-3	a. SFA 5.5 (2) E-8016-C3, E-8018-G, E-8016-G b. MIL-E-18193 B-4 (2) c. SFA 5.23 EA3(2)	RV upper shell to RV flange
P-3	P-8	a. NiCrFe filler metal b. SFA 5.4 E309L-16 c. SFA 5.9 ER309L	POSRV nozzle to POSRV safe end
P-3	P-43	a. NiCrFe filler metal	Buttering (NiCrFe filler metal) of J-grooves in RV closure head
P-8	P-8	a. SFA 5.4 E308, E308L, E308L-16, E309, E309L-16, E316, E347 b. SFA 5.9 ER308, ER308L, ER309, ER309L, ER316, ER347	Surge line piping to surge line elbows

Table 5.2-2 (5 of 5)

Base Material Type (5)	Base Material Type (5)	Type of Weld Material	Example of Use
Weld Materials for Reactor Coolant Pressure Boundary Components (cont.)			
P-8	P-43	a. NiCrFe filler metal	Pressurizer instrument nozzles to pressurizer instrument nozzle safe ends
P-43	P-43	a. NiCrFe filler metal	RV CEDM nozzles to J-groove buttering (NiCrFe filler metal)
Stainless steel cladding (1)		a. SFA 5.4 E308, E308L, E308L-16, E309, E309L, E309L-16 b. SFA 5.9 ER308, ER308L, ER 309, ER309L c. SFA 5.22 E308LT1-1, E309LT1-1	-
Nickel alloy cladding (1)		a. NiCrFe filler metal	-

(1) Materials exposed to reactor coolant

(2) Special weld wire with low residual elements of copper, nickel and phosphorous as specified when used in the RV core beltline region

(3) Material to be provided in the thermally treated condition

(4) SG secondary side pressure boundary materials including weld materials contain no greater than 0.010 wt% of sulfur (S)

(5) P-number designations are per the ASME Section IX, Table QW-422

Table 5.2-3

SCS Suction Line Relief Valve Valves (SI-179 and SI-189) Design Parameters

Parameter	Value
Design pressure, kg/cm^2G (psig)	63.2 (900)
Design temperature, °C (°F)	204.4 (400)
Fluid	Reactor coolant
Nominal setpoint, kg/cm^2G (psig)	37.3 (530)[(1)]
Accumulation	10 %
Capacity, L/min (gpm)	29,337 (7,750) (@ 10 % accumulation)
Inlet line size, cm (in)	20.3 (8)
Outlet line size, cm (in)	25.4 (10)

(1) Pressure measured at the valve inlet

Table 5.2-4

ASME Section III Code Cases

Code Case	Title
N-4-13	Special Type 403 Modified Forgings or Bars, Section III, Division 1, Class 1 and CS
N-60-5	Materials for Core Support Structures, Section III, Division 1
N-71-18	Additional Materials for Subsection NF, Classes 1, 2, 3, and MC Component Supports Fabricated by Welding, Section III, Division 1
N-249-14	Additional Materials for Subsection NF, Class 1, 2, 3, and MC Component Supports Fabricated without Welding, Section III, Division 1
N-759-2	Alternative Rules for Determining Allowable External Pressure and Comprehensive Stress for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1

Table 5.2-5

Reactor Coolant Design Specification

Parameter	Limit	Remarks
pH	4.2 ~ 10.7 ppm	3.8~10.7 during cooldown (1)
Hydrazine	0 ~ 50 ppm	This specification applies when reactor coolant temperature is less than 65.6 °C during heatup. It also applies below 204.4 °C below during cooldown.
Ammonia	0 ~ 50 ppm	-
Lithium	0 ~ 3.5 ppm	-
Dissolved hydrogen	0 ~ 100 cc (STP) H_2/kg H_2O	-
Dissolved oxygen	0 ~ 0.1 ppm	This specification applies when the reactor coolant temperature is greater than 121.1 °C. The reactor coolant is air saturated during heatup with the coolant temperature below 121.1 and during cooldown with the coolant temperature below 65.6 °C.
Dissolved nitrogen	0 ~ 100 cc (STP) N_2/kg H_2O	-
Suspended solids	0 ~ 2.0 ppm	-
Chloride	0 ~ 0.15 ppm	-
Fluoride	0 ~ 0.15 ppm	-
Boron	0 ~ 2,500 ppm	4,400 ppm during cooldown (1)
Sulfate	0 ~ 0.15 ppm	-
Zinc	0 ~ 0.02 ppm	-

(1) The frequency and duration for cooldowns are as follows:

a) 40 cycles of 1 month’s duration

b) 260 cycles of 1 week’s duration

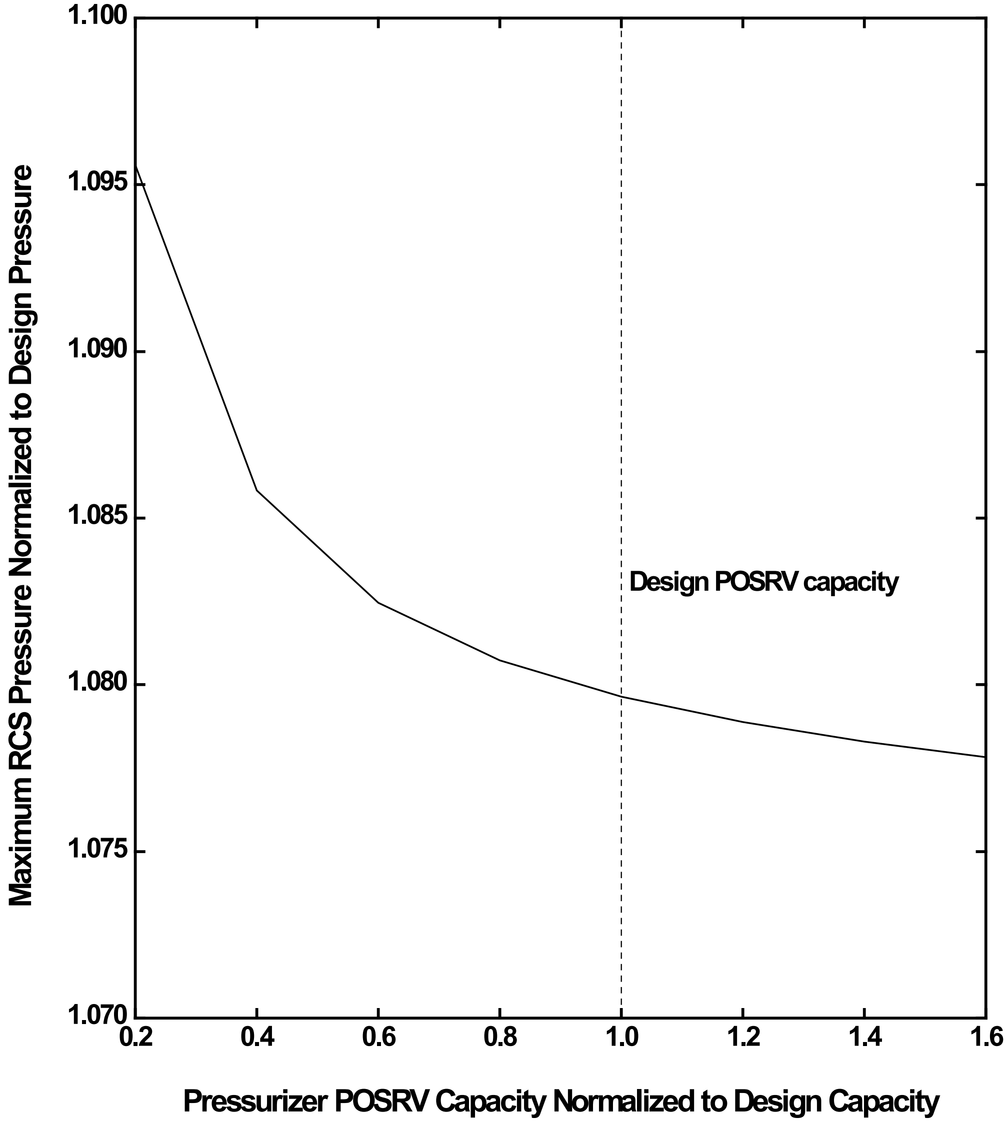

Figure 5.2.2-1 Optimized Pressurizer POSRV Capacity

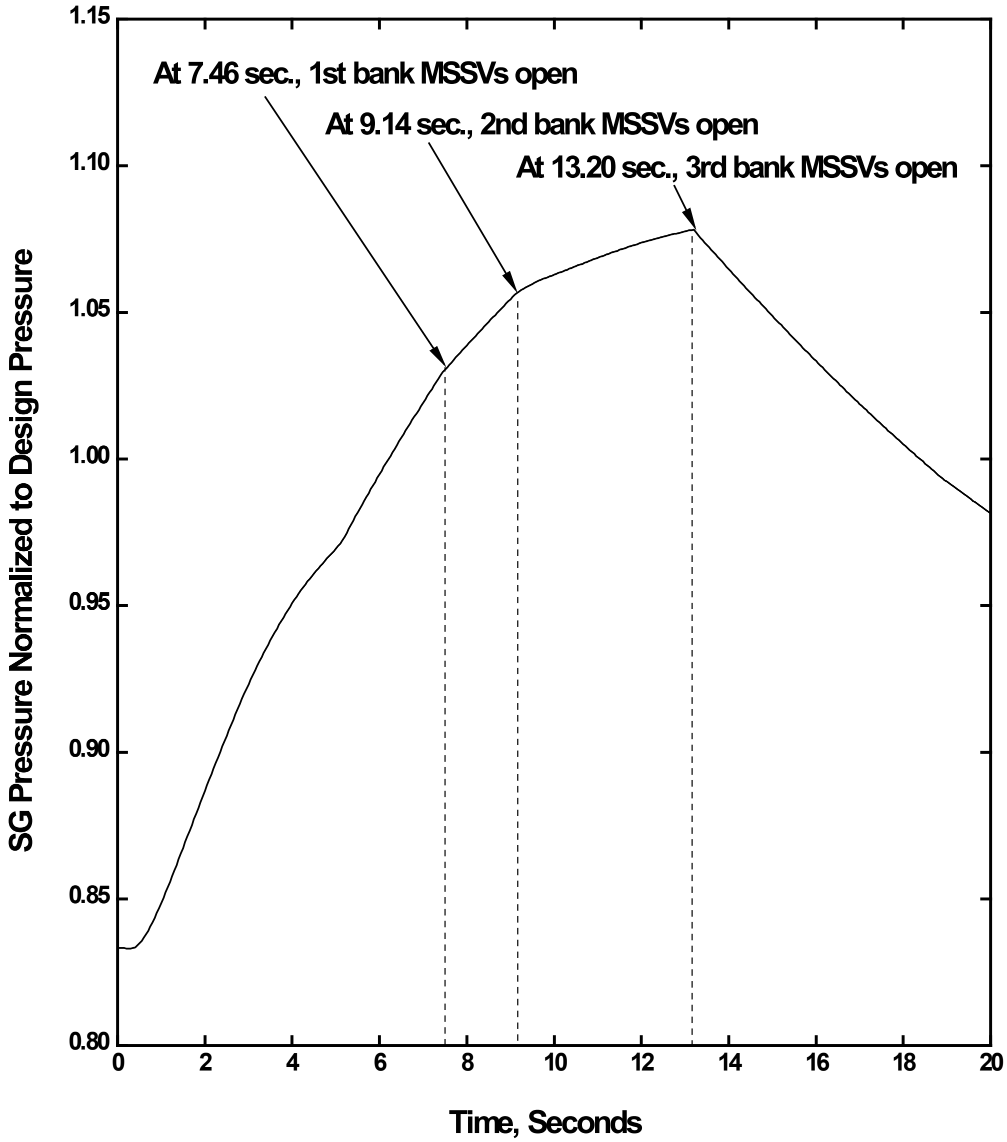

Figure 5.2.2-2 Steam Generator Pressure Normalized to Design Pressure vs. Time for the Worst-Case Loss-of-Load Event

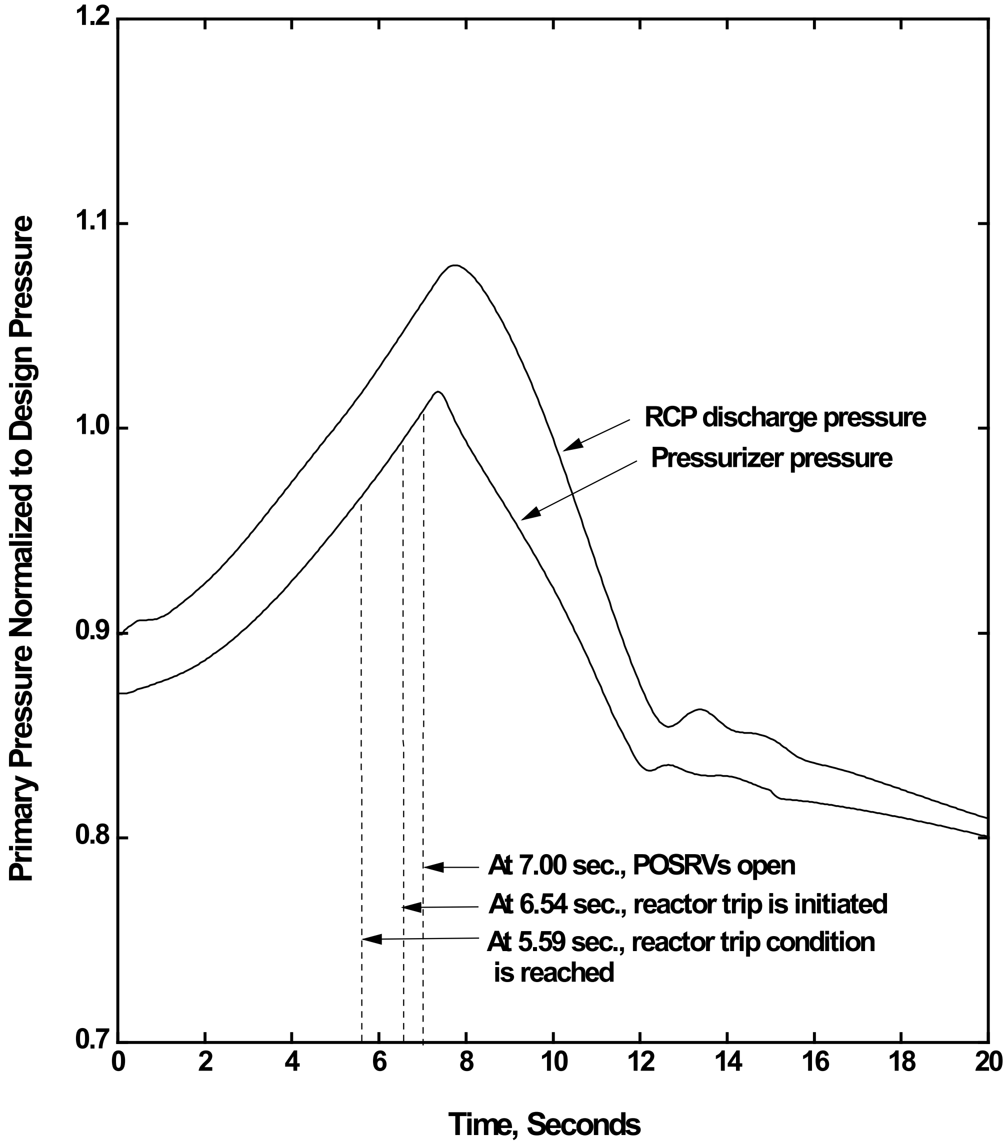

Figure 5.2.2-3 Primary Pressure Normalized to Design Pressure vs. Time for the Worst-Case Loss-of-Load Event

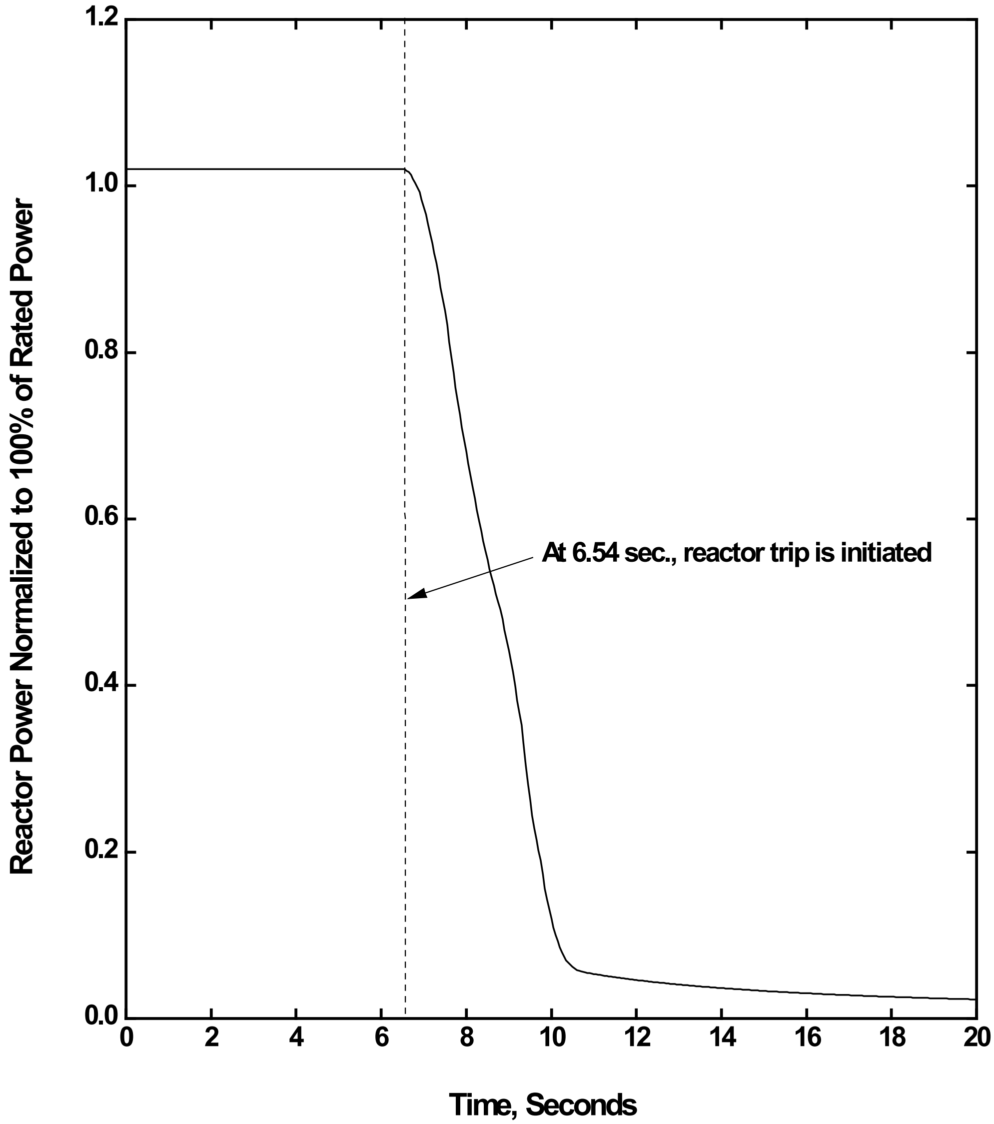

Figure 5.2.2-4 Reactor Power Normalized to 100 % of Rated Power vs. Time for the Worst-Case Loss-of-Load Event

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The APR1400 reactor vessel materials meet the following requirements:

a. General Design Criteria (GDC) 1 and 30 in Appendix A of 10 CFR Part 50 (Reference 1) as related to quality standards for design, fabrication, erection, and testing of structures, systems, and components

b. GDC 4 as related to the compatibility of components with environmental conditions

c. GDC 14 as related to the prevention of rapidly propagating fractures of the reactor coolant pressure boundary (RCPB)

d. GDC 31 as related to material fracture toughness

e. GDC 32 as related to the requirements for a materials surveillance program

f. 10 CFR 50.55a (Reference 2) as related to quality standards for design and determination and monitoring of fracture toughness

g. 10 CFR 50.60 (Reference 3) as related to RCPB fracture toughness

h. 10 CFR Part 50, Appendix B, Criterion XIII (Reference 4) as related to onsite material cleaning control

i. 10 CFR Part 50, Appendix G (Reference 5) as related to materials testing and acceptance criteria for fracture toughness

j. 10 CFR Part 50, Appendix H (Reference 6) as related to the determination and monitoring of fracture toughness

5.3.1.1 Material Specifications

The principal ferritic materials used in the reactor vessel are listed in Table 5.2-2. These materials are in accordance with ASME Section III. Ferritic reactor vessel pressure boundary materials satisfy the fracture toughness requirements of 10 CFR Part 50, Appendix G and NRC SRP BTP 5-3 (Reference 7).

The physical and mechanical properties of the reactor vessel material and the effects of radiation on the materials are described in Subsection 5.3.2.1.1.

5.3.1.2 Special Process Used for Manufacturing and Fabrication

The reactor vessel is fabricated in accordance with ASME Section III (Reference 8), NB-4000, and its materials satisfy the requirements of ASME Section III, NB-2000. Application of the appropriate Code Symbol and completion of a data report are in accordance with ASME Section III, NCA-8000. No special manufacturing methods that could compromise the integrity of the vessel are used. The reactor vessel is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical upper closure head. The construction consists of forged rings, forged hemispherical heads, forged flanges on the closure head, and forged nozzles. Pressure boundary reactor vessel forgings are made of ASME SA-508 Grade 3, Class 1, low-alloy steel. The forgings are supplied in a quenched and tempered condition. Vacuum degassing, to lower the hydrogen level and to improve the quality of the steel, is applied to the forgings. The forgings in the reactor vessel beltline and the as-deposited welds contain no greater than the weight percent of residual elements described in Subsection 5.2.3.1. The internal surfaces that are in contact with the reactor coolant and vessel flange surface are clad with austenitic stainless steel or NiCrFe alloy.

The reactor vessel except for the closure head consists of three shell sections (upper, intermediate, and lower) and a lower head. The upper shell section is forged to have a flange with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The length of each shell is adjusted not to include the welds within the effective core region. The vessel flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seals. Each shell consists of one 360-degree forged ring. The lower head is constructed of a single hemispherical forging. The three shell sections and the lower head forging are joined by welding, along with 4 inlet nozzle forgings, 2 outlet nozzle forgings, 4 direct vessel injection (DVI) nozzle forgings, and 61 in-core instrument nozzles to form a complete vessel assembly.

The closure head is fabricated separately since it is joined to the reactor vessel by bolting. The closure head consists of a head flange and a dome. The head flange is a forged ring. The flange is drilled to match the vessel flange stud hole locations, and the lower surface of the flange is machined to provide a mating surface for the vessel closure seals. The dome is constructed of a single hemispherical forging. The dome and flange are welded together to form the closure head, and the control element drive mechanism (CEDM) nozzles are welded into the head to complete the assembly.

Welding materials for the reactor vessel conform to ASME Sections II and III, or satisfy requirements for other welding materials as permitted in ASME Section IX. Table 5.2-2 shows the welding material specifications for the APR1400 RCS application.

Welding of the pressure boundary parts of the reactor vessel is performed in accordance with the welding procedure specifications (WPSs), which satisfy the requirements of ASME Sections III and IX.

Welding processes such as submerged arc welding (SAW), flux cored arc welding (FCAW), and gas tungsten arc welding (GTAW) are applied for stainless steel or nickel-based alloy cladding of reactor vessel internal surfaces. Welding processes such as GTAW and shielded metal arc welding (SMAW) are applied to dissimilar welds by buttering. Welding processes such as SMAW and SAW are applied to the reactor vessel dome and flange weld, flange-shell-head welds, and other girth seam welds. The build-up application uses primarily the SMAW process. Electroslag welding is not used in the reactor vessel.

Welding of low-alloy steel pressure boundary welds conforms with the recommendations of NRC RG 1.50 (Reference 9). Preheat temperatures utilized for low alloy steel are in accordance with ASME Section III, Appendix D. The maximum interpass temperature utilized is generally 260 °C (500 °F). In the event that the requirements in Paragraphs C.1, C.2, and C.3 of NRC RG 1.50 are not met, the soundness of the weld is demonstrated by an examination that meets the acceptance criteria specified in ASME Section III. Hydrogen is removed either by maintaining preheat until post-weld heat treatment is performed or by post-heating at a temperature and for the length of time that are sufficient to preclude the effects of hydrogen-assisted cracking.

Post-weld heat treatment temperature and time for welds of low alloy steels are in accordance with ASME Section III, NB-4620.

5.3.1.3 Special Methods for Nondestructive Examinations

Prior to, during, and after fabrication of the reactor vessel, nondestructive tests based on ASME Section III are performed on all welds and forgings as required. The nondestructive examination requirements including calibration methods, instrumentation, sensitivity, reproducibility of data, and acceptance standards are in accordance with the requirements of ASME Section III (see Table 5.2-1). These methods, procedures, and requirements are compatible with ASME Section XI (Reference 10) so that the results of the preservice inspections can be correlated with inservice inspections. Strict quality control is maintained in critical areas such as calibration of test instruments.

All full-penetration, pressure-containing welds are 100 percent radiographed to the standards of ASME Section III. Weld preparation areas, back-chip areas, and final weld surfaces are magnetic-particle or liquid-penetrant examined. Other pressure-containing welds, such as those used for the attachments of nonferrous nickel-chromium-iron CEDM nozzles and vent and instrument nozzles to the reactor vessel and head, are inspected by liquid-penetrant tests of the root pass (the lesser of half of the thickness or each 12.7 mm (0.5 in) of weld deposit and the final surface). Additionally, the base metal weld preparation area is magnetic-particle and is examined prior to overlay with nickel-chromium-iron weld metal.

All forgings are inspected by ultrasonic testing using longitudinal beam techniques. In addition, ring forgings are tested using shear wave techniques.

All carbon-steel and low alloy forgings and ferritic welds are also subjected to magnetic-particle examination after stress relief.

All vessel bolting material is examined using ultrasonic and magnetic-particle examination during the manufacturing process. The bolting material receives a straight-beam, longitudinal and radial-scan, ultrasonic examination with a search unit area not exceeding 645 mm^2 (1 in^2).

All hollow material excluding stud, nut, and washer material receives second ultrasonic examination using angle-beam in circumferential direction with a search unit area not exceeding 645 mm^2 (1 in^2). A reference specimen of the same composition and thickness containing a notch (located on the inside surface) of 25.4 mm (1 in) in length and a depth of 3 percent of nominal section thickness, or 6 mm (1/4 in), whichever is less, is used for calibration. Use of these techniques provides reasonable assurance that no materials that have unacceptable flaws, observable cracks, or sharply defined linear defects are used. The magnetic-particle inspection is performed both before and after threading of the studs.

Upon completion of all post-weld heat treatments, the reactor vessel is hydrostatically tested, and all accessible ferritic weld surfaces, including those used to repair material, are magnetic-particle inspected in accordance with ASME Section III.

Clad surfaces that are subject to high load conditions and carrying load of attachments are ultrasonically tested as necessary to provide reasonable assurance of the bond required for the intended service. In all other areas, the cladding is also ultrasonically inspected for lack of bond, transverse to the direction of welding.

Tables 5.3-8 and 5.3-9 summarize the nondestructive examination methods that are used for the reactor vessel base materials and welds, respectively.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Subsection 5.2.3 describes details concerning controls for welding of ferritic and austenitic stainless steels. Applicable NRC Regulatory Guides (RGs) for the reactor vessel for this purpose and subsections describing conformance to the RGs are identified as follows:

a. NRC RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal” (Reference 11), is addressed in Subsection 5.2.3.4.4.

b. NRC RG 1.34, "Control of Electroslag Weld Properties" (Reference 12), is addressed in Subsection 5.2.3.3.

c. NRC RG 1.28, "Quality Assurance Program Criteria (Design and Construction)" (Reference 13), is addressed in Subsection 5.2.3.4.2.

d. NRC RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components" (Reference 14), is addressed in Subsection 5.2.3.3.

e. NRC RG 1.44, "Control of the Processing and Use of Stainless Steel" (Reference 15), is addressed in Subsection 5.2.3.4.1.

f. NRC RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," is addressed in Subsection 5.2.3.3.

g. NRC RG 1.71, "Welder Qualification for Areas of Limited Accessibility" (Reference 16), is addressed in Subsection 5.2.3.3.

h. NRC RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials" (Reference 17), is addressed in Subsection 5.3.1.6.7.

i. NRC RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 18), is addressed in Subsection 4.3.3.3.

Tools used in abrasive work operations on austenitic stainless steel, such as grinding or wire brushing, do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress corrosion cracking (SCC).

5.3.1.5 Fracture Toughness

In accordance with 10 CFR Part 50, Appendix G, Paragraph IV A, the reactor vessel beltline materials have minimum upper-shelf energy of 102 Joules (75 ft-lbs) as determined from Charpy V-notch tests on unirradiated specimens in accordance with ASME Section III, NB-2320. In addition, the reactor vessel beltline materials satisfy the requirements of NRC SRP BTP 5-3. The detailed information on the upper shelf energy of the reactor vessel beltline materials is described in Subsection 5.3.2.4. Charpy impact tests are performed on transversely (weak direction) oriented specimens from the beltline forgings to establish RT_{NDT} as required by 10 CFR Part 50, Appendix G.

The RT_{NDT} of reactor vessel materials is determined based on a nil-ductility transition temperature (T_{NDT}) obtained through a drop weight test and the results of Charpy V-notch tests which are carried out at temperatures not greater than T_{NDT} + 33 °C (T_{NDT} + 60 °F). The T_{NDT} is the RT_{NDT} when the Charpy V-notch test results exhibit at least 0.89 mm (35 mils) lateral expansion and not less than 68 J (50 ft-lb) absorbed energy. Test coupons, test specimens, testing procedures, testing requirements, and acceptance criteria for RT_{NDT} determination are fabricated, established, or applied in accordance with ASME Section III, NB-2300.

An initial RT_{NDT} for the reactor vessel active core region base material, the girth seam between the intermediate and lower shell courses, and the girth seam between the lower shell courses and bottom head is -23.3 °C (-10 °F). An initial RT_{NDT} for the remaining material of the reactor vessel pressure boundary is -12.2 °C (10 °F).

As a result of fast neutron irradiation in the region of the core, RT_{NDT} of irradiated material increases with operation. The effect of neutron irradiation is taken into account in accordance with NRC RG 1.99.

5.3.1.6 Material Surveillance

The irradiation surveillance program for the APR1400 is conducted to assess the neutron-induced changes in the RT_{NDT} (reference temperature) and mechanical properties of the reactor vessel materials. Changes in the impact and mechanical properties of the material are evaluated by the comparison of pre-irradiation and post-irradiation test results. The capsules containing the surveillance test specimens used for monitoring the neutron-induced property changes of the reactor vessel materials are irradiated under conditions that represent, as closely as practical, the irradiation conditions of the reactor vessel.

The APR1400 reactor vessel surveillance program satisfies the ASTM E185 (Reference 19) requirements for surveillance tests in light water cooled nuclear power reactor vessels, and the requirements of Appendix H of 10 CFR Part 50 and NRC SRP BTP 5-3. The COL applicant is to provide a reactor vessel material surveillance program for a specific plant (COL 5.3(1)).

5.3.1.6.1 Test Material Selection

Materials selected for the surveillance program are those judged most likely to be controlling with regard to radiation embrittlement according to the recommendations of NRC RG 1.99.

Surveillance test materials are prepared from the actual materials used in fabricating the beltline region of the reactor vessel. The test materials are processed so they are representative of the materials in the completed reactor vessel. Specimens are prepared from three metallurgically different materials, including base metal, weld metal, and heat-affected zone (HAZ) material.

Base metal test material is from a section of the shell course forging that is selected from the beltline of the reactor vessel. Selection is based on an evaluation of initial toughness (characterized by an index temperature such as RT_{NDT}) and the estimated effect of chemical composition and neutron fluence on the toughness during reactor operation. Normally, the forging with the highest adjusted reference temperature at end-of-life is selected as the surveillance base metal test material.

Weld metal test material, representative of the controlling reactor pressure vessel weld, is produced by welding together sections of forgings from the beltline of the reactor vessel. The HAZ test material is manufactured from a section of the same forgings used for base metal surveillance test material. The weld metal test material is produced from the same heat of weld wire or rod and lot of flux used in the beltline of the reactor vessel. Welding parameters duplicate those used for the beltline welds.

Two additional sets of test specimens for each material are provided as representative stock (archival material) with documentation and identification.

5.3.1.6.2 Test Specimens

5.3.1.6.2.1 Type and Quantity

Drop weight, standard and precracked Charpy impact, tensile test, and compact tension fracture toughness specimens are provided for unirradiated baseline tests. Drop weight tests are conducted in accordance with ASTM E208 (Reference 20). Charpy impact tests are conducted in accordance with ASTM E23 and A370 (References 21 and 22, respectively). Tensile tests are conducted in accordance with ASTM E8/8M and E21 (References 23 and 24, respectively). Static fracture toughness tests are performed in accordance with ASTM E1820 and/or E1921 (References 25 and 26, respectively). Correlation of drop weight and Charpy impact tests to establish RT_{NDT} is made in accordance with ASME Section III, NB-2300. Standard and precracked Charpy impact, tensile test, and compact tension fracture toughness specimens are provided for post-irradiation tests.

The total quantity of specimens furnished for carrying out the overall requirements of this program is presented in Table 5.3-1.

Type and quantity of specimens for baseline testing are shown in Table 5.3-2, and for irradiation encapsulation are shown in Table 5.3-3.

5.3.1.6.2.2 Baseline Specimens

The type and quantity of test specimens provided for establishing the properties of the unirradiated reactor vessel materials are presented in Table 5.3-2. The data from tests of these specimens provide the basis for determining the neutron-induced property changes of the reactor vessel materials.

Twelve drop weight test specimens are provided for each base metal (transverse), weld metal, and HAZ material for establishing the nil-ductility transition temperature (T_{NDT}) of the unirradiated surveillance materials. These data form the basis for RT_{NDT} determination. RT_{NDT} is the reference temperature from which subsequent neutron-induced changes are determined.

Twenty-four standard Charpy test specimens are provided for each base metal (longitudinal and transverse), weld metal, and HAZ material. This quantity exceeds the minimum number of test specimens recommended by ASTM E185 for developing a Charpy impact energy transition curve and is intended to provide a sufficient number of data points for establishing accurate Charpy impact energy transition temperatures for these materials. These data, together with the drop weight T_{NDT}, are used to establish an RT_{NDT} for each material.

Twelve precracked Charpy impact test specimens are provided for each base metal (longitudinal and transverse) and weld metal in addition to the standard Charpy impact specimens. This quantity is sufficient to determine fracture toughness properties (critical stress intensity factors under dynamic loading) over the range extending from linear elastic to elastic-plastic fracture.

Twelve tensile test specimens are provided for each base metal (longitudinal and transverse) and weld metal. This quantity also exceeds the minimum number of test specimens recommended by ASTM E185 and is intended to permit a sufficient number of tests for accurately establishing the tensile properties for these materials at a minimum of three test temperatures.

Eight 1T compact tension test specimens and four 1/2T compact tension test specimens are provided for each base metal (transverse) and weld metal. For base metal (longitudinal), four 1T and four 1/2T compact tension test specimens are provided. These specimens are for augmenting the fracture toughness data determined from the precracked Charpy tests. This quantity of specimens is sufficient to determine fracture toughness properties over the range extending from linear elastic to elastic-plastic fracture behavior.

5.3.1.6.2.3 Irradiated Specimens

Specimens for the tensile test, standard and precracked Charpy impact test, and 1/2T compact tension test are used for determining changes in the strength and static and dynamic toughness properties of the materials due to neutron irradiation. A total of 360 standard Charpy impact, 162 precracked Charpy impact, 72 1/2T compact tension, and 54 tensile test specimens are provided. The type and quantity of test specimens provided for establishing the properties of irradiated materials over the 60 years life of the vessel are presented in Table 5.3-3. Compact tension specimens provided for capsule irradiation are precracked prior to insertion to reduce the time required for post-irradiation testing. The types and quantity of specimens provided exceed the minimum requirements of ASTM E185.

5.3.1.6.3 Surveillance Capsules

The surveillance test specimens are placed in corrosion resistant capsule assemblies for protection from the primary coolant during irradiation. The capsules also serve to physically locate the test specimens in selected positions within the reactor vessel and to facilitate the removal of a desired quantity of test specimens when a specified radiation exposure has been attained. Six identical surveillance capsule assemblies are provided for the reactor vessel. Four of the assemblies are for retrieval, and two are for standby. The type and quantity of specimens contained in each type of capsule assembly are presented in Table 5.3-4.

A typical capsule assembly, illustrated in Figure 5.3-1, consists of a series of three specimen compartments that are connected by wedge couplings and a lock assembly. Each compartment enclosure of the capsule assembly is internally supported by the surveillance specimens and is externally pressure tested to 219.7 kg/cm^2 (3,125 psi) during final fabrication. The wedge couplings also serve as end caps for the specimen compartments and position the compartments within the capsule holders, which are attached to the reactor vessel cladding. The lock assemblies fix the locations of the capsules within the holders by exerting axial forces on the wedge coupling assemblies; this causes the wedges to exert horizontal forces against the sides of the holders preventing relative motion. The lock assemblies also serve as a point of attachment for the tooling used to remove the capsules from the reactor.

Each capsule assembly is made up of the upper compartment assembly, the center compartment assembly, and the lower compartment assembly. Each capsule assembly is assigned a unique identification so that a complete record of test specimen location within each compartment can be maintained.

5.3.1.6.3.1 Upper Compartment Assembly

The upper compartment assembly contains 12 Charpy impact test specimens, nine base metal (longitudinal) Charpy impact test specimens, and a set of five flux spectrum monitors in the top section as shown in Figure 5.3-2. The bottom section contains three base metal (longitudinal) Charpy impact test specimens, six base metal (transverse) Charpy impact test specimens, three base metal (longitudinal) tension specimens, and four base metal (longitudinal) 1/2T compact tension specimens.

The Charpy test specimens are arranged vertically in 1 × 3 arrays and are oriented with the notch toward the reactor core. The 1/2T compact tension specimens are oriented so that the opening of the crack starter notch is facing the top of the compartment. This orientation results in a neutron flux gradient parallel to the crack front. The temperature differential between the specimens and the reactor coolant is minimized by using spacers between the specimens and the compartment and by sealing both sections of the assembly in an atmosphere of helium. This quantity of specimens provides an adequate number of data points for establishing a Charpy impact energy transition curve for a given irradiated material. Comparison of the unirradiated and irradiated Charpy impact energy transition curves permits determination of the RT_{NDT} changes due to irradiation for the various materials.

5.3.1.6.3.2 Center Compartment Assembly

The center compartment assembly contains six base metal (transverse) Charpy impact test specimens, 12 weld metal Charpy impact test specimens, and six base metal (longitudinal) precracked Charpy impact test specimens in the top section as shown in Figure 5.3-3. The bottom section contains three base metal (longitudinal) precracked Charpy impact test specimens, a set of 11 flux spectrum monitors, a set of four temperature monitors, three base metal (transverse) tension specimens, and four base metal (transverse) 1/2T compact tension specimens. Both compartment sections are sealed within an atmosphere of helium. The tension specimens are placed in a housing machined to fit the compartment. Split spacers are placed around the specimen gage length to minimize the temperature differential between the specimen gage length and the reactor coolant. The impact specimens are arranged vertically in 1 × 3 arrays and are oriented with the notch toward the reactor core. Spacers are utilized between the test specimens and the compartment.

5.3.1.6.3.3 Lower Compartment Assembly

The lower compartment assembly contains four weld metal 1/2T compact tension specimens, three weld metal tension test specimens, and nine base metal (transverse) pre-cracked Charpy impact test specimens in the top section as shown in Figure 5.3-4. The tension specimens are placed in a housing machined to fit the compartment. Split spacers are placed around the specimen gage length to minimize the temperature differential between the specimen gage length and the reactor coolant. The impact specimens are arranged vertically in 1 × 3 arrays and are oriented with the notch toward the reactor core. Spacers are utilized between the test specimens and the compartment. The bottom section contains a set of five flux spectrum monitors, nine weld metal precracked Charpy impact test specimens, and 12 Charpy impact test specimens. Both compartment sections are sealed within an atmosphere of helium.

5.3.1.6.4 Neutron Irradiation and Temperature Exposure

The changes in the RT_{NDT} of the reactor vessel materials are derived from specimens irradiated to various fluence levels in different neutron energy spectra. To accurately predict the RT_{NDT} of the vessel materials, complete information on the neutron flux energy spectra and the irradiation temperature of the encapsulated specimens are available.

5.3.1.6.4.1 Flux Measurements

Fast neutron flux measurements are obtained by insertion of threshold detectors into each of the six irradiation capsules. Such detectors are particularly suited for the proposed application because their effective threshold energies are in the range of interest (0.5 to 15 MeV).

These neutron threshold detectors and the thermal neutron detectors, presented in Table 5.3-5, can be used to monitor the thermal and fast neutron spectra incident on the test specimens. These detectors possess reasonably long half-lives and activation cross sections covering the desired neutron energy range.

Three sets of flux spectrum monitors are included in each capsule assembly. Each detector is placed inside a sheath that identifies the material and facilitates handling.

Cadmium covers are used for the materials with competing neutron capture activities. The flux monitors are placed in holes drilled in stainless steel housings, as shown in Figure 5.3-3, at three axial locations in each capsule assembly to provide an axial profile of the level of fluence that the specimens attain.

To determine the neutron fluence at the inner surface of the reactor vessel both of the measured value from detectors and the calculated value by transport theory are considered in accordance with NRC RG 1.190. Calculation methods for the neutron flux and fluence are described in Subsection 4.3.3.3.

5.3.1.6.4.2 <u>Temperature Estimates</u>

Because the changes in mechanical and impact properties of irradiated specimens are highly dependent on the irradiation temperature, it is necessary to have information on the specimens as well as the pressure vessel. During irradiation, instrumented capsules are not practicable for a surveillance program extending over the design lifetime of a power reactor. The maximum temperature of the irradiated specimens can be estimated with reasonable accuracy by including small pieces of low melting point alloys or pure metals in the capsule assemblies. The compositions of candidate materials with melting points in the operating range of power reactors are listed in Table 5.3-6. The monitors are selected to bracket the operating temperature of the reactor vessel. The temperature monitors consist of a helix of low melting alloy wire inside a sealed quartz tube. A stainless steel weight is provided to destroy the integrity of the wire when the melting point of the alloy is reached. The compositions and the melting temperatures of the temperature monitors are differentiated by the physical lengths of the quartz tubes that contain the alloy wires.

A set of temperature monitors is included in each capsule assembly. The temperature monitors are placed in holes drilled in stainless steel housings, as shown in Figure 5.3-3, and provide the maximum temperature to which the specimens are exposed.

5.3.1.6.5 <u>Irradiation Locations</u>

The test specimens are enclosed within six capsule assemblies. The axial positions of capsule assemblies are bisected by the midplane of the core as shown in Figure 5.3-6. A summary of the specimens contained in each of these capsule assemblies is presented in

Table 5.3-4. The test specimens contained in the capsule assemblies are used for monitoring the neutron-induced property changes of the reactor vessel materials. These capsules, therefore, are positioned near the inside wall of the reactor vessel so that the irradiation conditions (fluence, flux spectrum, temperature) of the test specimens resemble as closely as possible the irradiation conditions of the reactor vessel. The neutron fluence of the test specimens is expected to be approximately 1.5 times higher than that seen by the adjacent vessel wall, and the measured changes in properties of the surveillance materials are therefore able to predict the radiation induced changes in the reactor vessel beltline materials. The capsule assemblies are placed in capsule holders positioned circumferentially about the core at locations that include the regions of maximum flux. Figure 5.3-5 presents the typical exposure locations for the capsule assemblies in the plan view.

All capsule assemblies are inserted into their respective capsule holders during the final reactor assembly operation. The design also permits the remote installation of replacement capsule assemblies. The capsule holders are welded to the vessel cladding on the inside surface, and the welds are subject to inspection according to the requirements for permanent structural attachments as given in ASME Sections III and XI.

5.3.1.6.6 Withdrawal Schedule

The capsule assemblies remain within their holders until the specimens in the assemblies have been exposed to predetermined removal schedule based on effective full power years (EFPYs). At that time, the capsule assembly is removed, and the surveillance materials are evaluated. The target fluence levels for the surveillance capsules are determined at the azimuthal locations for the recommended withdrawal schedule of ASTM E185, extended to a design life of 60 years. The fluence values in Table 5.3-7 are accurate within +20 percent, -20 percent. The uncertainty is composed of errors in the calculational method and errors in the combined radial and axial power distribution.

Withdrawal schedules may be modified to coincide with the refueling outages or plant shutdowns most closely approaching the withdrawal schedule. The two standby capsules are provided in the event they are needed to monitor the effect of a major core change or annealing of the vessel or to provide supplemental toughness data for evaluating a flaw in the beltline.

5.3.1.6.7 Irradiation Effects Prediction Basis

Irradiation induced RT_{NDT} shift and reduction of upper shelf energy are predicted based on NRC RG 1.99, “Radiation Embrittlement of Reactor Vessel Materials.” Predicted changes in RT_{NDT} and upper shelf energy are used to select the surveillance materials (see Subsection 5.3.1.6.1) and to formulate the initial heatup and cooldown limit curves for plant operation. When actual post-irradiation surveillance data become available for each reactor vessel, the data are used to adjust plant operating limit curves.

5.3.1.7 Reactor Vessel Fasteners

The bolting material for the reactor vessel closure head is fabricated from SA-540 B24, Class 3 material (Reference 27). This material conforms to the requirements of 10 CFR Part 50, Appendix G, and the intent of NRC RG 1.65 (Reference 33). Nondestructive examination is performed according to ASME Section III, NB-2580, during the manufacturing process. More information on NDE is described in Subsection 5.3.1.3.

Material for reactor vessel studs is tested in accordance with the requirements of SA-540 Grade B24 Class 3, ASME Section III, NB-2220 and NB-2300. The maximum measured yield strength of the reactor vessel closure stud bolting material SA-540, Grade B-24 will not exceed 1035 MPa (150 ksi). Charpy V-Notch testing is required to be performed at a temperature of 4.4 °C (40 °F) or lower. Each specimen of one test (consisting of three specimens) is required to exhibit a minimum of 0.635 mm (25 mils) lateral expansion and 68 Joules (45 ft-lbs) absorbed energy. These requirements also satisfy NRC RG 1.65 and 10 CFR Part 50, Appendix G, Paragraph IV A, at a temperature of 4.4 °C (40 °F) or lower, depending on the actual test temperature for the reactor vessel stud preload temperature or lowest service temperature, whichever is lower.

Actual material property values are reported on the certified material test reports (CMTRs) for the reactor vessel stud material.

The use of a manganese phosphate coating on threads of studs, nuts, and washers is specified to improve anti-galling properties and resistance to corrosion. In addition, a nickel-based high purity anti-seize lubricant is specified to be added to threads and bearing surfaces at installation to further enhance anti-galling properties. Field experience to date has shown no evidence of deleterious breakdown of either phosphate coating or lubricant.

The stud holes in the vessel flange are protected from the refueling water using the seal plugs prior to refueling activity.

5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper Shelf Energy Data and Analyses

All components in the RCS are designed to withstand the effects of cyclic loads due to RCS temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operation. The design number of cycles for heatup and cooldown is based on a rate of 55.6 °C/hr (100 °F/hr). During unit startup and shutdown, the rate of temperature change is limited to less than 55.6 °C/hr (100 °F/hr) by administrative procedure. The maximum allowable RCS pressure at the corresponding minimum allowable temperature is based upon the stress limitations for brittle fracture. These limitations are derived using linear elastic fracture mechanics principles, the procedures prescribed by ASME Section XI, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” Appendix G to 10 CFR Part 50, “Fracture Toughness Requirements,” NRC RG 1.99, “Radiation Embrittlement of Reactor Vessel Materials,” 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” NRC SRP BTP 5-3, “Fracture Toughness Requirements,” and the procedures recommended by Welding Research Council (WRC) Bulletin 175, “Pressure Vessel Research Committee (PVRC) Recommendations on Toughness Requirements for Ferritic Materials” (Reference 28). The reactor vessel is also designed, fabricated, erected, and tested to conform with the requirements of 10 CFR 50.55a, 10 CFR 50.60, and 10 CFR Part 50, Appendix A (GDC 1, 14, 31, and 32).

5.3.2.1 Pressure-Temperature Limitation Curves

5.3.2.1.1 Material Properties

Pressure-temperature limitations (P-T limits) are determined using material property test data for reactor coolant pressure boundary materials, as required by ASME Section XI, Appendix G. Based on considerations of existing material property test data, an initial

RT_{NDT} for the reactor vessel beltline forging is -23.3 °C (-10 °F), and an initial RT_{NDT} for the remaining material including weld materials of the reactor coolant system (RCS) is -12.2 °C (10 °F). RT_{NDT} is determined in accordance with Article NB-2300 of ASME Section III.

As a result of fast neutron irradiation in the region of the core, the RT_{NDT} of irradiated material increases with operation. The maximum integrated neutron fluence on the reactor vessel wall beltline region is estimated to be 9.5×10^{19} n/cm^2. There are no longitudinal seam welds, and two of the circumferential seam welds are located near the fringes of core beltline region. It is conservatively assumed that weld materials are subjected to the maximum neutron fluence of 9.5 x 10^{19}n/cm^2 for evaluating RT_{NDT}. The techniques used to analytically and experimentally predict the integrated fast neutron ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Subsection 5.3.1.6.

The shift in RT_{NDT} of reactor vessel beltline materials can be analytically predicted based on the procedures described in NRC RG 1.99 because the RCS operating temperature (cold leg) is 290.6 °C (555 °F), which is above 274 °C (525 °F) as shown in Table 5.1.1-1. The surveillance program is prepared to obtain the reliable irradiation data for the adjustment or qualification of operating parameters. Reactor vessel shell materials are designed to limit RT_{NDT} values at 1/4T location within 93.3 °C (200 °F) at the end-of-life. The RT_{NDT} values at 1/4T location at the end-of-life are expected to be 20.2 °C (68.4 °F) for beltline forging and 47.2 ^{o}C (117 ^{o}F) for weld material per NRC RG 1.99 based on the weight percent of residual elements in Subsection 5.2.3.1.

The measured shift in RT_{NDT} for a specimen is applied to the adjacent section of the reactor vessel for later stages in plant life because the measured neutron spectra and flux at the specimen and reactor vessel inside radius are close. The measured shift in RT_{NDT} is adjusted for the difference in calculated flux magnitudes between the surveillance specimens and the point of interest in the reactor vessel wall.

The maximum fluence of the reactor vessel is obtained from the measured exposure by application of the calculated azimuthal neutron flux variation. The neutron fluence and the actual shift in RT_{NDT} are established periodically during plant operation by testing the reactor vessel surveillance material specimens that are irradiated in capsules secured to the inside wall of the reactor vessel, as described in Subsection 5.3.1.6 and shown in Figures

5.3-5 and 5.3-6. Because surveillance materials are irradiated close to the inside surface of the vessel, the specimens demonstrate the accuracy of the RT_{NDT} shift prediction as well as the effect of any additional long-term aging phenomenon. The P-T limits are then adjusted periodically, if necessary, to stay within allowable stress limits during normal operation.

5.3.2.1.2 Determination of Pressure-Temperature Limitation Curves

Figure 5.3-7 shows the pressure-temperature limitations determined in accordance with Appendix G of 10 CFR Part 50 for normal operation of the RCS. Details of P-T limit are described in the pressure and temperature limits report (PTLR) (Reference 29).

The P-T Limit curves are determined based on the following:

a. Minimum boltup temperature

   The minimum boltup temperature is governed principally by ASME Section XI, Appendix G, G-2222(c). The ASME Code requires that when the flange and adjacent shell regions are stressed by the full bolt preload and by pressure not exceeding 20 percent of the preoperational system hydrostatic test pressure, the minimum metal temperature in the stressed region must be at least the initial RT_{NDT} plus any effects of irradiation. From the assumptions that the initial RT_{NDT} for all regions of the RCS other than the beltline is -12.2 °C (10 °F) and that the flange and adjacent shell regions are not subjected to significant irradiation, the minimum boltup temperature per this requirement is -12.2 °C (10 °F). The minimum boltup temperature is taken conservatively to be 21.1 °C (70 °F).

b. Maximum pressure below the lowest service temperature (LST)

   The maximum allowable pressure below the lowest service temperature is defined by ASME Section III, NB-2332 (b), to be 20 percent of the preoperational hydrostatic test pressure. This test pressure is 125 percent of the design pressure (175.8 kg/cm^2A [2,500 psia]) or 219.7 kg/cm^2A (3,125 psia), and 20 percent of this value is 43.9 kg/cm^2A (625 psia). Therefore, the maximum pressure below the lowest service temperature is 43.9 kg/cm^2A (625 psia).

c. Minimum required temperature

The lowest service temperature (LST) is defined by ASME Section III, Article NB-2332 (b), to be the minimum allowable temperature at pressures above 20 percent of the preoperational hydrostatic test pressure. This value is defined to be no lower than RT_{NDT} + 55.6 °C (100 °F), where the RT_{NDT} is considered to be for the most limiting component in the RCS other than the beltline. From the assumption that the initial RT_{NDT} for all regions of the RCS is -12.2 °C (10 °F), the LST per this requirement is -12.2 °C (10 °F) + 55.6 °C (100 °F) = 43.4 °C (110 °F) (actual fluid temperature at the beltline). However, minimum allowable temperature in accordance with the minimum temperature requirements in Table 1 of 10 CFR Part 50, Appendix G, is calculated as 54.4 °C (130 °F), which is higher than the LST and, therefore, is used as the minimum required temperature.

d. Operation, heatup, and cooldown curves

1) Reactor vessel beltline

P-T limits for the reactor vessel beltline are examined for heatup and cooldown conditions. For the heatup analysis, both the 1/4 thickness (1/4T) and 3/4T locations are examined. For heatup, the thermal stresses, σ_t are compressive at the inside surface and tensile at the outside surface. The membrane stresses due to pressure, σ_m are always tensile, but more so at the inside than outside surface. As a result, the total stress is always greater at the outside than the inside surface. However, the maximum allowable stresses, taking irradiation effects into account, decrease more at the inside than outside surface because the effects of irradiation are more pronounced there. It is not clear which surface stresses approach this maximum first. Therefore, both locations are examined for the heatup transient.

For the cooldown analysis, only the 1/4T location needs to be examined. During cooldown, the thermal stresses are tensile at the inside surface and compressive at the outside. Again, membrane stresses due to pressure are always positive. As a result, the total stress is always greater at the inside than the outside surface. Since the maximum allowable stresses, taking irradiation effects into account, decrease more at the inside than outside surface, it is clear that the total stress at the inside surface approaches the maximum allowable before those at the outside surface. Therefore, only the 1/4T location needs to be examined for the cooldown transient.

Heatup and cooldown rates from 0 °C/hr (0 °F/hr) (i.e., isothermal pressurization) to the design limit of 55.6 °C/hr (100 °F/hr) are examined in determining the allowable heatup and cooldown rates as a function of temperature to meet LTOP requirements for the RCS. LTOP considerations are discussed in Subsection 5.2.2.

For the beltline analysis, during normal operations, the following condition is maintained:

$$K_{IC} = 2K_{Im} + K_{It}$$

Where:

K_{IC} = reference stress intensity factor specified by Figure G-2210-1 in ASME Section XI, Appendix G

K_{Im} = stress intensity factor for membrane stress due to pressure

K_{It} = stress intensity factor for thermal stress

K_{Im} = $\sigma_m\ M_m$

M_m = membrane correction factor defined in ASME Code Section XI, Appendix G, G-2214.1

σ_m = $(Pr)/t$

For inside defect:

K_{It} = $0.579 \times 10^{-6} \times CR \times t^{2.5}$ $\quad$ $MPa{\cdot}m^{1/2}$ or

K_{It} = $0.953 \times 10^{-3} \times CR \times t^{2.5}$ $ksi{\cdot}in^{1/2}$

For outside defect:

K_{It} = $0.458 \times 10^{-6} \times HU \times t^{2.5}$ $MPa{\cdot}m^{1/2}$ or

K_{It} = $0.753 \times 10^{-3} \times HU \times t^{2.5}$ $ksi{\cdot}in^{1/2}$

Where:

P = internal RV pressure

r = inside RV radius

t = RV wall thickness, mm (in)

CR = cooldown rate, °C/hr (°F/hr)

HU = heatup rate, °C/hr (°F/hr)

The right side of the above equation ($2K_{Im} + K_{It}$) is calculated for various pressures and then set equal to the left side (K_{IC}). 2 represents a safety factor required by ASME Section XI, Appendix G. The minimum allowable temperature corresponding to a given pressure can then be calculated from this equation. The resulting combination represents a (maximum) pressure-(minimum) temperature limit that is not exceeded in order to provide reasonable assurance of non-brittle material behavior.

A set of such pressure-temperature coordinates defines a P-T limit curve for a specific heatup or cooldown rate at a particular point in plant life. Figure 5.3-7 illustrates typical P-T limits for the end-of-life of 60 years or 55.8 EFPYs (based on 93 percent capacity factor).

The pressure used in computing K_{Im}, and K_{It} and the corresponding computed temperature are for the crack tip. Correction factors for pressure and temperature are taken into account when expressing P-T limits graphically to show the limits in terms of indicated pressurizer pressure and indicated RCS temperature. This is done for P-T Limit curve presentation in the Technical Specifications (Chapter 16). The correction factors take into account the effects of instrument error, pressure differentials due to flow in the RCS, static pressure differentials due to elevation differences, and temperature differentials due to thermal gradients in the reactor vessel wall. The P-T limits shown in Figure 5.3-7 are in terms of indicated fluid conditions.

K_{IC} is a function of the temperature, T, and the adjusted reference temperature (ART) of the material at the cracktip. The analytical expression for K_{IC} is as follows:

$$K_{IC} = 36.5 + 22.783 \exp\{0.036(T - ART)\} \quad MPa \cdot m^{1/2} \quad \text{or}$$
$$K_{IC} = 33.2 + 20.734 \exp\{0.02(T - ART)\} \quad ksi \cdot in^{1/2}$$

The calculation of the ART is in accordance with the procedure described in NRC RG 1.99. The ART is a function of the initial RT_{NDT} of the material, the shift in RT_{NDT} due to irradiation over a period of time, ΔRT_{NDT}, and a safety margin. The equation for the ART is:

$$ART = (Initial)RT_{NDT} + \Delta RT_{NDT} + Margin$$

The shift, ΔRT_{NDT}, is calculated from:

$$\Delta RT_{NDT} = (CF)\ f^{(0.28 - 0.10 \log f)}$$

where CF is a chemistry factor and f is the neutron fluence at the point of interest in the reactor vessel wall. The fluence, f, is calculated from the surface fluence, $f_{surface}$ (10^{19} n/cm^2, E >1 MeV), as

$$f = f_{surface}\ (e^{-0.24\ x}),\ n/cm^2$$

where x (in inches) is the depth into the RV wall.

The margin is calculated based on:

$$Margin = 2(\sigma_I^2 + \sigma_\Delta^2)^{1/2}$$

where σ_I is the standard deviation of the (Initial)RT_{NDT} data and σ_Δ is the standard deviation for ΔRT_{NDT} data as described in Regulatory Position C.1.1

of NRC RG 1.99. A value of zero is assigned to σI because a measured property is used to establish RT_{NDT}, not estimated values. For the beltline forging, the margin required to be added for uncertainties by RG 1.99 is 15.8 °C (28.4 °F). However, to cover unanticipated long-term aging phenomena, a margin of 27.8 °C (50 °F) for the vessel beltline base materials is used for additional conservatism. In case of weld materials, no additional conservatism is applied in evaluating the margin because the maximum neutron fluence is applied to the weld location. Once confirmatory data are available from the reactor vessel surveillance program, the margin can be reduced in subsequent shift calculations consistent with the RG 1.99.

2) Reactor vessel flange

P-T limits for the reactor vessel flange are examined for heatup and cooldown in accordance with the procedures in Article G-2220 in ASME Section XI, Appendix G. For the flange analysis, the following condition is maintained during normal operations:

$$K_{IC} = 2K_{I,\ primary} + K_{I,\ secondary}$$

Where:

$K_{I,\ primary}$ = $K_{m,\ pressure} + K_{m,\ boltup} + K_{b,\ boltup}$

$K_{I,\ secondary}$ = $K_{b,\ pressure} + K_t$

$K_{m,\ pressure}$ = stress intensity factor for membrane stress due to pressure

= $\sigma_{m,\ pressure} M_m$

$K_{m,\ boltup}$ = stress intensity factor for membrane stress due to boltup

= $\sigma_{m,\ boltup} M_m$

$K_{b,\ boltup}$ = stress intensity factor for bending stress due to boltup

= $\sigma_{b,\ boltup} M_b$

$\sigma_{m,\ pressure}$ = membrane (hoop) stress due to pressure

$\sigma_{m,\ boltup}$ = membrane stress due to boltup

$\sigma_{b,\ boltup}$ = bending stress due to boltup

M_m = membrane correction factor

M_b = bending correction factor

Where:

$K_{b,\ pressure}$ = stress intensity factor for bending stress due to pressure

= $\sigma_{b,\ pressure}\ M_b$

$\sigma_{b,\ pressure}$ = secondary bending stress due to pressure

K_t = stress intensity factor for thermal stress

= $M_b\ \sigma_t$

σ_t = thermal stress at the crack

The left side of the above equation (K_{IC}) is calculated for various temperatures; the right side is calculated as a function of pressure. The maximum allowable pressure corresponding to a given temperature can then be calculated from this equation. The resulting (maximum) pressure-(minimum) temperature coordinates define the flange P-T Limit for a specific heatup or cooldown at a particular point in plant life. In no case is the flange minimum temperature limit allowed to be less than the flange RT_{NDT} + 66.7 °C (120 °F) during normal operation (when the pressure exceeds 20 percent of the preoperational system hydrostatic pressure), and RT_{NDT} + 50 °C (90 °F) during hydrostatic pressure tests and leak tests in accordance with Appendix G of 10 CFR Part 50.

e. Preservice hydrostatic test limit

To conform with Table 1 of 10 CFR Part 50 Appendix G or the recommendation of Article G-2400(a) in ASME Section XI, Appendix G, the preservice hydrostatic test is to be performed at an isothermal temperature no lower than RT_{NDT} + 33.3 °C (60 °F). Assuming an initial RT_{NDT} of -12.2 °C (10 °F) for all regions of the RCS, the preservice hydrostatic test limit is 21.1 °C (70 °F) (actual fluid temperature measured at the beltline).

f. Inservice leak and hydrostatic test curves

P-T limits for inservice leak and hydrostatic tests are developed in the same manner as for the RV beltline described in the preservice hydrostatic test limit. The exception is that a safety factor of 1.5 is applied to the stress intensity factor for membrane stress due to pressure, K_{Im}, rather than 2, as allowed by Article G-2400(b) in ASME Section XI, Appendix G.

g. Core critical curves

The core critical curve is intended to provide additional margins of safety during core operation. The limit is defined as 22.2 °C (40 °F) above the minimum allowable temperature for heatup or cooldown and no less than the minimum temperature required in accordance with operating conditions 2.c and 2.d of Table 1 of 10 CFR Part 50 Appendix G.

The COL applicant is to develop P-T Limit curves based on plant-specific data (COL 5.3(2)).

5.3.2.2 Operating Procedures

Details of the limiting condition for operations related to the RCS P-T limits and their bases are specified and described in Subsections 3.4.3 and B3.4.3 of Chapter 16 (Technical Specifications and Bases). The P-T Limit curves that are provided were prepared in accordance with Appendix G of ASME Section XI.

The RCS pressure and temperature are maintained within the prescribed P-T limits, which provide reasonable assurance that the integrity of reactor coolant pressure boundary can be maintained.

5.3.2.3 Pressurized Thermal Shock

The reactor vessel meets the requirements of 10 CFR 50.61 (Reference 30), and NRC SRP BTP 5-3 (i.e., the PTS screening criteria are not projected to be exceeded by expiration of the operations).

RT_{PTS} is evaluated using the procedure described in 10 CFR 50.61, which is provided below:

$$RT_{PTS} = \text{(Initial) } RT_{NDT} + \Delta RT_{PTS} + \text{Margin}$$

The calculated maximum RT_{PTS} satisfies the screening criteria in 10 CFR 50.61(b)(2).

The PTS screening criteria are:

a. 132.2 °C (270 °F) for plates, forgings, and axial weld materials

b. 148.9 °C (300 °F) for circumferential weld materials

The following assumptions are applied in the calculation of RT_{PTS} for limiting beltline material:

a. The limiting case is the weld material subjected to the maximum integrated fast neutron fluence of 9.5×10^{19} n/cm^2.

b. For the weld material, maximum copper content is 0.05 wt% and maximum nickel content is 0.10 wt%, and the maximum initial RT_{NDT} is -12.2 °C (10 °F).

c. The adjustment in the reference temperature caused by irradiation (ΔRT_{PTS}) is calculated to be 31.4 °C (56.6 °F). The margin required by 10 CFR 50.61 is 31.1 °C (56 ˚F) for the weld materials.

The calculated RT_{PTS} is 50.6 °C (123 °F), which satisfies the above PTS screening criteria.

The COL applicant is to verify the RT_{PTS} value based on plant-specific material property and neutron fluences (COL 5.3(3)).

5.3.2.4 Upper-Shelf Energy

In accordance with 10 CFR Part 50, Appendix G, Paragraph IV A, and NRC SRP BTP 5-3, the reactor vessel beltline materials must have the minimum upper-shelf energy (USE) of 102 Joules (75 ft-lbs) initially and must maintain USE values of 68 joules (50 ft-lbs) through the life of the reactor vessel. The initial USE is determined from Charpy V-notch tests on unirradiated specimens in accordance with ASME Section III, NB-2320. Charpy impact tests are performed on transversely (weak direction) oriented specimens from the beltline forgings.

The change in the USE due to radiation embrittlement can be predicted in accordance with NRC RG 1.99 because the operating temperature (cold leg) is 290.6 °C (555 °F), which is higher than 274 °C (525 °F) as shown in Table 5.1.1-1.

Even with the limiting value of 102 Joules (75 ft-lbs) assumed for the initial USE, the end-of-life (EOL) USE is estimated to be 69.4 Joules (51 ft-lbs), which is greater than the limiting EOL value of 68 Joules (50 ft-lbs) based on the maximum neutron fluence of 9.5×10^{19} n/cm^2, a maximum copper content of 0.03 wt% for beltline forgings, and 0.05 wt% for weld materials, as assumed in Subsection 5.3.2.3. Previous data for the OPR1000 reactor vessel materials including weld metal show that initial USE values are mostly greater than 200 joules (147 ft-lbs) and that actual EOL values of USE are expected to be much greater than 68 Joules (50 ft-lbs).

The COL applicant is to verify the USE at EOL based on plant-specific material property and neutron fluences (COL 5.3(3)).

5.3.3 Reactor Vessel Integrity

Reasonable assurance of vessel integrity is provided because proven fabrication techniques are used and because well-characterized steels, which exhibit uniform properties and consistent behavior, are used. The characterization of these materials was established through industrial and governmental studies that examined the prefabrication material properties through irradiated service operation. Inservice inspection and material surveillance programs are also conducted during the service life of the vessel, which provide further reasonable assurance that vessel integrity is maintained. The reactor vessel material satisfies the screening criteria in 10 CFR 50.61 as described in Subsection 5.3.2.3, so reasonable assurance of the reactor vessel integrity against pressurized thermal shock events is provided. The reactor vessel is also designed, fabricated, erected, and tested to conform with the requirements of 10 CFR Part 50, Appendix A (GDC 1, 4, 14, 30, 31, and 32), 10 CFR 50.60, 10 CFR 50.55a, and 10 CFR Part 50, Appendices G and H.

5.3.3.1 Design

Table 5.2-1 contains the applicable design codes. A schematic of the reactor vessel is shown in Figure 5.3-8. Additional information on the design is provided in Subsection 5.3.1.2. The design permits all required inspections to be performed and does not preclude access to areas requiring inservice inspection in conformance with ASME Section XI, as detailed in Subsection 5.2.4.

Reactor Vessel

The reactor vessel is designed to operate for 60 years. The design temperature and pressure are 343.3 °C (650 °F) and 175.8 kg/cm^{2}A (2,500 psia), respectively.

The vessel consists of three cylindrical shell sections (upper, intermediate, and lower) and a bottom head. The upper, intermediate, and lower shell sections and bottom head forging are welded together to form a complete vessel assembly. There is no weld in the core beltline regionto enhance the prevention of brittle fracture.

The upper shell including vessel flange is a forged ring with a machined ledge on the inside surface for supporting the reactor vessel internal (RVI) and the core. In addition, four keyways are machined on the ledge to accommodate the RVI alignment keys. The vessel flange is machined to provide a mating surface for the reactor vessel closure seals and is drilled and tapped to engage the closure studs.

The bottom head is a single hemispherical forging. The bottom head contains 61 ICI penetration nozzles and four external shear key supports that mate with the keyway in the

RV support column base plate. A flow skirt is also attached inside the bottom head. The flow skirt is a right circular cylinder and is located in the reactor vessel lower head below the RVI. Flow holes in the skirt are distributed to direct and make a uniform coolant flow to the core.

The closure head is fabricated separately and is joined to the reactor vessel by bolting. The closure head consists of a head flange, a dome, and 101 CEDM nozzles including 8 spare nozzles, 2 heated junction thermocouple (HJTC) nozzles, a reactor coolant gas vent system (RCGVS) line, and pads for the integrated head assembly (IHA). The head flange is a forged ring and is drilled to match the vessel flange stud holes, and the lower surface of the flange is machined to provide a mating surface for the vessel closure seals. The dome is a single hemispherical forging.

The 54 closure stud bolts screw into the tapped holes in the vessel flange, and the nuts installed on the closure head are screwed onto the studs to provide a compressive load on the head for sealing. To provide uniform loading, the studs are hydraulically tensioned with the pre-load in the prescribed sequence.

Reasonable assurance is provided by the RV sealing by the use of two silver- plated NiCrFe alloy O-rings between the vessel and closure head flanges. The O-rings are hollow with openings to accommodate reactor coolant pressure. This configuration results in a self-energized seal that gets tighter as system pressure increases. The outer ring acts as a backup for the inner ring. A connection to the space between the seals is provided to monitor the integrity of the seals.

The core stop lugs are installed on the internal surface of the reactor vessel bottom head to catch the core and RVI in the event of normal support failures.

The core stop lugs and flow skirt are arranged to minimize resistance to the flow and not to interfere with the core stabilizing lug shim installation.

The core stabilizing lugs, which function to limit the horizontal movement of the core and RVI, are also installed on the internal surface of the reactor vessel lower shell.

The external seal ledge of the RV flange provides the mating surface for the reactor cavity pool seal, which provides the bottom surface of the cavity to fill water for refueling operation.

Reactor Vessel Supports

The RV is supported by four vertical columns located under the vessel inlet nozzles. A nozzle foot with a flange under each inlet nozzle provides a surface to which the column is bolted. The flange also acts as a horizontal key to positively locate the vertical centerline of the RV. The key is designed to mate with the building structure and allows free radial growth of the RV during thermal expansion while supporting the vessel horizontally during an earthquake, a postulated pipe break, and IRWST discharge.

The bottom of each column ends in a base plate drilled to accept anchor bolts. Preloaded anchor bolts are the mechanism by which column loads are transmitted to the building structure. The base plate also acts as a keyway for the shear key welded on the RV bottom head. The key and keyway provide horizontal support and limit motion of the bottom head due to an earthquake, postulated pipe break, and IRWST discharge.

Reactor Vessel Nozzles

There are two 106.7 cm (42 in) ID primary outlet nozzles and four 76.2 cm (30 in) ID primary inlet nozzles spaced equally around the vessel and welded to the intermediate shell of the RV. Each of four inlet nozzles has one foot below the nozzle attached with a bolting pad that serves as the support interfaces.

Four 21.6 cm (8.5 in) ID direct vessel injection (DVI) nozzles are welded to the intermediate shell above the centerline of the inlet and outlet nozzles to which the safety injection system (SIS) pipes are connected.

Sixty-one ICI nozzles are fitted in the bottom head penetration holes. ICI nozzles are made of Alloy 690TT and are welded by J-groove welds with Alloy 52/52M and/or 152 weld metals on the bottom head.

One hundred and one CEDM nozzles and two HJTC nozzles are fitted in the closure head penetration holes. They are welded by J-groove weld with NiCrFe alloy on the closure head.

5.3.3.2 Materials of Construction

The materials used in the construction of the reactor vessel, as listed in Table 5.2-2, are in accordance with ASME Section III. Detailed information about the reactor vessel materials including weld materials are described in Subsections 5.2.3 and 5.3.1.

Material requirements of ASME Section III NB and the applicable ASME Section II material specifications are applied to reactor vessel pressure boundary materials. The materials satisfy the fracture toughness requirements of 10 CFR Part 50, Appendix G. In addition, weld materials satisfy applicable ASME Section IX requirements.

The reactor vessel closure head and upper, intermediate, and lower shells including flanges, bottom head dome, and major nozzles are low-alloy steel that conforms to ASME Section II SA-508 Grade 3, Class 1. Austenitic stainless steel or nickel base alloy are used for the cladding materials of the reactor vessel.

The reactor vessel stud bolts are described in Subsections 5.3.1.7 and 5.3.3.8.

Material selection for reactor vessel is based on a consideration of strength, fracture toughness, fabrication, radiation embrittlement characteristics, operating experiences, and compatibility in the PWR environmental condition.

Appropriateness of the material that has been selected has been proven by successful long-term operating experience of reactor vessels at Korean nuclear power plants as well as experience throughout the world.

5.3.3.3 Fabrication Methods

Fabrication of the reactor vessel is described in Subsection 5.3.1.2. Fabrication processes used in the construction of the reactor vessel conform with ASME Section III and ASME

NQA-1. No special manufacturing methods that could compromise the integrity of the reactor vessel are used.

More than 10 reactor vessels in operation for many years or those being constructed in Korea were and are fabricated using similar processes.

The reactor vessel is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical upper closure head. Fabrication of the reactor vessel is based on drawings, fabrication procedures, and examination procedures in accordance with ASME Section III requirements. The reactor vessel shell consists of three shell sections (upper, intermediate, and lower) and a lower head. The length of each shell is adjusted so that the shell does not include the welds in the active core region.

The reactor vessel shells are joined by circumferential welds. Welding procedures applied to the welds of the reactor vessel pressure boundary are qualified in accordance with ASME Section III and Section IX requirements.

Welding processes such as GTAW, SMAW, FCAW, and SAW are used for the reactor vessel. Electroslag welding is not used for the reactor vessel. Detailed information on the welding of reactor vessel is described in Subsection 5.3.1.2.

Other fabrication processes including cutting, bending, drilling, and forming are performed in accordance with manufacture's procedures and/or instructions.

5.3.3.4 Inspection Requirements

Nondestructive examinations (NDE) performed on the reactor vessel in accordance with ASME Section III are described in Subsection 5.3.1.3.

Ultrasonic examination, and magnetic particle or liquid penetrant methods are performed for reactor vessel pressure boundary forgings and stud bolts in accordance with the requirements of ASME Section III and Section V(Reference 31), as applicable.

Pressure boundary welds of the reactor vessel are examined according to ASME Section III, and satisfy applicable acceptance requirements of ASME Section III. Examinations are performed in accordance with the methods in ASME Section V, if required by ASME Section III.

5.3.3.5 Shipment and Installation

The requirements of ASME NQA-1 (Reference 32) are followed for the packaging and shipment of the reactor vessel. Reactor vessels are prepared to be shipped by barge or rail to the site while mounted on the shipping skid used for installation. The vessels are protected by closing all openings (including the top of the vessel) with shipping covers. The closure heads are shipped with separate skids and covers. Vessel surfaces and covers are sprayed with a strippable coating or wrapped with shrink-wrap for protection against corrosion during shipping and installation.

Following the application of the removable coating or equivalent protection material on the interior surface and prior to wrapping or coating the exterior, all openings are sealed to provide protection against dust, moisture, and/or detrimental materials during shipment and storage at the site.

After the removable coating or equivalent protection material is applied, a desiccant is applied to the interior of the closure head in the area surrounding the nozzles, and the openings in the cover are resealed.

The reactor vessel is cleaned and protected from contamination prior to shipment as described in Subsection 5.3.1.4. Additional information on cleanliness and protection against contamination for austenitic stainless steel materials is provided in Subsection 5.2.3.4.

After arrival at the site and prior to installation, all covers, plugs, and similar items are left in place until they are installed unless otherwise specified. Prior to the welding of interconnecting piping and installation of insulation, the temporary protection material is removed. Only enough strippable protection material for installation requirements is removed. The balance remains in place until just prior to installation of the insulation of the reactor vessel.

The insulation for the reactor vessel is designed to have an annular flow path suitable for the external reactor vessel cooling (ERVC) operation during a severe accident.

5.3.3.6 Operating Conditions

Refer to Sections 3.9 and 4.4 for information on design transients and operating conditions, respectively. Pressure-temperature limitations (P-T limits) and protection against pressurized thermal shock issues for the reactor vessel are described in Subsection 5.3.2. P-T limits, which are described in Subsection 5.3.2.1, meet the requirements of 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements," during operating conditions such as hydrostatic pressure tests or normal operation including anticipated operational occurrences.

Reactor vessel integrity is verified for events or transients causing severe overcooling concurrent with or followed by significant pressure called pressurized thermal shock (PTS).

For such an event of the reactor vessel integrity, the beltline materials satisfy the PTS screening criteria in 10 CFR 50.61, which are described in Subsection 5.3.2.3.

5.3.3.7 Inservice Surveillance

Inservice Inspection

ASME Section XI requirements are followed for inservice inspection of the reactor vessel. Preservice inspections are also conducted after installation in accordance with applicable ASME Section III and Section XI requirements.

Inservice inspection for the APR1400 is described in Subsection 5.2.4. A detailed list of inservice and preservice inspections for the reactor vessel is described in Table 5.3-8 and Table 5.3-9. The COL applicant is to provide and develop the inservice inspection and testing program for the RCPB in accordance with ASME Section XI and 10 CFR 50.55a (COL 5.3(4)). The inservice inspection plan is implemented every 10 years. Accessibility to equipment for maintenance, testing, and inspection is a basic element of the APR1400 design process.

The outer surface of closure head and closure head nozzles beyond the peripheral nozzles can be accessed by removing the head removable insulation panels. Visual inspections around each nozzle on the head surface can also be applied using remote inspection devices. The head insulation is to maintain the gap from the surface of top head by approximately 2.5 inches to allow access for the inspection devices. The closure head is available for inspection whenever it is removed, and its removal makes available the vessel closure flange, the flange-to-shell weld, closure stud holes and ligaments, and the closure studs and nuts. Each control element drive mechanism is removable as a unit through a closure at the top of its housing. Because many of the reactor vessel closure head examinations are done from the underside of the head, the head laydown area provides access for examinations.

The closure head stud bolts, nuts, and washers are inspected periodically using visual magnetic particle and/or ultrasonic examinations.

All internals of the reactor vessel (which is an open structure offering insignificant impediment to access) are removable, making the entire inner surface of the vessel including the beltline welds, and the weld zones of the internal load-carrying structure attachments, available for the required surface and volumetric inspections. For interim inspections of the vessel, nozzle-to-shell welds and inner radii of the outlet nozzles are accessible from inside the reactor vessel by using remote automated equipment without removing the vessel internals.

The outside surface of the bottom head of the reactor pressure vessel is manually examined from outside of the vessel, and an access tunnel is therefore provided to allow personnel into the area below the bottom head. Insulation is provided by removable panels over the bottom head weld seams.

Material Surveillance Program

The reactor vessel surveillance program conforms to the requirements of Appendix H of 10 CFR Part 50 and ASME Section XI as described in Subsection 5.3.1.6. The design of the program is based on ASTM E185.

Tensile test specimens and Charpy impact test specimens are included to corroborate the post-irradiation surveillance data and fracture toughness specimens to enable determination of fracture toughness properties before and after irradiation. The number of standard Charpy specimens that are required by ASTM E185 are included in reactor surveillance program to increase the accuracy in defining post-irradiation index temperatures.

When combined with the use of highly radiation-resistant materials in the beltline of the reactor vessel, the surveillance program provides reasonable assurance of the integrity of the reactor vessel in terms of strength and fracture resistance.

5.3.3.8 Threaded Fasteners

The bolting material for the reactor vessel closure head is fabricated from SA 540 B24 Class 3 material. This material conforms to the requirements of 10 CFR Part 50, Appendix G and the intent of NRC RG 1.65 (Reference 33). Nondestructive examination is performed according to ASME Section III, NB-2580, during the manufacturing process.

Detailed information concerning the reactor vessel stud bolts and examinations applied to provide reasonable assurance of the integrity of the bolts is provided in Subsections 5.3.1.7 and 3.13.

The reactor vessel bolting materials that are used to provide reasonable assurance of the integrity of bolts are described in Subsection 5.3.1.7 and Section 3.13.

5.3.4 Combined License Information

COL 5.3(1) The COL applicant is to provide a reactor vessel material surveillance program for a specific plant.

COL 5.3(2) The COL applicant is to develop P-T Limit curves based on plant-specific data.

COL 5.3(3) The COL applicant is to verify the RT_{PTS} value and the USE at EOL based on plant-specific material property and neutron fluences.

COL 5.3(4) The COL applicant is to provide and develop the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a.

5.3.5 References

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

2. 10 CFR 50.55a, "Codes and Standards," U.S. Nuclear Regulatory Commission.

3. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation" U.S. Nuclear Regulatory Commission.

4. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," U.S. Nuclear Regulatory Commission.

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Table 5.3-1

Total Quantity of Specimens

Type of Specimen	Orientation	Number of Specimens			
		Base Metal	Weld Metal	HAZ (1)	Total
Drop weight	Transverse	12	12	12	36
Standard Charpy	Longitudinal	114	-	-	114
	Transverse	114	114	114	342
Precracked Charpy	Longitudinal	66	-	-	66
	Transverse	66	66	-	132
Compact tension	1T longitudinal	4	-	-	4
	1/2T longitudinal	28	-	-	28
	1T transverse	8	8	-	16
	1/2T transverse	28	28	-	56
Tensile test	Longitudinal	30	-	-	30
	Transverse	30	30	-	60
	Total	500	258	126	884

(1) Heat-affected zone

Table 5.3-2

Type and Quantity of Specimens for Baseline Tests

Type of Specimen	Orientation	Number of Specimens			
		Base Metal	Weld Metal	HAZ [(1)]	Total
Drop weight	Transverse	12	12	12	36
Standard Charpy	Longitudinal	24	-	-	24
	Transverse	24	24	24	72
Precracked Charpy	Longitudinal	12	-	-	12
	Transverse	12	12	-	24
Compact tension	1T longitudinal	4	-	-	4
	1/2T longitudinal	4	-	-	4
	1T transverse	8	8	-	16
	1/2T transverse	4	4	-	8
Tensile test	Longitudinal	12	-	-	12
	Transverse	12	12	-	24
	Total	128	72	36	236

(1) Heat-affected zone

Table 5.3-3

Type and Quantity of Specimens for Irradiated Tests

Type of Specimen	Orientation	Number of Specimens			
		Base Metal	Weld Metal	HAZ [(1)]	Total
Standard Charpy	Longitudinal	90	-	-	90
	Transverse	90	90	90	270
Precracked Charpy	Longitudinal	54	-	-	54
	Transverse	54	54	-	108
Tensile test	Longitudinal	18	-	-	18
	Transverse	18	18	-	36
1/2T Compact Tension	Longitudinal	24	-	-	24
	Transverse	24	24	-	48
	Totals	372	186	90	648

(1) Heat-affected zone

Table 5.3-4

Type and Quantity of Specimens Contained
in Each Irradiation Capsule Assembly

Type of Specimen	Number of Specimens				
	Standard Charpy	1/2T CT	Precracked Charpy	Tension	Total
Base metal (longitudinal)	15	4	9	3	31
Base metal (transverse)	15	4	9	3	31
Weld metal	15	4	9	3	31
Heat-affected zone	15	-	-	-	15
Total	60	12	27	9	108

Table 5.3-5

Material for Neutron Threshold Detectors

Material	Reaction	Threshold Energy (MeV)	Half-Life
Niobium	^{93}Nb (n, n`) ^{93m}Nb	0.5	16.4 years
Neptunium	^{237}Np (n, f) ^{137}Cs	0.5	30.2 years
Uranium	^{238}U (n, f) ^{137}Cs	1.5	30.2 years
Iron	^{54}Fe (n, p) ^{54}Mn	2.2	312.5 days
Nickel	^{58}Ni (n, p) ^{58}Co	2.1	70.78 days
Copper	^{63}Cu (n, α) ^{60}Co	5.0	5.27 years
Titanium	^{46}Ti (n, p) ^{46}Sc	4.4	83.83 days
Cobalt	^{59}Co (n, γ) ^{60}Co	Thermal	5.27 years

Table 5.3-6

Composition and Melting Points
of Candidate Materials for Temperature Monitors

Composition, wt%	Melting Temperature, °C (°F)
Cd - 17.4 Zn	266 (511)
Au - 20.0 Sn	280 (536)
Pb - 2.5 Ag	304 (580)
Pb - 1.75 Ag - 0.75 Sn	310 (590)

Table 5.3-7

Capsule Assembly Removal Schedule

Capsule	Azimuthal Location	Removal Time(1)	Target Fluence (n/cm^2)
A	217 °	6 EFPY	-
B	37 °	15 EFPY	-
C	224 °	32 EFPY	-
D	323 °	EOL	6.44×10^{19} (2)
E	44 °	Standby	-
F	143 °	Standby	-

(1) Schedule may be modified to coincide with the refueling outages or scheduled shutdowns most closely approximating the withdrawal schedule.

(2) Expected best estimated fluence level at the end of the plant design life (interface between reactor wall and cladding)

Table 5.3-8

Inspection Plan for Reactor Vessel Materials

	Requirements			
	Procurement	Manufacturing	Preservice	Inservice
Forgings: Shell and Flange	MT UT	MT (1)	-	-
Forgings: Closure Head	MT UT	MT (1)	VT (2)	VT(2)
Stud Bolts, Nuts and Washers	MT UT	MT (1)	UT (3) VT (3)	UT (3) VT (3)
Forgings: Vessel Nozzle	MT UT	MT (1)	-	-
Closure Head Nozzle	PT UT	PT (1)	ECT (4)	-
Closure Head Vent Pipe	PT UT	PT (1)	ECT (4)	-
Attachments: Ferritic Material	UT and MT	MT (1)	-	-
Attachments: Stainless Steel and A690 Alloy	UT and PT	PT (1)	-	-

ECT =Eddy current testing
MT = Magnetic particle testing
PT = Dye-penetrant testing
RT = Radiographic testing
UT = Ultrasonic testing
VT =Visual testing

(1) Only for machined surfaces

(2) Only for closure head dome portion

(3) UT applicable for stud bolts, VT applicable for nuts and washers
for stud bolts: UT
for nuts and washers: VT

(4) Only for inner surfaces of Alloy 690 materials

Table 5.3-9

Inspection Plan for Reactor Vessel Welds

	Requirements		
	Manufacturing	Preservice	Inservice
Full penetration welds : Pressure boundary ferritic forgings	RT UT MT	UT PT (1)	UT PT (1)
J-groove welds and vent pipe head nozzles	PT (2)	ECT	ECT
DVI nozzle to safe end welds	RT UT PT	UT PT	UT PT
Attachment welds to pressure boundary	MT or PT	-	-
Cladding	UT PT	VT	VT

(1) Only for closure head to flange welds

(2) Every 1.3 cm (0.5 in) thickness of weld and final weld surface

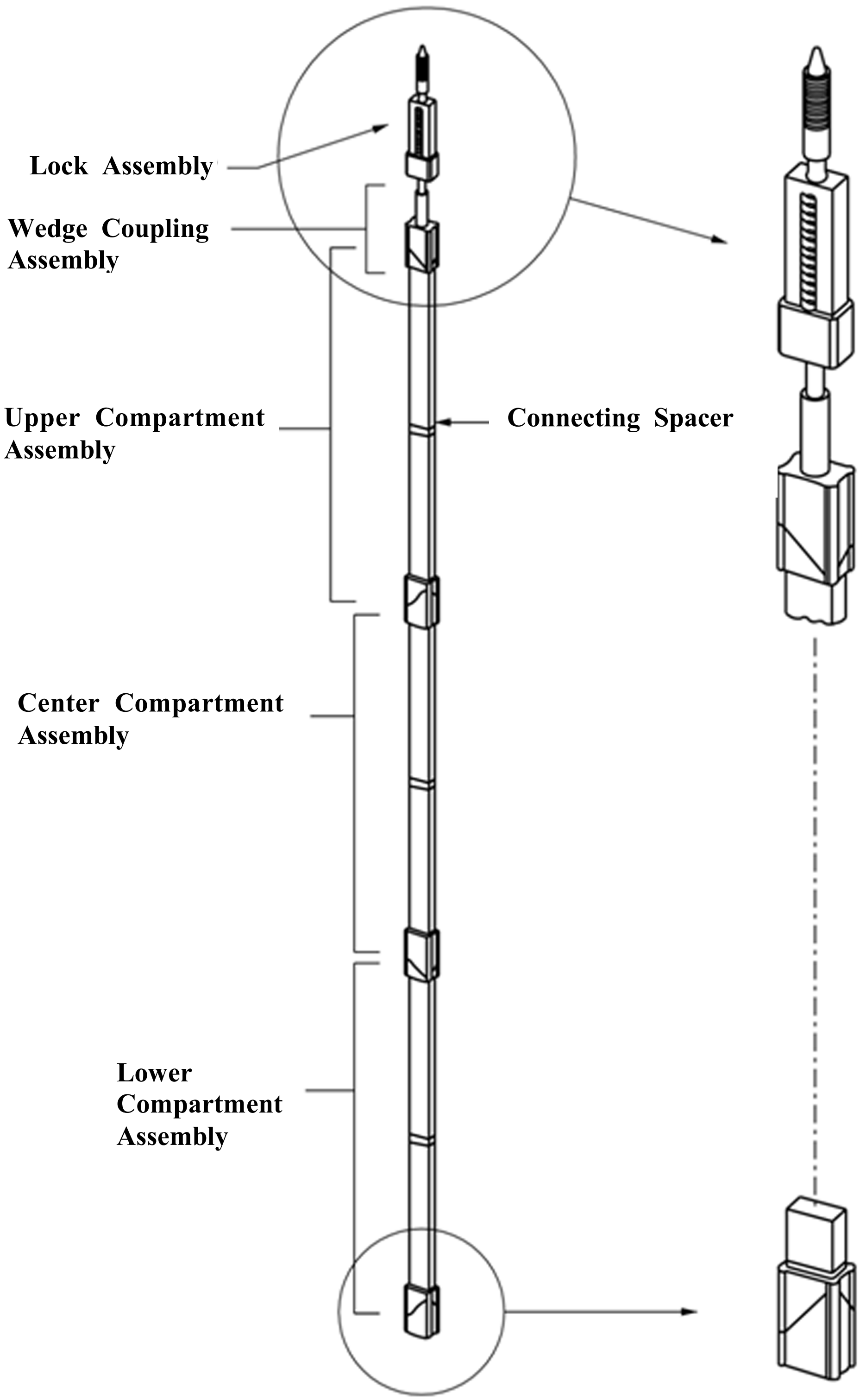

Figure 5.3-1 Typical Surveillance Capsule Assembly

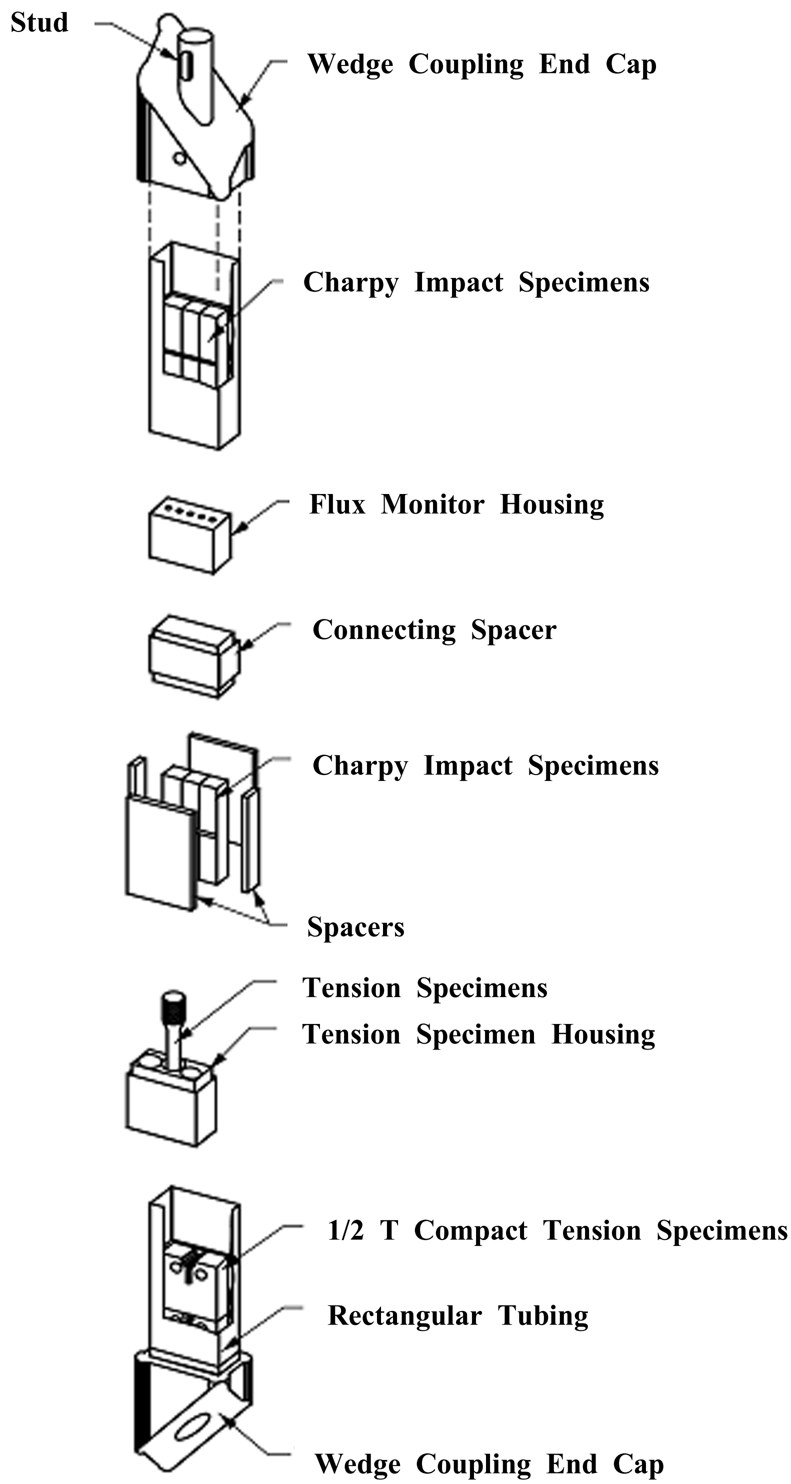

Figure 5.3-2 Upper Compartment Assembly

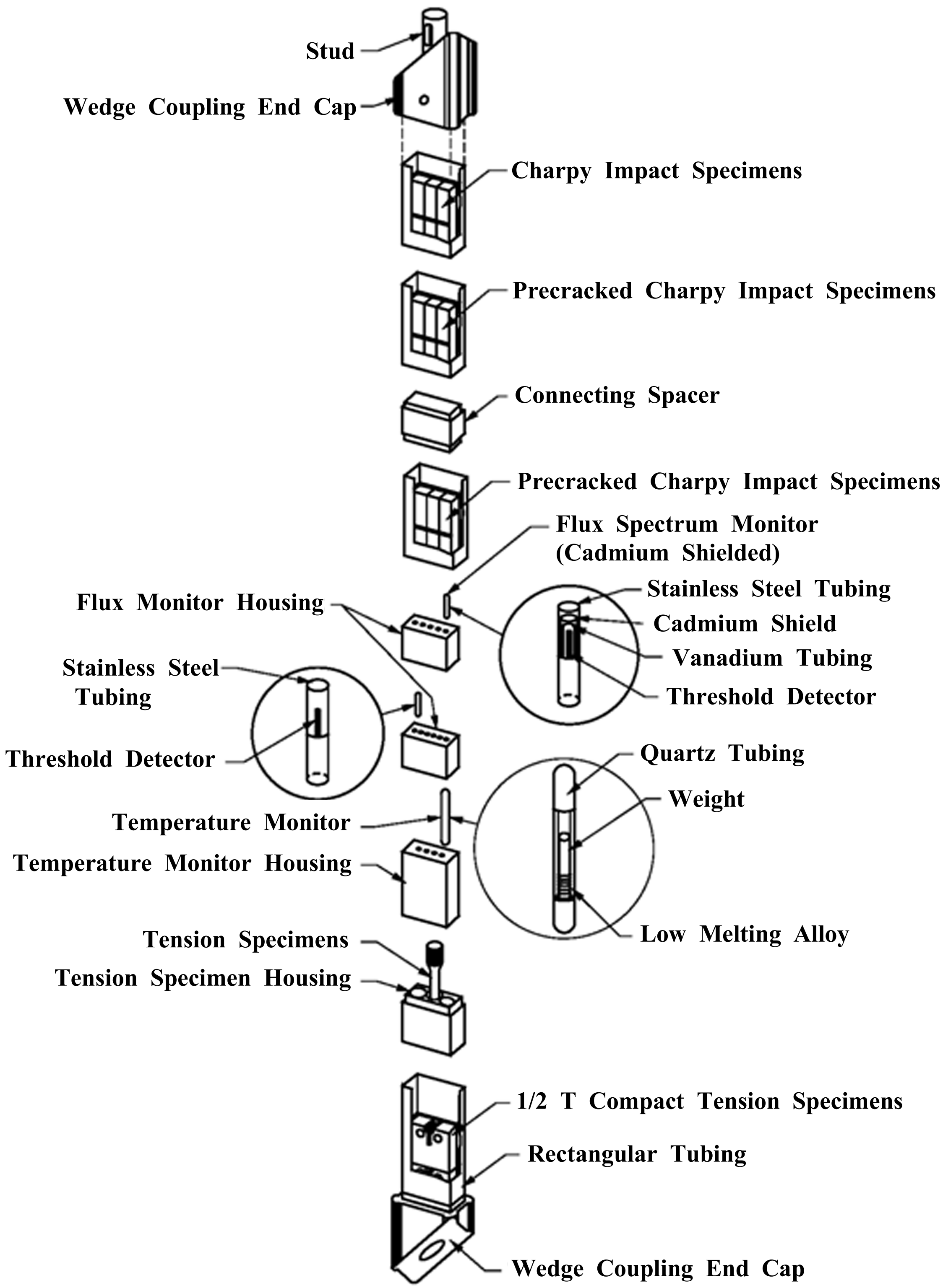

Figure 5.3-3 Center Compartment Assembly

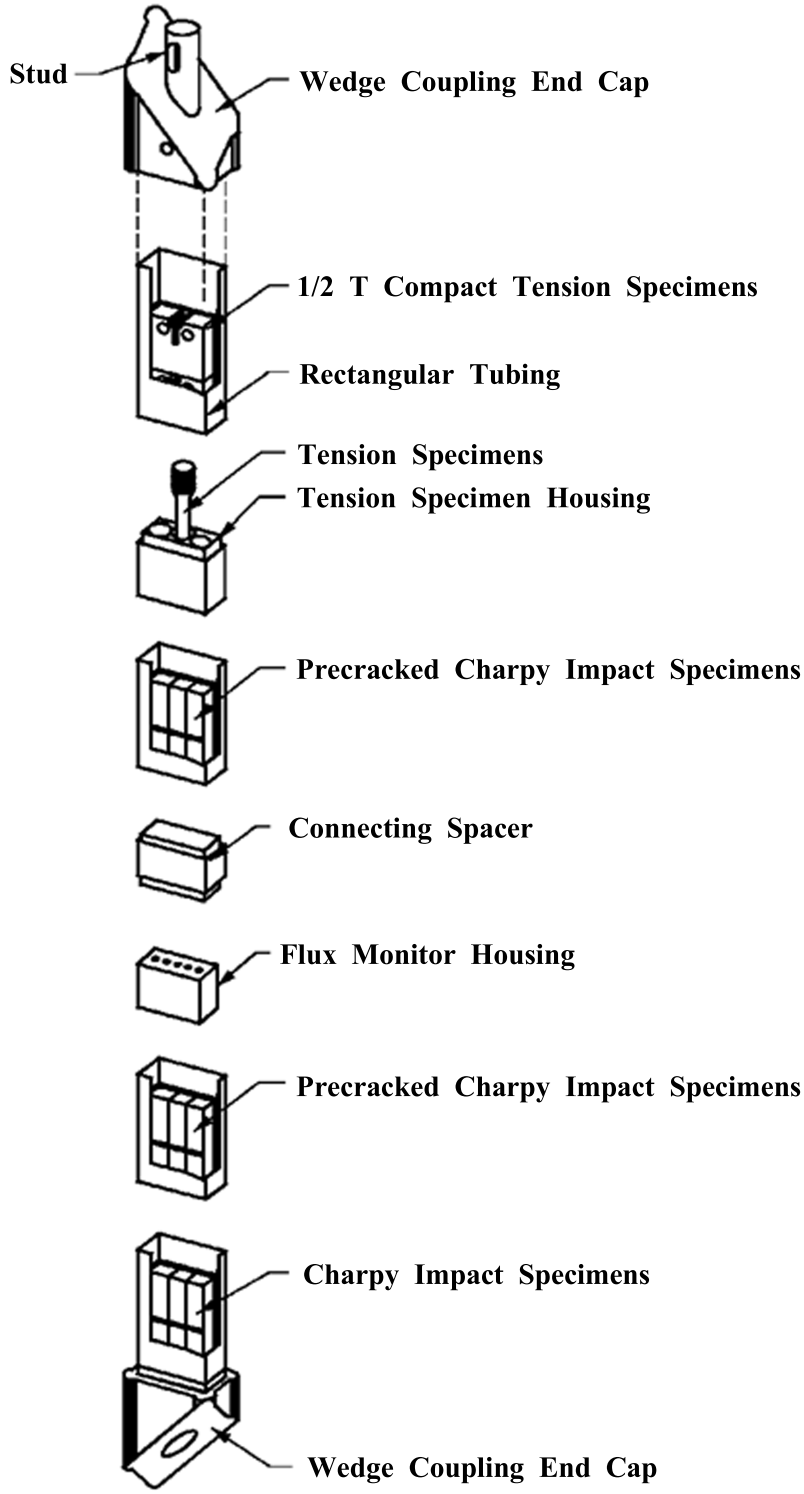

Figure 5.3-4 Lower Compartment Assembly

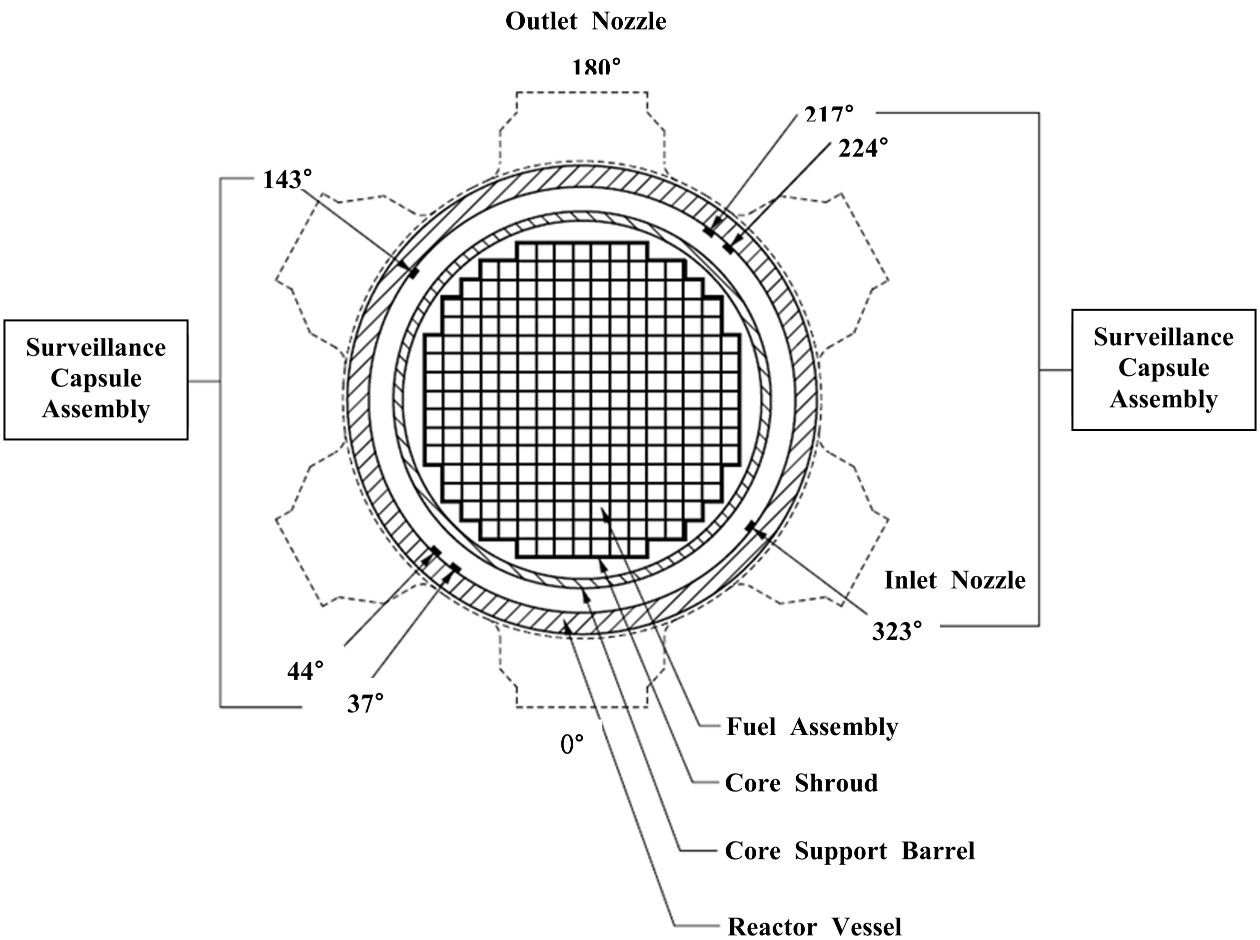

Figure 5.3-5 Location of Surveillance Capsule Assemblies (Plan View)

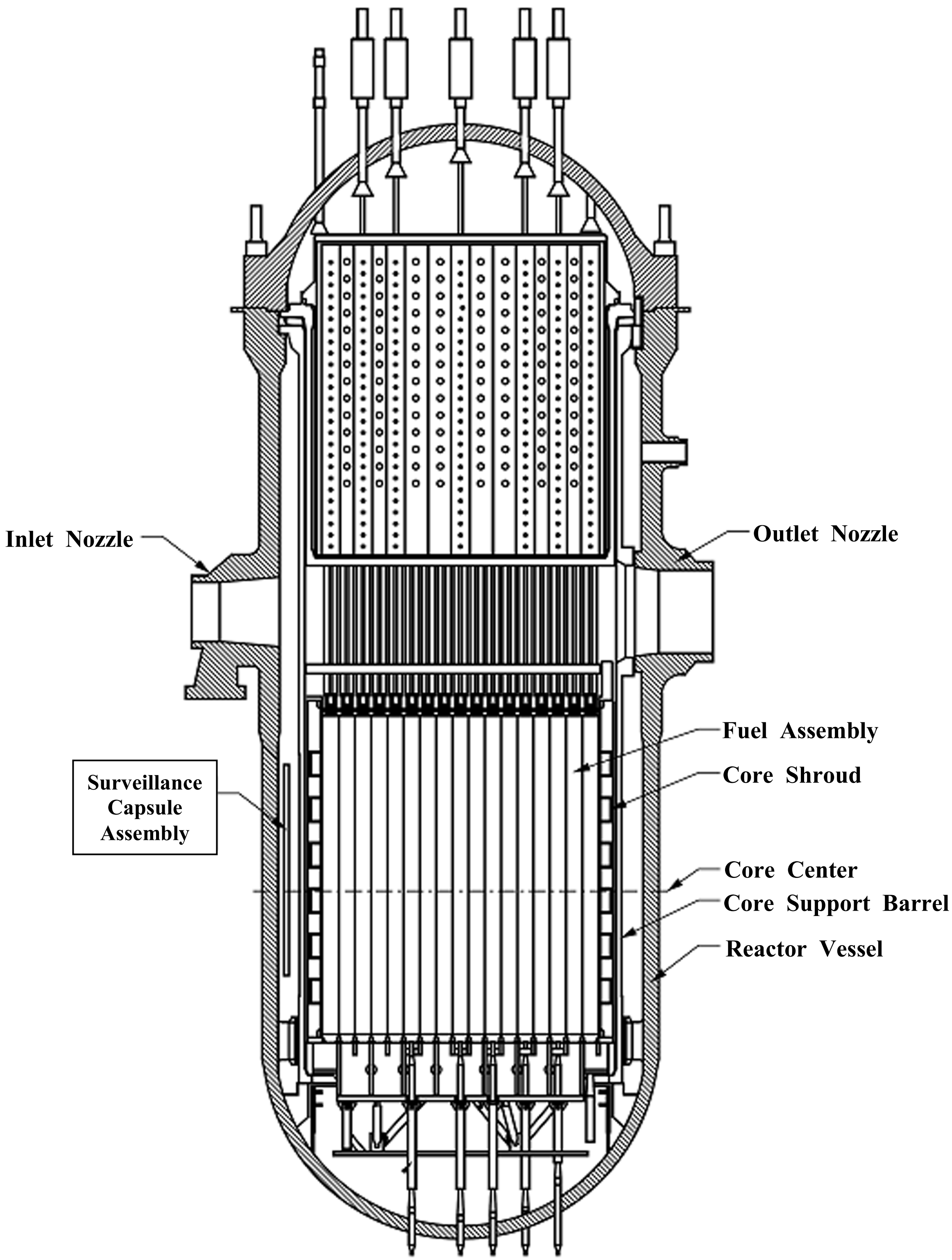

Figure 5.3-6 Location of Surveillance Capsule Assemblies (Elevation View)

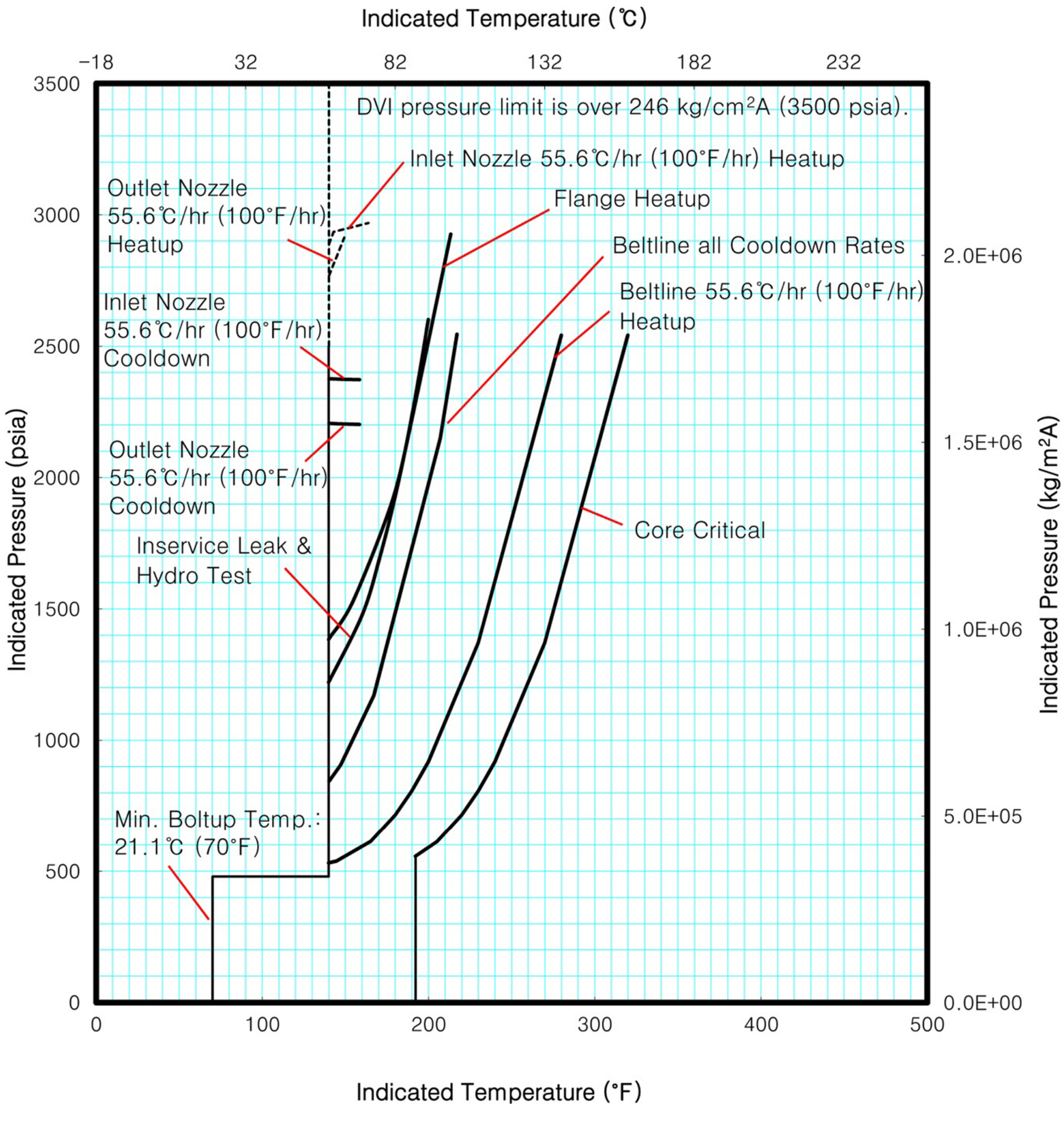

Figure 5.3-7 Pressure-Temperature Limit Curve (60 years)

NOZZLE SCHEDULE	
SERVICE	
COOLANT INLET	4
COOLANT OUTLET	2
CEDM AND INSTRUMENTATION NOZZLES	103
IN-CORE INSTRUENTATION	61
VENT-RCGVS	1
SEAL LEAK MONITOR	1
DIRECT VESSEL INJ.	4

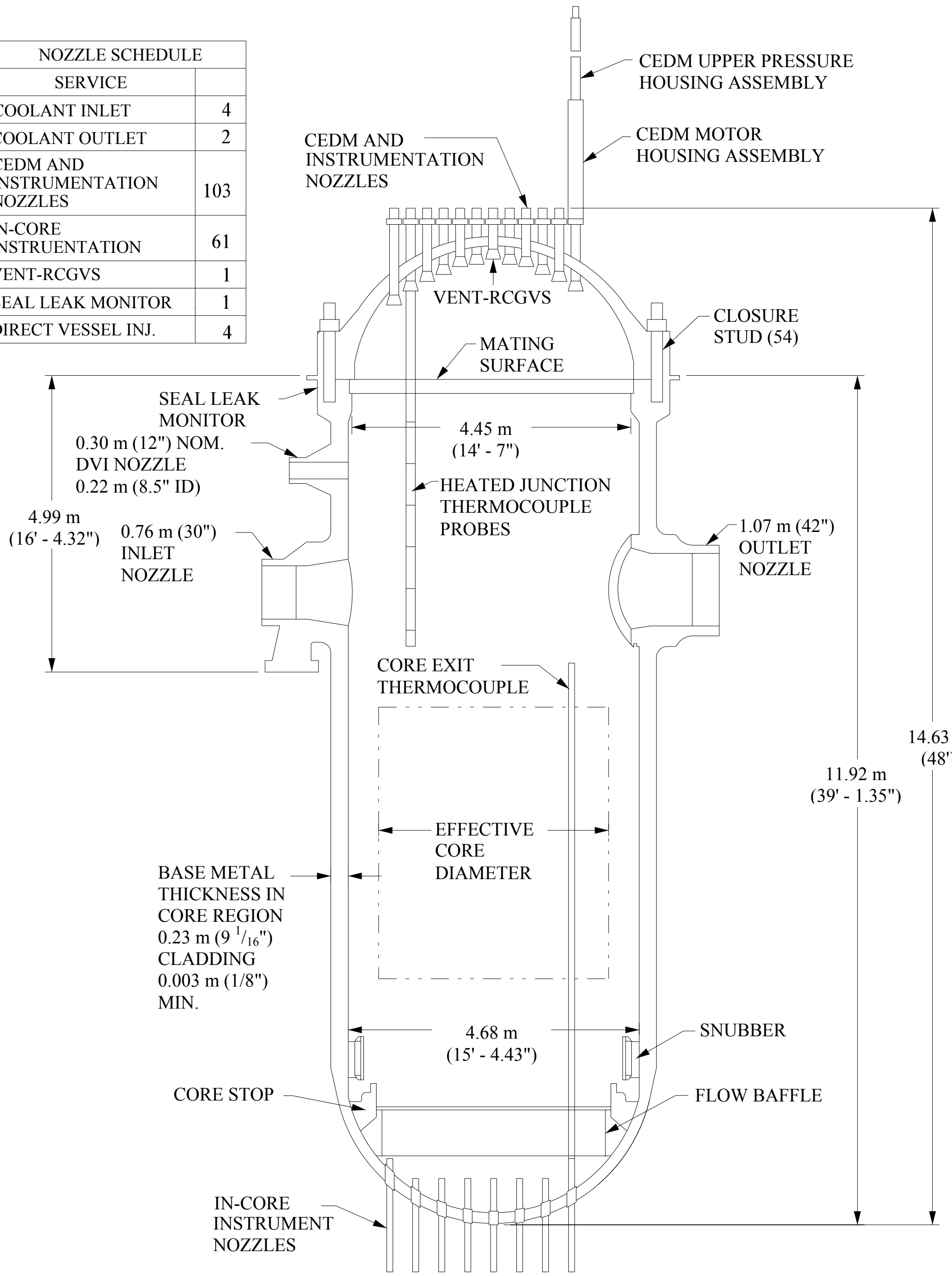

Figure 5.3-8 Reactor Vessel Assembly

5.4 Reactor Coolant System Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

The reactor coolant pumps provide sufficient forced circulation flow through the reactor coolant system (RCS) to provide reasonable assurance of adequate heat removal from the reactor core during power operation. A low limit on reactor coolant pump flow rate (i.e., design flow) is established to provide reasonable assurance that specified acceptable fuel design limits (SAFDLs) are not exceeded. Design flow is derived on the basis of the thermal-hydraulic considerations presented in Subsection 4.4.4.5.1.

The reactor coolant pump and motor assembly in conjunction with the flywheel provide sufficient coastdown flow following loss of power to the pumps to provide reasonable assurance of adequate core cooling.

The reactor coolant pump pressure boundary is designed for the transients given in Table 3.9-1 so the ASME Section III (Reference 1) allowable stress limits are not exceeded for the specified number of cycles. Stress criteria concerning earthquake and pipe rupture conditions are presented in Subsection 3.9.3.

The design overspeed of the reactor coolant pump is 125 percent of normal speed.

5.4.1.1 Pump Flywheel Integrity

The RCP flywheel maintains the integrity to prevent the possibility of producing high-energy missiles and excessive vibration of the pump assembly under standstill, normal and anticipated operating condition consistent with the intent of General Design Criteria 1 and 4 (Reference 2) and Regulatory Guide 1.14 (Reference 3).

The flywheel uses a shrink fit design to couple it to a shaft. The stresses in the flywheel are a function of three attributes which are the material, the shrink fit, and the rotational speed.

5.4.1.1.1 Material Selection and Fabrication

The material used to manufacture the flywheel is produced by a commercially acceptable process that minimizes flaws such as the vacuum melting and degassing process. The flywheel material is a quenched and tempered forging with the German material designation 26NiCrMoV14-5, which is a high strength ductile forged material. This material meets the requirements described in Subsection 5.4.1.1.2 and 5.4.1.1.3, and its mechanical properties are equal to or exceed SA-508 Class 2, which is a typical U.S. forged flywheel material. One of the attributes that led to selection of this material is its resistance to non-ductile-type failures. This provides adequate fracture toughness properties under operating conditions. No welding is performed on the flywheel. If the flywheel is flame cut, at least 13 mm (1/2 inch) of stock is left on the outer and bore radii for machining to final dimensions.

5.4.1.1.2 Fracture Toughness

The K_{IC} of the flywheel material at the normal operating temperature of the flywheel is greater than 165 MPa√m (150 ksi√in). Conformance is demonstrated by an indirect test.

Justification is provided to establish the equivalence of fracture toughness in the proposed flywheel material and certain steels (ASME SA-533-B Class 1, ASME SA-508 Class 2, ASME SA-508 Class 3, and ASME SA-516 Grade 65). The RT_{NDT} of the flywheel materials is determined in accordance with NB-2320 and NB-2330 of the ASME Section III.

5.4.1.1.3 Design

The flywheel is designed to withstand normal conditions, anticipated transients, loss of coolant accident with the largest mechanical pipe break remaining after application of leak before break as described in Subsection 3.6.3, and a safe shutdown earthquake (SSE) without loss of structural integrity. The flywheel integrity analysis is summarized in the technical report APR1400-A-M-NR-14001-P (Reference 4) which is performed for a stress analysis at standstill, normal and overspeed conditions and a fracture mechanics analysis to predict the critical speed for fracture of the flywheel.

The design overspeed of flywheel is 125 percent of the synchronous speed of the RCP motor. The design overspeed is at least 10 percent greater than the highest anticipated overspeed of the pump. The highest anticipated overspeed is predicted for loss of coolant accident with the largest break size remaining after the application of leak before break, as described in Subsection 3.6.3. The largest break size remaining after the application of leak before break that may affect the maximum overspeed of the RCP is a 10.16 cm (4 in) pressurizer spray line.

The shaft and bearings supporting the flywheel are able to withstand any combination of the loads of normal operation, anticipated transients, loss of coolant accident with the largest mechanical pipe break remaining after application of leak before break as described in Subsection 3.6.3, and an SSE.

The flywheel is accessible for 100 percent in-place volumetric ultrasonic inspections. The flywheel-motor assembly is designed to allow such inspection with a minimum of motor disassembly.

5.4.1.1.4 Test and Inspection

a. Each flywheel is tested at the design overspeed.

b. The flywheel is subjected to a magnetic particle or liquid-penetrant examination per ASME Section III before final assembly. The inspection is performed on areas of high stress concentrations.

c. Each finished flywheel is subjected to a 100 percent volumetric ultrasonic inspection from the flat surface as per ASME Section III. This inspection is performed on the flywheel after final machining and the overspeed test.

d. The inservice inspection program includes ultrasonic examinations of the areas of high stress concentration at the bore and keyway at about three and one-third year intervals, during the refueling or maintenance shutdown coinciding with the inservice inspection schedule as required by ASME Section XI (Reference 5). Removal of the flywheel is not required.

e. A surface examination of all exposed surfaces and a 100 percent volumetric examination by ultrasonic methods are conducted at approximately 10-year intervals during the plant shutdown coinciding with the inservice inspection schedule as required by ASME Section XI.

f. Each flywheel receives a preservice baseline inspection that incorporates the methods defined above for an inservice inspection. Examination procedures and acceptance criteria are determined in accordance with ASME Section III.

5.4.1.2 Description

Table 5.4.1-1 lists the principal parameters of the reactor coolant pumps, and Figure 5.4.1-1 depicts the arrangement of the pump and motor. Reactor coolant pump supports are described in Subsection 5.4.15. The flow diagram for the reactor coolant pump is given in Figure 5.1.2-2.

The four reactor coolant pumps are vertical, single stage, bottom suction, horizontal discharge, motor-driven centrifugal pumps. The pump impeller is splined and locked to its shaft. Pump shaft alignment is maintained by a water lubricated radial bearing within the pump and by radial and thrust bearings located in the motor stand. The pump and motor shafts are directly connected by a coupling.

The pump rotating assembly is mounted in a diffuser-type pump casing. The pump casing is a one-piece design in accordance with applicable sections of ASME Section III. The one-piece casing reduces the ASME Section XI examination requirements.

The pressure boundary materials used for the reactor coolant pump assembly are listed in Table 5.2-2, are compatible with the reactor coolant addressed in Subsection 5.2.3.2.1.

The shaft seal assembly consists of two face types, mechanical seals in series, with a controlled leakage bypass to provide the same pressure differential across each seal. The seal assembly is designed for the pressure differential of 175.8 kg/cm^2 (2,500 psi) and to reduce the leakage pressure from the RCS pressure to the volume control tank pressure. A third, face-type, low-pressure vapor seal at the top is designed to withstand system operating pressure when the pumps are not operating. The leakage past the second pressure seal and the controlled leakage are piped to the volume control tank in the chemical and volume control system. Leakage past the low-pressure vapor seal is collected and piped to the reactor drain tank.

The temperature of the water in the seal assembly is maintained within acceptable limits by a water-cooled heat exchanger (HX). Water is also injected into the seal area from an external seal injection system. The performance of the shaft seal system is monitored by pressure and temperature sensing devices in the seal system. The seal assembly can be replaced without draining the pump casing or removing the shaft.

The RCP shaft seals are cooled by (1) seal injection water from the CVCS and (2) the component cooling water (CCW) through a high-pressure seal cooler. Pump seal operation may continue indefinitely provided either seal injection water or the CCW is available. The APR1400 design includes an additional support system (i.e., auxiliary charging pump). This system features a positive displacement pump to provide a diverse means of seal injection to the RCPs if the normal means of seal cooling are lost. Detailed descriptions are given in Subsections 9.2.2.3 and 9.3.4.2.

In the event of loss of either seal injection to the seal assembly or loss of CCWS flow to the high-pressure seal cooler, the seal cooling water temperature increases. Performance tests and analyses have shown that a minimum margin of 12.2 °C (22 °F) exists between the seal cooling water outlet temperature and the seal cooling water temperature limit specified by the pump manufacturer.

If there is a simultaneous loss of CCW to all RCP and motor bearing assemblies but seal injection water is available to the seals, the RCP can operate for at least 30 minutes without bearing seizure, which could affect normal RCP coastdown. This is discussed further in Subsection 5.4.1.3.

The seal assemblies are designed to limit seal leakage plus controlled bypass flow per pump to approximately the following values:

All seals functioning (normal)	12.1 L/min (3.2 gpm)
One seal functioning (abnormal)	16.7 L/min (4.4 gpm)

An acoustic leak sensor is provided on the shaft seal housing for use in detecting leakage coming through in the vapor seal of the pump. A description of the acoustic leak monitoring system (ALMS) is provided in Subsection 7.7.1.5

The need for improved station blackout (SBO) performance has led to selected improvements in the design of the seal. The seal uses an enhanced manufacturing process and improved composition of elastomer materials to provide high temperature seal performance at 300 ^{o}C (572 ^{o}F) and 163.1 kg/cm^{2}G (2,320 psig). Key materials used in the seals are silicon carbide/graphite compound material for the glide rings (primary seals) and specially treated (high-temperature-resistant) ethylene propylene diene monomer (EPDM) elastomers (secondary seals). Both materials are capable of withstanding long-term exposure to a high temperature and pressure environment. Seal dimensions and normal operational characteristics are basically unchanged from those in pump assemblies in existing operating plants. As the seal is intended to withstand adverse SBO conditions, it is verified by a robust test program.

The motor is sized for continuous operation at the flows resulting from a four-pump or one-pump operation with a 1.0 to 0.74 specific gravity of water. The motors are designed to start and accelerate to speed under full load with a drop to 80 percent of normal rated voltage at the motor terminals.

Each RCP motor is equipped with two air-to-water heat exchangers such that the temperature of motor cooling outlet air discharged to the containment is less than the maximum containment ambient temperature. Electrical insulation of the motor is suitable for a high humidity and high radiation environment.

Each motor is provided with an anti-reverse rotation device. The device is designed to prevent impeller rotation in the reverse direction, assuming each of the following two conditions: (1) motor starting torque if the motor was incorrectly wired for reverse rotation and (2) reactor coolant flow through the pump in the reverse direction due to the largest pipe break remaining after the application of leak before break as described in Subsection 3.6.3.

The RCP assembly is equipped with an oil collection system to collect oil leakage. The oil collection system is capable of collecting lube oil from all potential pressurized and unpressurized leakage sites in the RCP assembly. The oil collection system is designed to withstand an SSE.

5.4.1.3 Evaluation

The reactor coolant pumps are sized to deliver flow that equals or exceeds the design flow rate utilized in the thermal hydraulic analysis of the RCS. Analysis of steady-state and anticipated transients is performed assuming the minimum design flow rate. Tests are performed to evaluate reactor coolant pump performance during the post-core load hot functional testing to verify adequate flow.

If the offsite and onsite electrical powers are lost, the RCPs are tripped and begin to coast down, and the forced coolant flow through the reactor core gradually declines. Each RCP is designed to coast down at a rate that is slow enough to prevent core damage during the first few seconds of the core flow transient. Reasonable assurance that core damage will be prevented is provided by the flywheel in each RCP motor, which provides an additional rotating inertia to extend the pump coastdown. The total rotating inertia of the pump, motor, and flywheel is no less than 6,717 kg-m^2 (159,400 lbs-ft^2). The coastdown capability of the pump is reasonably assured to be maintained following a loss of offsite or onsite electrical power combined with an SSE, as addressed in Subsection 3.2.1. Core flow transients are addressed in Subsection 15.3.1.

Leakage from the pump via the pump shaft is controlled by the shaft seal assembly. Reactor coolant entering the seal chambers is cooled and collected in closed systems to prevent reactor coolant leakage to containment. Instrumentation is provided to monitor seal operation.

The design overspeed of the flywheel is 125 percent of normal speed. A test of each flywheel at the design overspeed is performed prior to assembly to confirm that no missile is generated. Pump flywheel integrity is described in Subsection 5.4.1.1.

In the event of a break that is not eliminated by leak before break and that could result in increased flow through the pump tending to accelerate the pump impeller, the highest predicted pipe break induced overspeed is less than the lowest critical speed of the flywheel.

The pump and motor oil-lubricated bearings are lubricated by internal oil systems. Each bearing assembly has an internal oil system consisting of an oil bath or a force-feed type system. During normal operation, no external pumps are required because pumping action is accomplished by internal pumping devices. Lubricating oil is cooled by cooling coils submerged in the oil sumps. Both sumps and cooling coils are internal to the pump and motor structural frame, and are designed for seismic Category I operation. They adhere to ASME Section III, Class 3, as a guide for design and construction. Although the pump-motor assembly operation is not considered necessary for plant safety, this design minimizes the direct effects of seismic events on the reactor coolant pump and motor assembly oil lubricating systems so that adequate coastdown characteristics are not detrimentally affected.

Bearing metal temperatures, oil flow or pressure, oil levels, cooling water flow, and temperature are monitored and alarmed in the control room.

In the unlikely event that component cooling water to the reactor coolant pump and motor oil lubricating systems is not available or that an oil leak occurs during operation, the operator is alerted as soon as cooling water to the oil system is lost, and has at least 30 minutes in which to reduce power, if necessary, isolate the cooling water, and shut the reactor coolant pump motor assembly down to prevent bearing seizure. In the remote possibility of a simultaneous loss of component cooling water to all reactor coolant pump motor assemblies, 30 minutes is adequate to secure the plant and maintain the normal coastdown capabilities of the reactor coolant pump motor assemblies.

A shaft seizure due to bearing failure is unlikely during a loss of component cooling water event for the following reasons:

a. The design is such that the heat generated in the bearing, normally carried away by the cooling water, is removed through alternate paths. The lube oil sump baths surrounding the bearings, the stagnant cooling water remaining in the heat exchanger coils, and the bearing and sump assembly metal masses, all act as heat sinks. In addition, conduction down the pump shaft and radiation from the outer sump shell help reduce the temperature increase.

b. The rotation of the bearing assemblies provides reasonable assurance of adequate oil flow and mixture of heated oil so the heat transfers as described above.

c. In the event that the oil temperature increases such that the viscosity degrades significantly, the design of the thrust bearing provides reasonable assurance that the bearing will continue to produce a hydrodynamic film to preclude metal-to-metal contact.

If the event of an oil leak, redundant instrumentation alerts the operator to shut down the reactor coolant pump motor assembly, thereby avoiding bearing damage.

In the event of an oil leak, the separation of lubrication systems limits the problem to a single reactor coolant pump.

The loss of the oil in the bearing oil reservoir would not result in bearing seizure for the following reasons:

a. Temperature and oil level monitors provide appropriate indication of an abnormal condition.

b. The vibration monitoring device furnished on the pump responds to bearing degradation and allows the operator to shut down the pump.

c. If the above protective measures fail, the high torque produced by the motor causes a slow breakdown of the bearings but not a rapid shaft seizure. Industry experience indicates that the babbitt bearing surfaces wear away and the bearing pads and sleeves are badly worn but the shaft continues to rotate.

If the extremely remote possibility of bearing seizure occurs while the reactor coolant pump motor assembly is in operation, adequate flow to the core is available from the other reactor coolant pump motor assemblies as demonstrated in Subsection 15.3.3.

Figure 5.1.2-2 shows a separate oil lift system that is required for startup of the pump assembly. The oil lift system furnishes high pressure oil to the pump assembly thrust bearings, thereby lifting the rotor and reducing bearing friction during pump startup. Interlocking devices are furnished, which prevent pump startup until oil lift flow is established. The oil lift system shuts down automatically when the pump reaches full speed. Because an oil lift is not necessary during normal operation, an oil leak in this system does not cause a bearing failure.

5.4.1.4 Tests and Inspections

The reactor coolant pump pressure boundary is nondestructively inspected as required by ASME Section III for Class 1 components. The pump undergoes a hydrostatic pressure test in the vendor's shop and with the RCS. Inservice inspection of the pump pressure boundary is performed during the plant life in accordance with ASME Section XI.

The pump assembly is performance tested in the vendor's shop over at least the normal operating range in accordance with ASME PTC 8.2 (Reference 6). The tests also demonstrate the ability of the pumps to function under the various operating conditions specified. Tests commonly performed are hot and cold performance and stop-start cycling. Vibrations are monitored at several places on the pump during shop testing. In addition to meeting an absolute criterion for vibration amplitude, the test results are examined for evidence of critical speed problems.

The pump motors undergo a "routine" test in accordance with NEMA MG-1 (Reference 7). This test also confirms that the motors are within their vibration limits. One motor is used as the driver for the pump assemblies during the pump manufacturer's shop testing.

To the greatest extent practicable, all conditions of operation within the reactor coolant pump are duplicated.

Reactor coolant pump flywheel inspections and testing are described in Subsection 5.4.1.1.

5.4.1.4.1 Reactant Coolant System Flow Rate Verification

Initial verification of the RCS flow rate is made during the plant startup tests. RCS flow rates are measured during the pre- and post-core hot functional tests, and during the power ascension tests. The objective of these tests is to verify that the RCS flow rate meets the flow rate range in Subsection 3.4.1 of the Technical Specifications (Chapter 16).

5.4.2 Steam Generators

The two steam generators are designed to transfer 4,000 MWt from the RCS to the secondary system, producing approximately 8.141×10^6 kg/hr (17.95×10^6 lb/hr) of 70.30 kg/cm^{2}A (1,000 psia) saturated steam, when provided with 232.2 °C (450 °F) feedwater. Moisture separators and steam dryers in the shellside of the steam generator limit the moisture content of the steam to maximum 0.25 wt% during normal operation at full power. The steam generator design parameters are listed in Table 5.4.2-1. The steam generators, including the tubes, are designed to withstand the consequences of the design transients of Table 3.9-1 so that the ASME Code allowable stress limits are not exceeded for the specified number of cycles. All transients have been established based on conservative assumptions of operating conditions in consideration of supporting system design capabilities.

The steam generator is designed to provide reasonable assurance that critical vibration frequencies are well out of the range that are expected during normal operation and during abnormal conditions. The tubing and tubing supports are designed and fabricated with considerations given to both secondary side flow induced vibrations and reactor coolant pump induced vibrations. In addition, the steam generator assemblies are designed to withstand the blowdown forces resulting from a steam line break and feedwater line break. The two accidents are not considered simultaneously.

Vapor bubbles are generated if the static pressure in a flowing liquid is dropped below the saturated pressure corresponding to the liquid temperature. The region of the flow where bubbles exist is the cavitating region, whereas the observed damage is at the location of the bubble collapse. In the APR1400 steam generator, a pressure drop inside the tubes is not as significant as a cavitation, which can occur because the operating pressure of 158.19 kg/cm^{2}A (2,250 psia) inside the tubes is much higher than the saturation pressure of 121.13 kg/cm^{2}A (1,723 psia) of the maximum coolant temperature of 323.8 °C (615 °F) in the steam generator, and the flow paths of tubes from the tube inlet to the tube outlet are smooth enough not to cause a sudden drop of velocity of flow. The secondary side flow path outside-tubes is also smooth on the secondary side so that cavitation does not occur. No cavitation has been reported to date in steam generators that are identical to the APR1400 steam generators.

The steam generator tube material is thermally treated NiCrFe Alloy 690 (ASME SB-163). The outside diameter is 19.05 mm (0.75 in) with 1.0668 mm (0.042 in) nominal wall thickness. An analysis is performed to establish the maximum allowable tube wall degradation for the steam generator tubes in accordance with the requirements of NRC RG 1.121 (Reference 8). Load conditions considered are maximum tube differential pressures during normal operation and faulted load conditions. The margin of safety against tube rupture under normal operating condition is not less than 3.0, and the margin of safety against tube failure under postulated accidents, such as a loss of coolant accident, main steam line break, or feedwater line break concurrent with an SSE, are consistent with the margin of safety determined by the stress limits specified in the ASME Code.

The more probable modes of tube failure, which result in smaller break areas, are the results from involving the occurrence of pinholes or small cracks in the tubes and of cracks in the seal welds between the tubes and tubesheet. Detection and control of steam generator tube leakage are described in Subsection 5.2.5.

The concentration of radioactivity in the secondary side of the steam generators is dependent on the concentration of radionuclides in the reactor coolant, the primary-to-secondary leak rate, and the rate of steam generator blowdown. The specific activities that are expected in the secondary side of the steam generators during normal operation are given in Section 11.1.

The recirculation water within the steam generators contains volatile additives necessary for proper chemistry control. These and other chemistry considerations for the steam generators are discussed in Subsection 10.3.5.

5.4.2.1 Steam Generator Materials

5.4.2.1.1 Selection, Processing, Testing, and Inspection of Materials

The pressure boundary materials used in the construction of the steam generators are listed in Table 5.2-2. These materials are in accordance with ASME Section III. The Code Cases used in the fabrication of the steam generators are described in Subsection 5.2.1.

The corrosion-resistant cladding (NiCrFe alloy or austenitic stainless steel) on the tubesheet and on other primary side components is weld-deposited, fabricated, and inspected according to the requirements in Section III NB and Part QW of Section IX of the ASME Code.

5.4.2.1.2 Steam Generator Design

5.4.2.1.2.1 Design Description

The steam generator is illustrated in Figure 5.4.2-1. Moisture-separating equipment in the shell side of the steam generators limits moisture content of the exit steam. Manways and handholes are provided for access to the steam generator internals. Reactor coolant enters at the bottom of each steam generator through a single inlet nozzle, flows through the U-tubes, and leaves through two outlet nozzles. A vertical divider plate separates the inlet and outlet plenums in the lower head.

The steam generator with integral economizer (Figures 5.4.2-2 and 5.4.2-3) is in most respects similar to earlier U-tube recirculating steam generators. The basic difference is that instead of introducing feedwater only through a sparger ring to mix with the recirculating water flow in the downcomer channel, feedwater is also introduced into a separate, but integral section of the steam generator. A semi-cylindrical section of the tube bundle, at the cold leg or exit end of the U-tubes, is separated from the remainder of the tube bundle by vertical divider plates. Feedwater is introduced directly into this section and pre-heated before discharge into the evaporator section.

The economizer section is designed in consideration of operating transients, startup and standby operation, and accident conditions such as loss of feedwater flow and feedwater line break. The structural design of the various parts is adequate to withstand the thermal and pressure loadings from these various conditions, consistent with the appropriate load classifications and design rules in ASME Section III, Appendix G.

The components of the steam generator economizer section are designed for the primary stresses that occur due to the blowdown associated with a feedline break. The divider plates, which separate the economizer region from the evaporator region of the secondary side, are supported from the vessel shell and the central cylindrical support welded to the tubesheet. This divider cylinder becomes an extension of the primary tubesheet stay cylinder, though less massive, and extends the full height of the economizer. The tube support/flow baffle plates are supported from the vessel shell, the divider cylinder, and the tubesheet via an array of support rods. The support rods, which also serve as support plate spacers, are solid and designed for either tensile or buckling loads. Thin plates are not used because of the potential for collapse when subjected to differential pressure.

The lower portion of the evaporator section and the downcomer channel occupy only half of the steam generator cross section. The effect of this non-symmetry is considered in the calculation of the recirculation ratio, internal flow considerations, and the design of the tube support structures.

The steam-water mixture leaving the vertical U-tube heat transfer surface enters the separators, which produce a centrifugal motion in the mixture and separate the water particles from the steam. The water exits from the perforated separator housing and recirculates through the downcomer channel to repeat the cycle. Final drying of the steam is accomplished by passage of the steam through corrugated plate dryers.

The steam generators are designed to provide a margin of 10 percent of the total heat transfer surface area for the plugging.

A recirculation system allows the circulation of water through the steam generator during wet lay-up and the addition of chemical cleaning agents. The recirculation system consists of a distribution ring located above the tube bundle below the normal water level with connecting piping to the blowdown system. This piping needs a nozzle penetration through the pressure boundary (shell). Suction is taken at the blowdown nozzle and recirculated through the distribution ring. The recirculating header can effect a rapid changeover of the steam generator inventory if a chemical intrusion requiring the rapid removal of impurities occurs.

The primary head draining capability is provided by the channel head drains. Tubesheet drains allow secondary side draining. The drain capability enhances access for inspection and maintenance.

In order to enhance the steam generator integrity, the feedwater box is designed to prevent the inflow of foreign objects greater than the 6.35 mm (0.25 in) tube gap through the economizer feedwater region, which is a major path of foreign object inflow, and to remove the foreign objects during the overhaul period.

The steam generator blowdown system (Subsection 10.4.8) is designed for high-flow blowdown. The system is effective in removing particulate accumulations at the tubesheet.

The pressure drop from the steam generator economizer feedwater nozzles to the steam outlet nozzle including the economizer is designed to be less than 3.52 kg/cm^{2}D (50 psid).

The steam generator supports are described in Subsection 5.4.15.

5.4.2.1.2.1.1 Flow-Induced Vibration of the Tube Bundle

The tube support system uses the eggcrate grid and upper tube supports, both of which have proven effective in suppressing excessive tube vibration while providing superior thermal hydraulic characteristics for maximum protection against the shell side chemical “hideout” and tube corrosion. The test data provided the basis for determining acceptable shell side fluid velocities in the most critical regions of the tube bundle, the flow entrance, and the flow exit regions. Based on the test data and results from the shell side flow distribution analysis, the design of tube support spacing is shown to have conservative design margins against the onset of the fluid elastic instability (FEI). The steam generator design configuration possesses the tube vibration stability ratio below the design goal of one. Tube displacements due to the random turbulent excitation (RTE) are based on the turbulent buffeting methodology of ASME Section III, nonmandatory Appendix N (Reference 9).

Fluid elastic instability (FEI): FEI is the dominant mechanism for inducing the vibrational instability in tube bundles representative of the APR1400 design. This phenomenon occurs when sufficient flow velocity exists to put the tube into motion, which leads to the activation of a feedback mechanism. This causes an ever-increasing amplitude of vibration until (1) a balance is reached between the fluid energy that is absorbed and the energy that is dissipated through damping by the tube or (2) impacting ensues. The pertinent relationships are the critical velocity, effective velocity, and stability ration.

Random turbulent excitation (RTE): In a turbulent fluid flow environment, a tube can extract some of the energy from the fluid that has a frequency at or near the tube span natural frequency. Once the tube is put in motion, it may tend to coordinate the flow in such a way as to make it possible to absorb even higher quantities of fluid energy. ASME Section III non-mandatory Appendix N recommends an analytical methodology for determining tube response (displacements) due to RTE.

Vortex shedding: This mechanism manifests itself in a classic resonance situation when the Von-Karman vortex shedding frequency coincides with a tube span natural frequency. Theoretically when resonance exists, the magnification factor becomes infinite and even the smallest forcing function can produce vibrational failure. In reality, the forcing function introduces more energy into the system than can be dissipated through damping or a stable level of vibration will be established. However, the design of the tube and tube support for the APR1400 steam generator is not susceptible to the vortex shedding resonance based on the results of the tube vibration program implemented on the tube bundle with tightly packed tube arrays.

To evaluate the flow-induced vibration, the thermal-hydraulic conditions including interstitial fluid velocity, fluid density, void fraction, and steam quality of APR1400 steam generator secondary side are analyzed using EPRI's ATHOS3 computer program specified in Subsection 3.9.1.2.1.11. The design concept of tube support structures for the APR1400 steam generator is identical to that of the OPR1000 steam generator. This design has been used in other nuclear power plants and has proven to effectively suppress flow-induced vibrations of large-sized tube bundles. Refer to Subsection 5.4.2.1.2.2 for detailed descriptions of the tube support structures.

5.4.2.1.2.1.2 Water Hammer at Feedwater Ring

Provisions related to the design and operating procedure of economizer-type steam generators are made to prevent the water hammer. A goose-neck design and top-discharge spray nozzle of the downcomer feedwater piping inside the steam generator is included in the design to prevent the generation of steam bubbles regardless of water level. An auxiliary feedwater line is also connected to the downcomer feedwater line, and a plant startup test to confirm that there is no water hammer in the feedwater line is performed. With respect to the operating procedure, all feedwater flow to the steam generator is directed through the downcomer feedwater nozzle below 20 percent of full power, which eliminates the possibility of a condensation-induced water hammer in the economizer feedwater line.

5.4.2.1.2.1.3 Thermal Stratification at Feedwater nozzle

NRC Bulletin 79-13 (Reference 10) addresses the effect of thermal stratification that leads to cracking of the feedwater line.

The APR1400 feedwater lines are designed to minimize thermal stratification. The feedwater lines are angled downward from the horizontal to minimize the potential for thermal stratification.

Thermal stratification could occur in the horizontal sections of piping when the incoming feedwater flow rate is low and there is a large temperature difference between the incoming feedwater and the steam generator coolant, which results in a density difference. Fluctuations in the elevation of the interface between the hot and cold coolants cause thermal fatigue damage.

As shown in Figure 5.4.2-1, the upward bend using a goose-neck design is incorporated to avoid the stratified flows in the piping connecting the thermal sleeve in the downcomer feedwater nozzle to the downcomer feedwater piping inside the steam generator.

5.4.2.1.2.2 Material Design

The design of the APR1400 steam generators limits the potential for degradation so the integrity of the steam generator, including the tubes, is maintained during the operating period between inspections. Degradation of the steam generator tubes and other secondary side components that could affect tube integrity is manageable through the steam generator program (see Subsection 5.4.2.2). In addition, degradation of steam generator RCPB materials is manageable through the inservice inspection program (See Subsection 5.2.4).

Tube Supports

The three types of structures in the APR1400 steam generators that support the tubes are horizontal grid or eggcrate, vertical, and diagonal. All three types are fabricated from Stainless Steel Type 409. A design consideration for the supports is the prevention of dryout at support locations. With one exception, all tube supports in the APR1400 steam generator are constructed of flat strips that present a flat surface to the tube. The exception is the flow distribution plate just above the entrance to the economizer section of the tube bundle. At this location, secondary water is subcooled, and dryout will therefore not occur.

The eggcrates have three configurations depending on their location in the evaporator: a full circular structure, a half circular structure, and a structure bounded by the circumference and a chord. An eggcrate is composed of strips intersecting at an angle of 60 degrees and joined at the outer and inner perimeters with a pair of square bars on top and bottom. The strips alternate between a 50.8 mm (2 in) slotted type and a 25.4 mm (1 in) unslotted type; both are 2.286 mm (0.090 in) thick (refer to Figure 5.4.2-4). The eggcrates themselves are supported and spaced by tie rods located throughout the tube bundle and by the weldment to the tube bundle shroud.

The eggcrates form an open lattice and thus minimize the potential for local dryout conditions. The number of eggcrates is selected to maintain the natural frequency of the tubes that is significantly higher than the exciting frequencies induced by cross flow at the fluid entrances to the bundle. Both analysis and test results have been applied to define spacing that precludes vibration-induced damage (fretting and wear). In addition, careful attention is paid to localized flow path details where velocities may be higher than nominal. The vertical supports, shown in Figure 5.4.2-5, are assembled concurrently with tube installation and are composed of vertical, slotted 50.8 mm (2 in) strips intersecting with horizontal 12.7 mm (0.5 in) strips; both 2.286 mm (0.090 in) thick. The assembly is bounded about the periphery by either square bars or custom-shaped plates depending on the location. The vertical supports in the bend region are made of perforated strips to enhance their free-flowing nature. As shown on Figure 5.4.2-5, the vertical supports are also integrated with perforated diagonal strips, which provide thorough vibration support of the bend region without compromising the free-flowing character of the supports.

One set of eggcrate flow distribution plate is installed at the full eggcrate of steam generator to prevent the secondary-side flow from concentrating into central cavity. This design has been proven to effectively reduce localized wear since having been applied to previous nuclear power plants. The flow distribution plate is fixed onto the square-section ring in the full eggcrate with retainer ring segment and hexagon head cap screws. The cover plate is provided in the center of the flow distribution plate for manufacturing process. Lock washers and initial preloads for the cap screws are applied.

Steam Generator Tubes and Other Secondary Non-pressure Boundary Material

To limit the potential for the tubes to be dented, the tube support structures are fabricated from stainless steel Type 409 material (see Table 5.2-2). Tube denting is associated with the corrosion of tube support structures and creates a hard corrosion product that fills the crevice between the tube and the tube support. Dents can result in the restriction of primary coolant flow and stress-corrosion cracking of the tubes.

To limit the susceptibility of steam generator tubes to corrosion and to optimize the corrosion resistance of the microstructure, the APR1400 steam generator tubes are made of NiCrFe Alloy 690 that is thermally treated (TT). To reduce residual stresses in the U-bent region of short-radius (less than or equal to 279.4 mm [11 in]) U-bent tubes, the U-bent region of short-radius tubes is stress-relieved after bending. The materials that support the tubes and other materials on the secondary side, such as flow distribution plate and eggcrate flow distribution plate, are stainless steels that are sufficiently resistant to degradation to provide reasonable assurance that the tubes will remain adequately supported and to reduce the potential for the generation of loose parts, which can result in loss of tube integrity.

In addition, to prevent PWSCC, thermally treated alloy 690 (690TT) and 52/52M/152 weld metals are used for the steam generators. An additional discussion of the avoidance of stress corrosion cracking for Ni-base alloys is included in Subsection 5.2.3.5.

Tube Fastening to Tubesheet

The method of fastening tubes to the tubesheet conforms to the requirements of ASME Sections III and IX. Tube expansion into the tubesheet is such that no voids or crevices occur along the length of the tube in the tubesheet. The tube is expanded into the tubesheet by the use of hydraulic expansion. For hydraulic expansion, the expansion mandrel length is set to provide full-depth expansion, and hydraulic pressure is accurately applied, measured, reapplied, and controlled inside the tube so that the crevice between the tube and tubesheet is as small as possible. Expansion of tubes creates residual stresses in the transition zone between the expanded and the unexpanded regions of the tube. Residual stress measurements have been taken on the transition zone. The residual stress measurements verify the absence of any high residual tensile stress in the transition zone. Material specifications such as the use of TT tubing, welding procedures and fabrication procedures preclude the need for complete-bundle stress relief after assembly.

Corrosion Allowance

Carbon or low alloy steel materials, which compose the pressure boundary of secondary side, have the corrosion allowance of 1.5875 mm (1/16 in). Other secondary side materials and primary side materials are Ni-based alloys or austenitic stainless steels or clad with these materials, which have sufficiently high corrosion resistance.

Bolting Materials

Primary studs and nuts of the APR1400 steam generators are SB-637 N07718, and secondary studs and nuts are SA-540 Grade B24, or SA-193 Grade B7. These studs and nuts have performed adequately under service conditions and have not shown stress-corrosion cracking. The yield strength of ferritic fastener materials is limited to a maximum of 10,546 kg/cm^2 (150 ksi).

5.4.2.1.3 Fabrication and Processing of Ferritic Materials

Fracture Toughness

The Class 1 components of the steam generator meet the fracture toughness requirements of the ASME Code. Fracture toughness testing is described further in Subsection 5.2.3.3.

Welding

Subsection 5.2.3.3 describes the controls for welding ferritic steels. Conformance to the applicable NRC RGs for steam generators is summarized as follows:

a. NRC RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," is addressed in Subsection 5.2.3.3.

b. NRC RG 1.71, "Welder Qualification for Areas of Limited Accessibility," is addressed in Subsection 5.2.3.3.

c. NRC RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," is addressed in Subsection 5.2.3.3.

5.4.2.1.4 Fabrication and Processing of Austenitic Stainless Steel

Limiting Susceptibility to Cracking

Cold-worked austenitic stainless steel is not used for steam generator RCPB materials.

Fabrication of the steam generator is consistent with the recommendations of NRC RG 1.44, except for the criterion used to demonstrate freedom from sensitization. ASTM A 262 Practice A or E is used to demonstrate freedom from sensitization in fabricated and unstabilized stainless steel. Stabilized stainless steels are not subject to sensitization. Stress corrosion cracking of unstabilized austenitic stainless steels in the pressure boundary of the APR1400 steam generators is prevented through the following:

a. Solution heat treatment

b. Implementation of the procedures and/or practices demonstrated not to produce a sensitized structure for the fabrication of RCPB components

c. Delta ferrite control of weld filler metal, which is controlled to 8FN-15FN (8FN-16FN for Type 309 (L)) with no reading below 5FN as deposited

d. Prohibition of exposure of unstabilized austenitic Type 300 series stainless steels to temperatures ranging from 427 to 816 °C (800 to 1,500 °F)

e. Limit of carbon content of unstabilized austenitic Type 300 series stainless steels to a maximum of 0.065 percent

f. Control of welding parameters:

   1) Weld heat input to less than 23.6 kJ/cm (60 kJ/in)

   2) Interpass temperature to a maximum of 176.7 °C (350 °F)

Avoidance of stress corrosion cracking is described further in Subsection 5.2.3.4.

Requirements for cleanliness and contamination protection are included in the equipment specification for the steam generator fabricated with austenitic stainless steel. Procedures for contamination control during fabrication, shipment, and storage of the steam generator meet the requirements of NRC RG 1.28. Avoidance of contamination causing stress corrosion cracking is described further in Subsection 5.2.3.4.

Tools used in abrasive work operations on austenitic stainless steel such as grinding or wire brushing do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress corrosion cracking.

The thermal insulation that is used for the steam generator is either reflective metal insulation or nonmetallic insulation that meets the criteria of NRC RG 1.36.

<u>Welding</u>

Subsection 5.2.3 describes the controls for welding austenitic stainless steels. Conformance to the applicable NRC RGs for the steam generator is summarized as follows:

a. NRC RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” is addressed in Subsection 5.2.3.4.

b. NRC RG 1.34, “Control of Electroslag Weld Properties,” is addressed in Subsection 5.2.3.3.

c. NRC RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” is addressed in Subsection 5.2.3.3.

5.4.2.1.5 Compatibility of Materials with the Primary and Secondary Coolant and Cleanliness Control

Localized corrosion of tubing material has led to steam generator tube leakage in some operating reactor plants. Examinations of tube defects that have resulted in leakage have shown that two mechanisms are primarily responsible. Localized corrosion mechanisms are referred to as stress assisted caustic cracking and wastage or tube wall thinning. Both types of corrosion have been related to steam generators that have operated on phosphate chemistry. The caustic stress-corrosion type of failure is precluded by controlling bulk water chemistry to the specification limits shown in Subsection 10.3.5.

Operating steam generators have experienced the following corrosion degradation mechanisms: phosphate wastage, sulfate wastage, intergranular attack, secondary side stress corrosion cracking, and pitting and denting resulting from tube support corrosion. With respect to these phenomena, the most important design feature of the APR1400 steam generators is the selection of tubing and tubing support materials. NiCrFe Alloy 690 TT is specified for the APR1400 steam generator tubes. Stainless Steel Type 409 material is specified for the tube supports (see Table 5.2-2).

Volatile chemistry has been successfully used to minimize corrosion in many of the steam generators that have gone into operation since 1972. Secondary water chemistry and operating chemistry limits for secondary water and feedwater are addressed in Subsection 10.3.5. Removal of solids from the secondary side of the steam generator is addressed in Subsection 10.4.8.

The onsite cleaning and cleanliness of the steam generator are controlled according to NRC RG 1.28.

5.4.2.1.6 Provisions for Accessing the Primary and Secondary Sides of the Steam Generator

The steam generators have 533.4 mm (21 in) manways (Figure 5.4.2-1). On the primary side, there is one manway for the cold leg side and another for the hot leg side. Manway locations are optimized for use of remote manipulators for inspection and maintenance. Access for eddy current testing is through the primary-side manways.

On the secondary side, two manways are provided to allow access to the separator and dryer area. In addition, an internal hatch provides access to the top of the tube bundle. These openings allow inspection, which provides information on the condition of separation equipment, feedwater ring, and top of the tube bundle. Two 203.2 mm (8 in) handholes, at the tubesheet elevation, are included to provide access for tubesheet sludge lancing as well as for inspection of the downcomer annulus. These handholes can be used to remotely inspect for and retrieve loose parts. In order to enhance the steam generator integrity, the feedwater box is designed to limit the introduction of foreign objects greater than the 6.35 mm (0.25 in) tube gap through economizer feedwater region, which is a major path of foreign object inflow. Two 127.0 mm (5 in) inspection holes are provided to remove the foreign objects trapped in the feedwater box.

5.4.2.2 Steam Generator Program

The purpose of a steam generator program (SGP) is to maintain the structural and leakage integrity of steam generator (SG) tubes. An SGP provides effective monitoring and management of tube degradation and degradation precursors for prompt preventive and corrective actions. The SGP contains a balance of prevention, inspection, evaluation and repair, and leakage monitoring measures. The SGP is established and maintained based on the requirements of NEI 97-06 (Reference 11) and its referenced EPRI guidelines.

The SGP complies with the relevant requirements of the following NRC regulations:

a. GDC 32 of Appendix A to 10 CFR 50 requires, in part, that the designs of all components that are part of the RCPB permit periodic inspection and testing of critical areas and features to assess their structural and leak-tight integrity.

b. 10 CFR 50.55a(g) requires that inservice inspection (ISI) programs meet the applicable inspection requirements in ASME Section XI. The SGP is a portion of the ISI program. In addition, 10 CFR 50.55a(b)(2)(iii) addresses SG tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.

c. 10 CFR 50.36 applies to the SGP in the Technical Specifications.

d. Appendix B to 10 CFR 50 applies to the implementation of the SGP. Of particular note are Criteria IX, XI, and XVI. Criterion IX requires, in part, that measures be established to ensure that special processes, including nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures. Criterion XI requires, in part, the establishment of a test program to ensure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in applicable design documents. Criterion XVI requires, in part, that measures be established to ensure the prompt identification and correction of conditions that are adverse to quality.

e. 10 CFR 50.65 requires that licensees monitor the performance or condition of SSCs against goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

5.4.2.2.1 Design Description

The SGs are designed to permit access required for tube inspections, testing, plugging, and repairs. The design is described further in Subsection 5.4.2.1.6.

5.4.2.2.2 Implementation of SGP

KHNP has been operating the SGP comprehensive integration procedure, which includes the constitution and application of SGP documents, the roles and responsibilities of organizations, and also the technical program including the QA process. The SGP includes degradation assessment, inspection, integrity assessment, tube plugging and repairs, primary-to-secondary leak monitoring, maintenance of secondary-side integrity, secondary-side water chemistry, primary-side water chemistry, foreign material exclusion, contractor oversight, self-assessment, and reporting.

5.4.2.2.2.1 Degradation Assessment

A degradation assessment is performed prior to the preservice inspection (PSI) and planned ISI for SGs during commercial operation to address the RCPB integrity within the SG (e.g., plugs, sleeves, tubes, and components that support the pressure boundary such as secondary-side components).

The assessment determines the size and location of existing and potential degradations that are likely to become harmful cracks.

The assessment considers operating experience and provides reasonable assurance that appropriate inspections are performed during the upcoming outage and that the requisite information for integrity assessment is provided.

Some of the important features of the degradation assessment are:

a. Identifying existing and potential degradation mechanisms

b. Choosing techniques to test for degradation based on the probability of detection and sizing capability

c. Establishing the number of tubes to be inspected

d. Establishing the tube integrity limits for condition monitoring and operational assessment

5.4.2.2.2.2 Inspection

SG tube inspections based on degradation assessments are conducted and follow the inspection guidance in the EPRI PWR Steam Generator Examination Guidelines (Reference 12).

Some of the important features of SG tube inspections are:

a. Sampling as supported by the degradation assessment

b. Obtaining the information necessary to develop degradation, condition monitoring, and operational assessments

c. Qualifying the inspection program by determining the accuracy and defining the elements for enhancing nondestructive examination (NDE) system performance, including technique, analysis, field analysis feedback, human performance and process controls

5.4.2.2.2.3 Integrity Assessment

SG tube integrity is assessed after each SG tube inspection. The assessment includes:

a. Condition Monitoring (CM): A backward-looking assessment that confirms that adequate SG tube integrity has been maintained during the previous inspection interval.

b. Operational Assessment (OA): A forward-looking assessment that demonstrates that tube integrity performance criteria will be met throughout the next inspection interval.

5.4.2.2.2.4 Tube Plugging and Repairs

Plugging and repair methods are qualified and implemented in accordance with industry standards (i.e., ASME Section XI, IWA-4700). The qualification of the plugging and repair techniques considers the SG conditions and mockup testing. The EPRI guidance document, “Pressurized Water Reactor Steam Generator Examination Guidelines” (Reference 12), provides a pre-service inspection of the plugging or repair. The EPRI documents, “PWR Steam Generator Tube Plug Assessment” and “PWR Steam Generator Sleeving Assessment,” provide further guidance on tubing maintenance and repair.

5.4.2.2.2.5 Primary-to-Secondary Leak Monitoring

In order to prevent the leakage of primary coolant to the environment through SGs, primary-to-secondary leak monitoring procedures are established in accordance with the Technical Specifications and the EPRI "PWR Primary-to-Secondary Leak Guidelines" (Reference 13) provides further guidance. Monitoring gives operators information that is needed to safely respond to situations in which tube integrity becomes impaired and significant leakage or tube failure occurs. There are three action levels of reactor operation according to the leak rate in SGs.

5.4.2.2.2.6 Maintenance of Steam Generator Secondary-Side Integrity

Secondary-side SG components that are susceptible to degradation are monitored if their failure could prevent the SG from fulfilling its intended safety-related function. The monitoring includes design reviews, assessment of potential degradation mechanisms, industry experience for applicability, and inspections, as necessary, to provide reasonable assurance that degradation of these components does not threaten tube structural and leakage integrity or the ability of the plant to achieve and maintain safe shutdown.

The secondary-side visual inspection program defines the scope of inspection, and the inspection procedures and methodology to be used, when secondary-side visual inspections are performed. Additional guidance is provided in the EPRI "Steam Generator Integrity Assessment Guidelines" (Reference 14).

5.4.2.2.2.7 Secondary-Side Water Chemistry

Procedures are prepared for monitoring and controlling secondary-side water chemistry to inhibit secondary-side corrosion-induced degradation such as outside diameter stress corrosion cracking in accordance with the EPRI "PWR Secondary Water Chemistry Guidelines" (Reference 15).

5.4.2.2.2.8 Primary-Side Water Chemistry

Procedures are prepared for monitoring and controlling primary-side water chemistry to inhibit primary-side corrosion-induced degradation such as primary water stress corrosion cracking in accordance with the EPRI "PWR Primary Water Chemistry Guidelines" (Reference 16).

5.4.2.2.2.9 Foreign Material Exclusion

Procedures are prepared for control and monitoring of foreign objects and loose parts to prevent fretting and wear degradation of the tubing The SG program includes secondary-side visual inspections and procedures to preclude the introduction of foreign objects into either the primary or secondary side of the SG whenever it is opened for inspections, maintenance, repairs, modifications, or other reasons.

Such procedures include the following as a minimum:

a. Detailed accountability for all tools and equipment used during any activity when the primary or secondary side is open

b. Appropriate controls and accountability for foreign objects such as eyeglasses and personal dosimetry

c. Cleanliness requirements

d. Accountability for components and parts removed from the internals of major components (e.g., reassembly of cut and removed components)

The potential for introduction of loose parts or foreign objects from secondary-side systems is also considered.

5.4.2.2.2.10 Contractor Oversight

Procedures are established for oversight of contractor work for the SG inspections and cleaning work that will be performed during the refueling outage. Critical aspects of the oversight includethe following as a minimum:

a. Review and approval of the scope of work to be performed by a contractor

b. Review and approval of the degradation assessment

c. Review and approval of the contractor's examination procedures

d. Monitoring of the contractor's examination work progress

e. Review and approval of the contractor's deliverables

f. Review and approval of the tube integrity assessment (CM/OA) and associated support documents

5.4.2.2.2.11 Self-Assessment

Self-assessment of the SGP is performed by the SG expert team and independent peer review by knowledgeable personnel in the power stations on a periodic basis. The self-assessment identifies areas for program improvement along with program strengths.

5.4.2.2.2.12 Reporting

The following reports will be submitted to the NRC within 6 month after completion of an inspection performed under the SGP:

a. The scope of inspection performed on each SG

b. Active degradation mechanisms found,

c. NDE techniques utilized for each degradation mechanism

d. Location, orientation(if linear), and measured sizes (if available) of service-induced indications

e. Number of tubes plugged (or repaired) during the inspection outage for each active degradation mechanism

f. Total number and percentage of tubes plugged (or repaired) to date

g. The results of condition monitoring, including the results of tube pulls and in situ testing

h. The effective plugging percentage for all repaired tubes in each SG

i. Repair method utilized and the number of tubes repaired by each repair method

The SG tube surveillance program, including performance criteria for tube integrity, tube repair criteria, and tube inspections, is described in the Technical Specifications (Chapter 16), Subsection 5.5.9. The repair criteria are determined based on NRC RG 1.121 and the EPRI guidelines. Limiting conditions for operation and reactor coolant system operational leakage limits, including primary-to-secondary leakage limits, are described in the Technical Specifications (Chapter 16), Subsections 3.4.12 and 3.4.17.

Preservice inspection of all tubes in accordance with ASME Section XI and the EPRI PWR Steam Generator Examination Guidelines described in NEI 97-06 is performed using techniques that will also be used during subsequent inspections.

If Technical Specifications include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the Technical Specifications govern.

5.4.2.3 Tests and Inspections

Prior to, during, and after fabrication of the steam generator, nondestructive tests based on Section III of the ASME Code are performed.

Initial hydrostatic tests of the primary and secondary sides of the steam generator are conducted in accordance with ASME Section III. Leak tests are also performed. Following satisfactory performance of the hydrostatic tests, magnetic-particle inspections are made on all accessible welds.

Inservice inspections of the steam generator are performed in accordance with ASME Section XI, including automatic ultrasonic for SG transition region.

5.4.3 Reactor Coolant Piping

5.4.3.1 Design Basis

Applicable design codes are provided in Table 5.2-1. The reactor coolant loop piping is designed and analyzed for the transients specified in Table 3.9-1. In addition, those nozzles subjected to local thermal transients, caused by fluid entering the RCS from an auxiliary system, are analyzed to provide reasonable assurance that the nozzles can accommodate the transients.

In addition to being specified as seismic Category I, the piping is designed to provide reasonable assurance that critical vibration frequencies are out of the range expected during normal operation and abnormal conditions. Additional descriptions relating to seismic and dynamic analyses and criteria for the reactor coolant piping are contained in Subsections 3.7.2 and 3.9.2, respectively.

Leak-before-break (LBB) is applied to reactor coolant piping, including the surge line. The LBB evaluation procedure is described in Subsection 3.6.3.

5.4.3.2 Description

Each of the two heat transfer loops contains five sections of pipe: one 1,066.8 mm (42 in) internal diameter pipe between the reactor vessel outlet nozzle and steam generator inlet nozzle, two 762 mm (30 in) internal diameter pipes from the steam generator's two outlet nozzles to the reactor coolant pumps suction nozzle, and two 762 mm (30 in) internal diameter pipes from the pumps discharge nozzle to the reactor vessel inlet nozzles. These pipes are referred to as the hot leg, suction legs, and cold legs, respectively. The other major pieces of reactor coolant piping are the surge line, a 300 mm (12 in) pipe between the pressurizer and the hot leg, and the spray line, a 100 mm (4 in) pipe at the pressurizer end reduced to an 80 mm (3 in) pipe and connected to two cold legs.

The 1,066.8 mm (42 in) and 762 mm (30 in) pipe internal diameter are selected to obtain coolant velocities that provide a reasonable balance between erosion-corrosion, pressure drop, and system volume. The surge line is sized to limit the frictional pressure loss through it during the maximum in-surge so that the pressure differential between the pressurizer and the heat transfer loops is no more than 5 percent of the system design pressure. The spray line sizing is discussed in Subsection 5.4.10.

To reduce the amount of field welding during plant fabrication, the 1,066.8 mm (42 in) and 762 mm (30 in) pipes are supplied in major pieces, complete with shop-installed instrumentation nozzles and connecting nozzles to the auxiliary systems. Where required, the nozzles are supplied with safe ends to facilitate field welding of the connecting piping.

Flow restricting orifices (5.6 mm dia. × 25.4 mm long) (7/32 in dia. × 1 in long) are provided in the nozzles for the RCS instrumentation and sampling lines to limit flow in the event of a break downstream of a nozzle.

The analysis, loadings, and limits for the structural evaluation of the reactor coolant piping for each condition are discussed in Subsection 3.9.3.

5.4.3.3 Materials

The materials used in the fabrication of the piping are listed in Table 5.2-2. These materials are in accordance with ASME Section III. The provisions taken to control those factors that contribute to stress corrosion cracking are discussed in Subsection 5.2.3.

Fracture toughness of the reactor coolant piping is discussed in Subsection 5.2.3.

The fracture toughness properties of all ferritic reactor coolant pressure boundary (RCPB) materials are required to be in accordance with the requirements of ASME Section III, NB-2300 and Appendix G of 10 CFR Part 50. The SA 516 Grade 70 or SA-508 Grade 1 or 1a material used for reactor coolant piping is in accordance with these requirements.

Piping materials are required to meet the impact test requirements of ASME Section III, NB-2300, at a temperature of RT_{NDT} + 33.33 °C (60 °F) or less.

5.4.3.4 Tests and Inspections

Prior to, during, and after fabrication of the reactor coolant piping, nondestructive tests based upon ASME Section III are performed. In addition, the fully assembled RCS is hydrostatically tested in accordance with the ASME Code. Inservice inspection of the RCS piping is discussed in Subsection 5.2.4.

5.4.3.5 Design Features for Minimization of Contamination

The APR1400 is designed with specific features to meet the requirements of 10 CFR 20.1406 and Regulatory Guide 4.21. The basic principles of RG 4.21, and the methods of control suggested in the regulations, are specifically delineated in four design objectives and two operational objectives discussed in Subsection 12.4.2 of this DCD. The following evaluation summarizes the primary features to address the design and operational objectives for the RCS.

The RCS SSCs, including the facility that houses the components, are designed to limit leakage and/or control the spread of contamination. In accordance with RG 4.21, the RCS has been evaluated for leak identification from the SSCs that contain radioactive or potentially radioactive materials, the areas and pathways where probable leaks may occur, and methods of control incorporated in the design of the system. The leak identification evaluation indicated that the RCS is designed to facilitate early leak detection, provide capability for prompt assessment and evaluation of the responses, and sufficient spaces for the mitigation of the leak areas. Thus, unintended contamination of the facility and the environment is minimized or prevented by the SSC design, supplemented by operational procedures and programs for inspection and maintenance activities.

Prevention/Minimization of Unintended Contamination

a. The RCS is designed in accordance with ASME Section III and proven nuclear power industry experience. The RCS is designed with and overpressure protection system and other engineered safety features (ESF) to provide reasonable assurance of safe operation and mitigation of accident conditions, thus preventing unintended contamination.

b. The RCS components are designed to provide the first barrier to contain the radioactivity resulting from power operation and damaged fuel. These components are designed with proven materials and fabrication techniques, and are housed inside the reactor containment structure, which provides the second barrier. The basemat of the reactor containment structure is equipped with steel liners to prevent and minimize the spread of contamination from leaked reactor coolant. The in-core instrumentation (ICI) sump is provided at the basemat level to collect leaks, and the reactor containment sump is provided at the reactor vessel operating floor to collect leaks from the other components. The reactor floor is also lined with a steel layer to prevent contamination of the concrete structure.

c. The system, including wetted parts of the reactor vessel, steam generators, pressurizer, reactor coolant pumps, and associated piping, is designed with stainless steel cladding or base material, and has welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment.

d. The components are designed with material resistant to primary water stress corrosion cracking and reduced SG hot leg temperature to minimize the potential for stress corrosion and cracking.

e. The containment also includes the IRWST and a holdup volume tank with sufficient capacity to provide temporary storage of reactor coolant from anticipated operational occurrences and design basis accidents.

Adequate and Early Leak Detection

a. The RCS components are designed with multiple level, pressure, temperature, and radiation detection instruments to provide reasonable assurance of safe operation of the SSCs, including the associated piping, and provide alarms to the operating personnel in the event of off-normal conditions. Releases from overpressurization and leaks are collected in the reactor drain tank (RDT), the ICI sump, the containment sump, and the IRWST. A detailed description of the early leak detection instruments, locations, leak collections and pathways, and operation notifications for response actions is presented in Subsection 5.2.5.

b. Leak detection methods are segregated into two classifications: “unidentified leakages” and “Identified leakages”. Unidentified leakages are monitored by containment sump level, area particulate radiation detection, and containment atmosphere humidity instruments. Leakages are collected in sumps and a computer program is developed to estimate leakage rates based on flow measurements. This approach can detect 0.5 gpm leakage within 1 hour. Additionally, 19 locations within the containment are equipped with acoustic leak detection instruments where leakage is considered critical. See Subsection 5.2.5.1.1 for details.

i. Containment area radiation monitors, temperature and pressure monitors, sump levels and sump pump flow rates are programmed to provide leakage information and notification in the MCR for operator actions.

ii. Acoustic leak monitoring instruments are installed in other critical areas within the reactor coolant pressure boundary, including in-core instrumentation nozzles at reactor lower head, SG manway seals, reactor hot and cold leg welding areas, reactor coolant pump shaft seal housings, pressurizer lower head area, and CEA nozzle areas near the reactor head. These leak monitoring instruments provide early detection of leak and alarm signals for operator actions when required.

c. The “identified leakages” for the primary RCS leak detection methods and locations, and other indications of reactor coolant leakage are summarized in Subsections 5.2.5.1.2. The leakage sources and instruments for early detection of reactor coolant leakage are as follows:

i. Steam generator leaks are monitored by N-16 monitors on each of the main steam lines. An increase in radioactivity, as indicated by the condenser vacuum vent effluent monitor and SG blowdown monitor, will reveal reactor coolant leakage through SG tubes to the secondary side. Routine analysis of the SG secondary side will also indicate leakage of reactor coolant into the secondary system.

ii. The RCPs are designed with two mechanical seals and one mechanical vapor seal made of renewable silicon carbide rings. Each of these three seals can withstand full operating pressure. Any leakage from these seals is collected and routed to the RDT. An acoustic sensor is provided with each pump to detect any leak through the seal.

iii. Temperatures downstream of the pilot-operated safety relief valves (POSRVs) are continuously monitored for early detection of leaks.

iv. Leakage between the two metal O-rings that seal the reactor vessel head flange is routed through a leakoff line to the RDT. A normally closed, remotely activated isolation valve and a pressure indicator are installed in the leakoff line. Leakage from the reactor vessel head flange causes pressure in the leakoff line to rise. The pressure in the leakoff line is continuously monitored to detect the presence of a leak. Any leakage is bled off to the RDT by opening the isolation valve.

d. Other primary coolant leak detection methods and locations are discussed in Subsection 5.2.5.2.2.

<u>Reduction of Cross-Contamination, Decontamination, and Waste Generation</u>

a. Each RCP is equipped with an oil collection system in accordance with NRC RG 1.189 that collect oil from non-welded parts. Leaked oil is collected and drained to an oil collection tank that can store the entire inventory. This design approach prevents the potential generation of mixed waste.

b. Shop welding of major piping to larger pieces is maximized to reduce field welding. Piping safe-ends are provided for field welding connections to minimize the potential for leaking.

c. The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radioactive environment, thus minimizing waste generation.

d. The RCS is designed with continuous letdown of reactor coolant for purification in the CVCS and provision of sampling to maintain reactor coolant chemistry and quality. Similarly, the steam generators are designed with continuous blowdown of dissolved and suspended solids on the secondary side for treatment and with provisions for sampling and analysis. This design approach minimizes the buildup of radioactivity and impurities, enhances safe operation, and reduces decontamination and waste generation during decommissioning.

e. The RCS components are made of steel with low cobalt content as much as is practicable. The approach minimizes waste generation.

f. Manual valves 2 inches and larger are designed with a double-packed stem and a lantern ring with an intermediate leak drain connection. The drains are routed to the drain system for collection and treatment. This design approach minimizes the spread of contamination and the contamination of the facility.

g. Sampling lines from the RCS are designed to be as short and straight as possible to minimize traps and pockets. Flow velocity is maintained in the turbulent region to prevent settling of suspended solids. Gaseous sampling lines are sloped upward to facilitate draining of condensate. This design approach minimizes accumulation of condensate and provides more accurate samples, thus minimizing waste generation. Sampling lines are purged with adequate demineralized water or nitrogen gas in order to maintain quality of sampling.

h. The process piping containing primary coolant is properly designed to be the shortest between components, sized for turbulent flow, and laid out to minimize traps and erosion. This approach facilitates easier flow and with sufficient velocities to prevent settling of solids, thus reducing the decontamination needs and waste generation.

i. Utility connections are designed with a minimum of two barriers to prevent cross-contamination.

<u>Decommissioning Planning</u>

a. The top surface of the containment basemat is sloped and lined with a layer of steel to prevent contamination of the basemat by facilitating draining of leakage to the ICI sump and the containment sump. This layered approach facilitates easier decommissioning and minimizes waste generation.

b. The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal.

c. The RCS is designed with no embedded or buried piping, thus facilitating decommissioning planning and preventing unintended contamination.

Operations and Documentation

a. The RCS is designed for remote and automatic operation with operator supervision. Adequate instrumentation, including level, flow, temperature, and pressure elements, radiation monitors, and acoustic leak detection instruments, is provided to monitor and control the operations to prevent undue interruption, and minimize the spread of contamination and waste generation.

b. The RCS is designed for a minimal amount of maintenance. The layout takes into account the necessary access, laydown areas, handling and lifting cranes, and the auxiliary equipment in the event that maintenance is required.

c. The RCS components are equipped with loose parts monitoring instruments and vibration monitoring instruments at specific locations. This design approach enhances equipment life and minimizes waste generation.

d. Areas adjacent to the RCS pressure boundary are designed to have adequate clearance to access for inservice inspections. Personnel access is provided for component maintenance and inspections.

e. The COL applicant is to prepare operational procedures and maintenance programs for leak detection and contamination control of the RCS (COL 5.4(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.

f. Complete documentation of design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item (COL 5.4(2)).

Site Radiological Environmental Monitoring

The RCS is located inside the reactor containment building. Because of its location and associated safety design features, the potential for environmental contamination of soil and groundwater from reactor coolant is minimal. Therefore, RCS inclusion in the site radiological environmental monitoring program is not required. However, a site radiological environmental monitoring program is included for the whole plant for detection radiological of contamination.

5.4.4 [Reserved]

5.4.5 [Reserved]

5.4.6 Reactor Core Isolation Cooling System (Boiling Water Reactors Only)

This section is not applicable to the APR1400.

5.4.7 Shutdown Cooling System

5.4.7.1 Design Bases

5.4.7.1.1 Summary Description

The shutdown cooling system (SCS) is a safety-related system that is used to reduce the temperature of the reactor coolant system (RCS) in post shutdown periods from the hot shutdown operating temperature to the refueling temperature. The initial phase of a cooldown is accomplished by heat rejection from the steam generators (SGs) to the condenser or atmosphere. After the reactor coolant temperature and pressure have been reduced to approximately 176.7 °C (350 °F) and 31.6 kg/cm^{2}A (450 psia), the SCS is put into operation for normal shutdown cooling to reduce the RCS temperature to the refueling temperature, and to maintain this temperature during refueling.

Additionally, the SCS is used in conjunction with the atmospheric dump valves (ADVs) and the auxiliary feedwater system to cooldown the RCS following a small break loss of coolant accident (LOCA) (refer to Section 6.3). The SCS is also used subsequent to steam and feedwater line breaks, steam generator tube ruptures, and is used during plant startup prior to RCP restart to maintain flow through the core. After an accident, the SCS can be put into operation when the RCS temperature and pressure are below approximately 193.3 °C (380 °F) and 28.1 kg/cm^{2}A (400 psia), respectively.

5.4.7.1.2 Functional Design Bases

The following functional design bases apply to the SCS:

a. The SCS, SCS piping and its support, and components including instrumentation and controls are designed to meet GDC 2, 4, 5, 19, and 34 and BTP 5-4.

b. The SCS is designed to remove decay heat, RCS sensible heat, and heat generated by the shutdown cooling pumps (SCPs) during normal plant cooldown after partial cooldown has been accomplished, and during safe cold shutdown condition.

c. Two independent trains of the SCS are provided, each with its own suction and discharge connections to the RCS.

d. No single active failure prevents at least one complete train of the SCS from being brought on line from the main control room (MCR), whether this is during normal plant cooldown or following a design basis event (DBE).

e. The design bases defined in Subsection 5.4.7.1.1 are met assuming the failure of a single active component during shutdown cooling or a single active or limited leakage passive failure of a component during long-term operations following a DBE. Redundant components and instrumentations are used to provide reasonable assurance of the availability of the SCS. Limited leakage passive failure is defined based on maximum flow through a failed valve packing or pump (e.g., SCP mechanical seal).

f. The SCS is designed so that the SCPs and containment spray pumps (CSPs) are identical and functionally interchangeable. Provisions are made to control the status of those valves used in the SCS/containment spray system (CSS) interconnection. When used in a containment spray configuration, the SCPs are capable of being automatically started by a safety injection actuation signal (SIAS) or containment spray actuation signal (CSAS).

g. The SCS is designed for a pressure of 63.3 kg/cm^{2}G (900 psig) and a temperature of 204.4 °C (400 °F) to address NRC intersystem loss of coolant accident (ISLOCA) issues.

h. No single failure allows the SCS to be overpressurized by the RCS. The SCS components whose design pressure is less than the RCS design pressure are provided with overpressure protection (refer to Subsection 5.4.7.2.3).

i. The SCS reduces the RCS temperature as follows:

   1) Two train cooldown (normal operation): With no component failures assuming two trains are in service, the SCS is designed to be capable of reducing reactor coolant temperature as follows:

      a) To 60 °C (140 °F) within 24 hours after reactor shutdown

      b) To 54.4 °C (130 °F) by the time reactor vessel head stud detensioning operations are started (within approximately 40 hours after reactor shutdown)

      c) To 48.9 °C (120 °F) within 96 hours after reactor shutdown

      Achievement of the specified temperatures within the prescribed time is accomplished assuming that shutdown cooling is initiated at 3.5 hours after reactor shutdown.

   2) One train cooldown (safety shutdown operation): Assuming one train is out of service, the SCS is designed to be capable of reducing reactor coolant temperature as follows:

a) To 93.3 °C (200 °F) within 24 hours after reactor shutdown in conjunction with other heat removal systems (e.g., steam generator ADVs, auxiliary feedwater system)

This meets BTP 5-4 (Reference 19) criteria to be in cold shutdown within a reasonable time, typically defined as 36 hours. Achievement of the specified temperature within the prescribed time is accomplished assuming shutdown cooling initiated at 14 hours after reactor shutdown.

Typical cooldown curves are shown in Figures 5.4.7-1 and 5.4.7-2.

j. The SCS components are designed in accordance with the applicable codes and classifications discussed in Subsection 5.4.7.2.4.

k. Materials are selected to preclude system performance degradation due to the effects of short and long-term corrosion.

l. The SDCHXs are sized to remove decay heat 96 hours after shutdown based upon an average refueling water temperature of 48.9 °C (120 °F) and a component cooling water temperature of 35 °C (95 °F) with an average reactor core burnup of two fuel cycles.

m. The SCS is designed so that the SCPs can be tested at design flow conditions with the reactor operating at power.

n. The SCS is designed to transfer RCS fluid to the CVCS for purification during SCS operation.

o. The SCS is designed to transfer refueling pool water back to the IRWST following refueling operations.

p. The SCS is designed to provide cooling of the IRWST during post-accident feed and bleed operations utilizing the SIS and the safety depressurization and vent system.

q. The SCS is protected by relief valves with sufficient capacity to prevent overpressure in isolated sections and is designed to provide for RCS overpressure protection by the event initiating the pressure from either an operator or equipment malfunction during a low-temperature condition. Discharges from relief valves are collected in the EDT, RDT, or IRWST.

r. The SCS is designed to be operated with reduced RCS inventory including mid-loop condition.

s. The SCS piping and its support are designed to withstand loads arising from the various operating and design conditions. All SCS piping is provided with protection against pipe whip and missiles. Components of the SCS, including associated instrumentation and controls, are designed to perform their designed safety function under the environmental conditions of Section 3.11. The SCS is designed to accommodate seismic loads so that in the event of an SSE, the SCS components will remain functional and are protected natural phenomena such as floods and hurricanes. Fire protection is provided for components and instruments of the SCS. Piping of the SCS connected to the RCS that could be subjected to temperature distributions that would result in unacceptable thermal stress is described in Subsection 3.12.5.9.

t. The control room is provided for shutdown cooling operation under both normal and accident conditions. Adequate SCS instrumentation and controls are provided at appropriate locations outside the MCR to permit prompt shutdown to hot shutdown with a potential capability for subsequent cold shutdown of the reactor using suitable procedures.

u. The SCS is provided with appropriate isolation from the RCS when the RCS is at high pressure. The interlocks associated with six valves on the two SCS suction lines are provided to prevent the valves from opening in the event RCS pressure exceeds the SCS operating pressure.

v. The pumps are provided with minimum flow protection (recirculation lines) to prevent damage when starting against an isolated discharge pathway.

w. All SCS and subsystems connected to the RCS that extend outside the containment boundary are designed to the extent practicable to an ultimate rupture strength at least equal to normal RCS operating pressure. For those interfacing systems or subsystems that do not meet the full RCS ultimate rupture strength requirement, the degree and quality of isolation or reduced severity of the potential pressure challenges is sufficient to preclude an intersystem LOCA.

x. The SCS is designed for a single power unit only that is used only for the SCS. Each train of the SCS is isolated physically and electrically from the other to provide reasonable assurance that a failure in one train, or the effects thereof, will not result in the failure of the other train. The SCS is capable of being powered from both the plant's normal and emergency electrical power sources. In addition, the SCS components and valves are arranged and installed to prevent flooding.

y. The SCS is provided with provisions for a leakage detection and control program to minimize the leakage from those portions of the system outside of the containment that contain or may contain radioactive material following an accident.

z. For external reactor vessel cooling (ERVC) under hypothetical core melting severe accident conditions, one SCP is used for initial ERVC injection.

5.4.7.2 System Design

5.4.7.2.1 System Schematic

The flow diagrams of SCS are shown in Figures 5.4.7-3 and 6.3.2-1, respectively. The SCS contains two heat exchangers, two pumps, and two pump miniflow heat exchangers. One SCP is capable of meeting the safety shutdown cooldown criteria specified above; two SCPs are needed to meet the normal cooldown design criteria. SCS detailed design parameters are given in Table 5.4.7-1. In addition, the mode diagram is shown in Figure 5.4.7-5.

During initial shutdown cooling, a portion of the reactor coolant flows out of the SCS nozzles located on the reactor vessel outlet pipes (hot legs), and is circulated through the

SDCHXs by the SCPs. The return to the RCS is through SIS direct vessel injection (DVI) nozzles. The pressure and temperature of the RCS varies from 31.6 kg/cm^{2}A (450 psia) and 176.7 °C (350 °F) at the initiation of shutdown cooling to atmospheric pressure and 48.9 °C (120 °F) at refueling conditions. The SCS suction side pressure and temperature follow RCS conditions. The discharge side pressure is higher by an amount equal to the pump head. The operation temperature at the SDCHX outlet is lower than the RCS temperature.

Shutdown cooling flow is measured by orifice flow meters installed in each train of the SCP discharge piping. The information provided is used by the operator for flow control during the SCS operation. The cooldown rate is controlled by adjusting flow through the heat exchangers with throttle valves that are located in the outlet piping of each heat exchanger. The operator maintains a constant total SCS flow to the core by adjusting the heat exchanger bypass flow to compensate for changes in flow through the heat exchangers.

5.4.7.2.2 Component Description

a. Shutdown cooling heat exchangers

   The design temperature is based upon the temperature of the reactor coolant at the initiation of shutdown cooling plus a design tolerance. The SDCHXs are used to remove core decay, RCS sensible heat, and SCP heat during normal plant cooldown after a partial cooldown has been accomplished and during safe cold shutdown conditions. The SDCHXs are designed to maintain an average refueling water temperature of 48.9 °C (120 °F), with a component cooling water temperature of 35 °C (95 °F) at 96 hours after shutdown following an assumed reactor core average burnup of two fuel cycles. A fouling resistance is assumed for additional margin for heat exchanger performance.

b. Instrumentation

   The instrumentation and controls for the SCS are designed in accordance with the applicable portions of the IEEE Standards, as described in Subsection 7.1.2.

The operation of the SCS is controlled and monitored using dedicated redundant instrumentation that provides the capability to monitor the cooldown rate and shutdown cooling flow to detect degradation of flow or SCS heat removal capabilities. The instrumentation provided for monitoring SCS components during normal SCS operation is discussed below, and is also summarized in Section 7.5.

1) Shutdown cooling heat exchanger inlet and return line temperature

   The temperature of each shutdown cooling heat exchanger inlet and return line (combined HX and HX bypass flow) is indicated in the MCR and at a remote shutdown room (RSR). Recording capabilities for the heat exchanger inlet temperature and return line temperature are also provided in the MCR for each SCS train. These indications are used to provide a measurement of system performance, and provide information allowing the operator to adjust the cooldown rate.

2) Shutdown cooling heat exchanger outlet temperature

   The temperature at the outlet of each heat exchanger is indicated in the MCR. This instrument functions to monitor heat exchanger performance by directly measuring the outlet temperature.

3) Shutdown cooling pump suction and discharge pressure

   The pump suction and discharge pressures are indicated in the MCR and function to monitor pump performance. Low pressure alarms are provided for SCP discharge in the MCR.

4) Shutdown cooling flow

   A shutdown cooling flow indicator in each train of the SCS measures shutdown cooling flow, and indicates the flow rate in the MCR and at an RSR. A low flow alarm is provided in the MCR. The alarm alerts the operator to a low flow condition that may lead to a loss of shutdown cooling due to either a loss of adequate pump suction or the closure of a system valve.

5) SCS valve position indication

   a) Open/closed position indication is provided in the MCR for SCP local manual isolation valves SI-106, SI-107, SI-578, and SI-579. An alarm is provided to alert the operator when a valve is not fully open.

   b) Open/closed position indication is provided in the MCR and at an RSR for IRWST recirculation line isolation valves SI-300, SI-301, SI-688, and SI-693.

   c) Open/closed and full range 0 to 100 percent position indication is provided in the MCR and an RSR for SDCHX and heat exchanger bypass flow control valves SI-310, SI-311, SI-312, and SI-313.

   d) Open/closed and full range 0 to 100 percent position indication is provided in the MCR and at an RSR for IRWST recirculation line flow control valves SI-314 and SI-315.

   e) Open/closed position indication is provided in the MCR and an RSR for SCS/CSS pump suction cross connect valves SI-340 and SI-342 and is provided in the MCR for SCS/CSS pump discharge cross connect valves SI-341 and SI-343.

   f) Open/closed and full range 0 to 100 percent position indication is provided in the MCR and at an RSR for SCS DVI line isolation/flow control valves SI-600 and SI-601.

   g) Open/closed position indication is provided in the MCR and at an RSR for SCS suction line isolation valves SI-651, SI-652, SI-653, SI-654, SI-655, and SI-656. Valve position alarms are described in Subsection 5.4.7.2.3.

h) Open/closed and full range 0 to 100 percent position indication is provided in the MCR and at an RSR for SCS warm-up line isolation/flow control valves SI-690 and SI-691.

c. Piping

All SCS piping is austenitic stainless steel. All piping joints and connections are welded, except for a minimum number of flanged connections that are used to facilitate equipment maintenance or accommodate component design.

The design features to prevent air entrainment during mid-loop operation are described in Subsection 5.4.7.2.6. The SCS piping is designed to accommodate venting the SCPs to the RCS, if necessary, after the pumps have been stopped. High points in the SCS piping are minimized. Vent valves and the pipe arrangement design also provide system venting for reasonable assurance that maintenance can be performed on each component.

d. Valves

The location of valves, along with their type, type of actuator, position during the normal operating mode of the plant, type of position indication, and failure position is shown in Figure 6.3.2-1.

1) Relief valves

Protection against the overpressurization of components within the SCS is provided by conservatively designing the system piping, appropriate valving between high pressure sources and lower pressure piping, and by relief valves. The pressure of the SCS suction lines, up to and including SI-653 and SI-654, is designed to be equal to the RCS design pressure. In addition, the pressure of the SCS discharge lines, up to and including SI-168 and SI-178 from DVI nozzles, is designed to be equal to the RCS design pressure. Relief valves are provided as required by applicable codes. All relief valves are of the totally enclosed, pressure tight type, with suitable provisions for gagging.

Thermal relief valves (SI-169, SI-187, SI-188, SI-287, SI-289, SI-422, SI-423, SI-450, SI-461, SI-462, SI-466, and SI-467) are provided in the system to prevent overpressurization due to thermal transients in isolated sections of piping. A relief valve on each of the SCS suction lines (SI-179, SI-189) is sized to have sufficient capacity to provide LTOP for the RCS due to accidental operation of the SIPs, pressurizer heaters, the charging pump, and the reactor coolant pump (RCP) while in shutdown cooling (see Table 5.2-3). These relief valves also can prevent overpressurization of the SCS.

2) Actuator-operated throttling and stop valves

The failure position of each valve on loss of actuating signal or power supply is selected to provide reasonable assurance of safe operation. System redundancy is considered when defining the failure position of any given valve. Valve position indication is provided at the main control panel. Valve control with appropriate status control on the main control panel is provided where necessary for efficient and safe plant operation. All motor operated valves consist of a manual override handwheel. All actuator operated valves have stem leakage controlled by a double packing with a lantern ring leak-off connection.

Isolation valves are provided to isolate equipment for maintenance and to align the SCS for operation (SI-310, SI-311, SI-312, SI-313 for SDCHX; SI-340, SI-341, SI-342, SI-343 for SCS/CSS cross connection; and SI-391, SI-393, SI-395 for ERVC injection). Throttle valves (SI-310, SI-312, SI-311, SI-313 for heat exchanger tube side flow control; SI-690, SI-691 for SCS warmup line flow control; SI-600, SI-601 for SCS flow control) are provided for remote control. The SCS suction isolation valves (SI-651, SI-653, SI-655, SI-652, SI-654, and SI-656) are interlocked to prevent overpressurizing the SCS. The ERVC injection line isolation valves (SI-391, SI-393) are normally power removed to prevent inadvertent ERVC injection.

3) Vent and drain valves

Vent and drain valves are provided for reasonable assurance that maintenance can be performed on each SCS component.

e. Shutdown cooling pumps

The function of the SCPs is to provide flow through the SDCHXs and reactor core for normal plant shutdown operation or for long term post-accident core cooling. During normal power operation, the SCPs are isolated from the RCS by redundant motor-operated valves in series on the suction side of the pumps and a combination of redundant check valves and motor-operated valves on the discharge side of the pumps.

The shutdown cooling and containment spray functions are evaluated to select a single pump to serve both functions. The flow available with a single SCP is sufficient to either maintain an acceptable cooldown rate (41.7 °C/hr (75 °F/hr) maximum) during shutdown cooling operation or supply the CSS. Net positive suction head (NPSH) available exceeds NPSH required for both pumps for all conditions under which the pumps will be operated.

The SCP data are provided in Table 5.4.7-1. The design temperature for the SCPs is based upon the temperature of the reactor coolant at the initiation of shutdown cooling (176.7 °C (350 °F) nominal) plus a design tolerance, resulting in a design temperature of 204.4 °C (400 °F). The characteristic curve of an SCP is provided in Figure 5.4.7-4.

The SCPs are vertical, single-stage centrifugal units equipped with mechanical seals backed up by a bushing, with a leak-off to collect the leakage past the seals. The seals are designed for operation with a pumped fluid temperature of 204.4 °C (400 °F). The pump motors are specified to have the capability of starting and accelerating the driven equipment, under load, to design point running speed within 5 seconds, based upon an initial voltage of 75 percent of the rated voltage at the motor terminals, and increasing linearly with time to 90 percent voltage within 3 seconds, and increasing to 100 percent voltage in another 2 seconds. The rated power of the SCP motor is be determined at a sufficient level to operate at any point from shut-off to runout flow conditions.

The pumps are provided with drain and flushing connections to facilitate reduction of radiation levels before maintenance. The pressure containing parts are fabricated from stainless steel; the internals are selected for compatibility with boric acid solutions. The pumps are provided with minimum flow protection (recirculation lines) to prevent damage when starting against an isolated discharge pathway.

The SCPs are designed for operation at any point on the characteristic curve from minimum bypass recirculation flow (miniflow) to runout flow. A miniflow path is provided for each pump to provide protection in the event that a pump is inadvertently operated against a closed discharge line and miniflow capacity is sufficient to protect the pump against hydraulic instability. Each miniflow path is routed from pump discharge back to pump suction and contains a heat exchanger for cooling and an orifice sized to meet pump vendor miniflow requirements. A locally operated manual valve is also provided to facilitate pump maintenance; the valve is locked open during all plant operating modes.

Runout flow for each SCP is limited when the SCS is set up during preoperational testing. A flow-limiting device, located in each SCS train downstream of the SCS/CSS discharge side cross-connection, prevents pump runout flow from exceeding the maximum operation flow to protect each SCP from possible damage.

In the SCS configuration, each SCP takes suction from an RCS hot leg and returns flow to the RCS through a DVI nozzle. In this closed recirculation loop, the RCS pressure does not affect the pump flow rate. The SCPs are not required to operate at a reduced flow, such as what may occur in a safety injection application following a small break LOCA. For post-accident long-term cooling, shutdown cooling is initiated after the RCS fluid level has stabilized and the RCS pressure and temperature have been reduced to shutdown cooling entry conditions. The SCP flow is manually controlled by the operator from the MCR. The SCP flow through each SDCHX is adjusted to control the RCS cooldown rate, and flow that bypasses each heat exchanger is adjusted to maintain total SCP flow in the range from the design point to the runout limit. The ability to maintain a constant total pump flow during shutdown cooling precludes low flow operations, described in the NRC Bulletin 88-04 (Reference 20), which may cause pump damage resulting from flow instability phenomena.

f. Shutdown cooling pump miniflow HXs

The shutdown cooling pump miniflow heat exchangers remove heat generated by running SCP during operating at miniflow (i.e., against a closed discharge path). Required shutdown cooling pump miniflow heat exchanger performance is based on removing heat to limit the temperature increase.

5.4.7.2.3 Overpressure Prevention

Overpressurization of the SCS by the RCS is prevented in the following ways:

a. The SCS suction line isolation valves (SI-651, SI-652, SI-653, and SI-654) are powered by four independent power supplies so that a fault in one power supply or valve will neither line up the RCS to either of the two SCS trains inadvertently nor prevent the initiation of shutdown cooling with at least one SCS train.

b. Relief valves SI-179 and SI-189, located on the SCS suction lines, are sized to provide LTOP of the RCS (see Subsection 5.2.2). Since the LTOP relief valve setpoint pressure is lower than the design pressure of the SCS, these valves also provide overpressure protection of the SCS. An interlock associated with the shutdown cooling suction isolation valves prevents the isolation valves from being opened at RCS for normal shutdown cooling pressures above 31.6 kg/cm^{2}A (450 psia). The interlock setpoint is calculated considering tolerances necessary to provide reasonable assurance that the pressure at the valves will not exceed the LTOP valve setpoint when the SCS is aligned to the RCS for normal shutdown cooling. The instrumentation and controls that implement the interlock are described in Section 7.6.

c. The redundant SCS suction line isolation valves inside the containment are designed for full RCS pressure with the second valve SI-653 (Train A) and SI-654 (Train B) forming the pressure boundary and safety class change. The motor operators for SI-651 and SI-652 are sized to open or close against a differential pressure of 158.1 kg/cm^{2}D (2,250 psid). This is consistent with Generic Letter 89-10 (Reference 21), which requires that safety-related motor-operated valves (MOVs) function when subjected to design basis conditions (both normal and abnormal events).

d. Alarms on SI-651, SI-652, SI-653, and SI-654 annunciate when the SCS suction line isolation valves are not fully open (with concurrent low RCS temperature). Also, if SI-651 and SI-653 or SI-652 and SI-654 valves are open, and RCS pressure increases to the maximum pressure for SCS operation, an alarm will notify the operator that a pressurization transient is occurring during low temperature conditions.

e. Relief valves are provided as discussed in Subsection 5.4.7.2.2.

f. System piping is conservatively designed and maximum utilization of welded connections is made.

g. The response of the SCS to intersystem LOCA challenges is presented in Subsection 19.2.2.5.

5.4.7.2.4 Applicable Codes and Classifications

a. The piping and valves from the RCS, up to and including SI-653 and SI-654, are designed to ASME Section III, NB.

b. The remainder of the piping, valves, and components of the SCS, with the exception of the above piping and valves from the RCS, are designed to ASME Section III, NC.

c. The component cooling water side of the SDCHX is designed to ASME Section III, ND.

d. The power operated valves are designed to applicable IEEE Standards.

e. The SCS is a seismic Category I system.

5.4.7.2.5 System Reliability Considerations

The SCS is designed to perform its design function assuming a single failure, as described in Subsection 5.4.7.1.2. To provide reasonable assurance of availability of the SCS when required, redundant components and power supplies are used. The RCS can be brought to refueling temperature utilizing one of the two redundant SCS trains. However, with the design heat load, the cooldown would be considerably longer than the specified 96-hour time period. Since the SCS is essential for a safe (cold) shutdown of the reactor, it is a seismic Category I system and designed to remain functional in the event of an SSE.

The SCS does not utilize any pneumatically operated valves. The instrumentation controls and electric equipment pertaining to the SCS is designed to applicable portions of IEEE Standards 308 and 603. In addition to normal offsite power sources, physically and electrically independent and redundant emergency power supply systems are provided to power safety-related components (refer to Chapter 8 for further information).

For long-term performance of the SCS without degradation due to corrosion, only materials compatible with the pumped fluid are used. Environmental envelopes are specified for system components to provide reasonable assurance of acceptable performance in normal and applicable accident environments (refer to Section 3.11).

A limited leakage passive failure is defined as the failure of a pump seal or valve packing, whichever is greater. The maximum leakage is expected to be from a failed SCP seal. Leakage to the pump compartment drains to the room sump. From there, it is pumped to the waste management system. The sump pumps in each room will handle expected amounts of leakage. If leakages are greater than the sump pump capacity, the affected SCS train will be isolated. In the event of a limited leakage passive failure in one train of the SCS, continued core cooling is provided by the unaffected independent SCS train. The limited leakage passive failure is identified via appropriate leak detection provisions. Makeup of the leakage is provided by manually aligning the SIS to the IRWST, or by opening the safety injection tank isolation valves. The affected SCS train can then be isolated and core cooling continued with the other train.

5.4.7.2.6 Manual Actions

a. Plant cooldown

   Plant cooldown is a series of manual operations that bring the reactor from hot shutdown to cold shutdown.

Cooldown to approximately 176.7 °C (350 °F) is accomplished by releasing steam from the secondary side of the steam generators. An inadvertent SIAS or safety injection tank (SIT) discharge is precluded during a controlled cooldown by manually decreasing the SIAS setpoint, and depressurizing and isolating the SITs.

When the RCS temperature and pressure decrease to 176.7 °C (350 °F) and the maximum pressure for SCS operation (31.6 kg/cm^{2}A (450 psia)), the SCS is used. The RCS needs to be depressureized to below the maximum pressure for SCS operation in order to clear the permissive SCS interlock (see Subsection 5.4.7.2.3).

If the SCS suction line relief valves (SI-179 and SI-189) are not aligned to the RCS before cold leg temperature is reduced to below the maximum RCS cold leg temperature requiring LTOP, an alarm will notify the operator to open the SCS suction line isolation valves (SI-651, -652, -653, and -654). The maximum temperature requiring LTOP is based upon the evaluation of applicable RCS pressure/temperature curves (see Subsection 5.2.2.2.2.2).

Shutdown cooling is initiated using the SCPs. The SCS is warmed up and then placed in operation.

A maximum rate of cooldown (not to exceed 41.7 °C/hr (75 °F/hr)) is maintained by adjusting the flow rate of reactor coolant through the SDCHXs, utilizing the SDCHX outlet flow control valves in conjunction with the SDCHX bypass flow control valves. With the shutdown cooling flow indicators, the operator maintains a total shutdown cooling flow rate by adjusting the amount of coolant, which bypasses the SDCHXs.

When the system is at first put into operation, the temperature difference for heat transfer across the SDCHX is large, and only a portion of the total flow from the SCPs is diverted through the heat exchangers. As the cooldown proceeds, the temperature differential across the heat exchanger decreases, and the flow rate through the heat exchangers is increased to maintain the maximum permissible cooldown rate.

The flow to the SDCHXs is increased periodically until full SCP flow through the heat exchangers is attained. A graph of RCS temperature versus time after shutdown for a typical normal design basis cooldown is presented in Figure 5.4.7-1.

Shutdown cooling is continued throughout the entire period of plant shutdown to maintain an average refueling water temperature of 48.9 °C (120 °F) or less. Whenever shutdown cooling is in operation, shutdown purification flow may be initiated through the CVCS.

b. Plant heatup

Plant heatup is a manual operation process that brings the RCS from cold shutdown to hot standby. The SDCHXs are bypassed to maintain flow through the core without the heat removal effect of the heat exchangers. Flow can be initiated to the heat exchangers if necessary to control the heatup rate. When the RCPs can be run and the pressure-temperature limitations for LTOP are no longer necessary, the SCPs are stopped and the system is isolated for the standby mode.

c. Abnormal operation

1) Initiation of shutdown cooling with the most limiting single failure (loss of one shutdown cooling train) can be accomplished via plant procedures using equipment in the operable train.

2) The SCPs can be used alternatively as CSPs. In addition, the CSPs can be used alternatively as SCPs.

3) The SCPs, in conjunction with the SDCHXs, can be used for IRWST cooling.

d. Design basis event operations

Following certain DBEs (feedwater line break, small break LOCA, steam line break, or loss of offsite power), shutdown cooling can be initiated with RCS hot leg conditions that exceed the normal shutdown cooling initiation temperature of

176.7 °C (350 °F). However, shutdown cooling is never initiated at conditions that exceed the design temperature of 204.4 °C (400 °F) for the SCS components.

e. Operator response to an LTOP transient alarm

During SCS operation, if suction line isolation valves SI-651 and -653 or SI-652 and -654 are open, and RCS pressure exceeds the maximum pressure for SCS operation, a LTOP transient alarm will occur in the MCR. The LTOP relief valves SI-179 and SI-189, located downstream of SI-653 and SI-654, respectively, will open if pressure at the valve increases to the relief valve set pressure of 37.3 kg/cm^2 (530 psig). The limiting events that cause a pressurization transient during low temperature conditions are evaluated in Subsection 5.2.2. During plant startup, the LTOP transient alarm will also occur if the SCS suction valves fail to isolate the SCS from the RCS as pressure is increased.

The principal operator actions taken any time the LTOP transient alarm occurs are to identify and terminate the cause of the pressurization. LTOP relief valve availability is maintained throughout the transient, as long as overpressure protection is required.

During RCS heatup, LTOP relief valve availability is required up to an RCS temperature equal to the LTOP disable temperature, as described in Subsection 5.2.2.2.2.2. The operator will isolate the SCS and LTOP relief valves from the RCS prior to exceeding an RCS temperature of 176.7 °C (350 °F). During RCS heatup between the LTOP disable temperature and 176.7 °C (350 °F) if an LTOP transient alarm occurs, the operator will close the SCS isolation valves to prevent a depletion of RCS inventory. At temperatures above the LTOP disable temperature, overpressure protection of the RCS is provided by the pressurizer POSRVs. The operator assesses SCS status and overall plant status prior to proceeding with RCS heatup.

If the operator determines that the LTOP transient alarm is due to an event at RCS temperature conditions requiring the protection via LTOP relief valve, the operator closes downstream SCS suction line isolation valves SI-655 and SI-656 to minimize the impact of the transient on the rest of the SCS. After RCS pressure is reduced below the pressure that corresponds to the relief valve blowdown pressure, the operator confirms that the LTOP relief valves have been reseated by monitoring RCS pressure and level and IRWST. If an LTOP relief valve is stuck open, the operator takes actions to provide adequate RCS makeup inventory and core cooling.

f. Operation with reduced RCS inventory

Reduced inventory including mid-loop operation is necessary for increasing the plant availability. During this operation, the RCS water level is lowered to below the reactor vessel flange. When the RCS water level abnormally decreases, air may be ingested into the shutdown cooling system with the possibility of affecting the SCS. The RCS level is maintained higher than the RCS low water level of 8.3 cm (3.28 inch) above the loop center, and a SCS flow rate of 14,385 to 15,710 L/min (3,800 to 4,150 gpm) is maintained for decay heat removal and prevention of an air ingestion.

The performance of the SCS operation is verified at the RCS mid-loop level during preoperational testing. For the air ingestion RCS cooldown rate, the shutdown cooling flow rate, SCP motor current, and SCP suction/discharge pressure are monitored.

The APR1400 design includes the features listed below to facilitate continued SCS operations during reduced RCS inventory in conformance with Generic Letter (GL) 88-17. The Shutdown Evaluation Report (Reference 22) provides an assessment of shutdown operation risk in conformance with GL 88-17.

1) Two independent instrumentation systems are provided for RCS level measurement. These instruments function to monitor the RCS level in order to preclude SCS suction line vortexing and subsequent air entrainment. Level instrument types and corresponding instrumentation ranges are optimized to encompass all reduced RCS inventory conditions.

2) Two independent thermocouples are provided to measure core exit temperature.

3) Instruments that monitor the state of the SCS performance (such as pump suction pressure, pump discharge pressure, flow rate, pump motor current, and RCS heat removal degradation) are provided. These instruments function to provide sufficient information to indicate an approaching SCS malfunction due to the formation of vortexing and subsequent air entrainment.

4) SCS suction line isolation valves are not automatically closed in the event of RCS pressurization during the shutdown cooling operation. This precludes a loss of shutdown cooling by automatic closure of the isolation valves.

Although the features described above do not describe SCS instrumentation completely, the descriptions are provided in this section because they focus on precluding SCS failures due to loss of the SCP operations.

g. Refueling pool drain operation

The SCS is used to drain refueling water from the refueling pool to the IRWST during refueling. During this operation, one SCS train transfers water while the other train removes decay heat. This is done by pumping water from the RCS through the pump full flow test line to the IRWST.

5.4.7.3 Performance Evaluation

The capability of the SCS to reduce RC temperature is evaluated using heat balance calculations between the RCS and SCS at stepped intervals following the initiation of SCS operation. The stepped decrease versus time of component cooling water (CCW) temperature though the shell side of SDCHX is conservatively assumed, based on the maximum CCW temperature of the CCWS design. The basic purpose of shutdown cooling process using the SCS is to transfer the decay of the primary loop (RCS) to the secondary loop (CCW). Once the key process parameters such as SCS initiation conditions, RCS volume, effective heat transfer area, heat transfer rate, resistance coefficient of SDCHX, and SCP miniflow HX are identified as the input parameters, the time-dependent decay heat removal performance is analyzed. At each time step a series of equations is used in an interactive process to determine the maximum amount of heat removal through the SDCHX and SCP miniflow HX.

The design condition of the SCS is taken at 96 hours after plant shutdown. At this point, the design basis is to maintain a 48.39 °C (120 °F) average refueling temperature with a CCW temperature of 35 °C (95 °F). Two SDCHXs and two SCPs are assumed to be in operation at the design flow. The SDCHX size is determined at this point because it requires the greatest heat transfer area due to the relatively small ΔT between primary fluid and component cooling water. The design input heat load at 96 hours is based on decay heat at 96 hours after reactor shutdown, assuming an average reactor core burnup of two fuel cycles. Additional energy input to the RCS from two SCPs running at design flow rate is also included with no credit taken for component energy losses to the external environment. The cooldown rate is limited to a maximum of 41.7 °C/hr (75 °F/hr) throughout the cooldown. A typical two-train cooldown curve is shown in Figure 5.4.7-1.

With the most limiting single active failure in the SCS, the RCS temperature can be brought to 93.3 °C (200 °F) within 24 hours following shutdown using one SCP and one SDCHX, assuming that the RCS pressure and temperature are reduced to SCS initiation conditions by other heat rejection means within 14 hours. A typical single-train cooldown curve is shown in Figure 5.4.7-2.

The SCS is designed using a philosophy of total physical separation of redundant trains so that the system can carry out its safety function assuming a single active failure during both normal and short-term (period of operation of up to 24 hours following an initiating event) post-accident modes and a single active or passive failure during long-term post-accident modes after event initiation. Total train separation provides reasonable assurance that a single failure in one train cannot preclude the other train from accomplishing its safety functions. A failure modes and effects analysis for the SCS is presented in Table 5.4.7-2.

Reasonable assurance of adequate sampling capability of the SCS is provided for all modes of SCS operation to verify boron concentration and fission product activity.

5.4.7.3.1 Performance Evaluation Assuming the Most Limiting Single Failure and Only Onsite Power Available

The results of a computer simulation of a natural circulation cooldown (NCC) of NSSS from normal operation to SCS entry condition are presented in this section. The simulation is in conformance with BTP 5-4 requirements. These requirements include the use of only safety-grade equipment, the concurrent loss of offsite power, and a single failure.

5.4.7.3.1.1 Natural Circulation Cooldown Sequence

The following sequence was followed to achieve a successful NCC analysis:

a. Following the reactor trip, the operator manually controls the ADVs to restore and maintain the secondary pressure to no-load hot standby conditions (77.3 kg/cm^{2}A (1,100 psia)).

b. The steam generator water level is restored, and then maintained at the normal water level by manually controlling the auxiliary feedwater flow rate within the available capacity of the auxiliary feedwater system.

c. After the 4-hour hot standby period, the RCS is depressurized by using the pressurizer vent until the RCS subcooling margin reaches the minimum limit value of 15 °C (27 °F).

d. RCS cooldown with a 27.8 °C/hr (50 °F/hr) cooldown rate is initiated. This rate is slower than the administrative maximum cooldown rate of 41.7 °C/hr (75 °F/hr) and hence conservatively increases the auxiliary feedwater usage.

e. One of the three SIS pumps is throttled (three trains of the SIS are available due to the assumed single failure of one diesel generator) for the RCS boration and inventory control.

f. The pressurizer level is maintained between 30 percent and 70 percent.

g. The RCS subcooling margin is controlled between 15 °C (27 °F) and 83.3 °C (150 °F).

h. The operator utilizes the pressurizer vent of the RCGVS to manually depressurize the RCS. This is the principal means of achieving the necessary SCS pressure entry conditions. The pressurizer vent is actuated during conditions of high RCS subcooling and terminated when the subcooling margin decreases to the minimum value or when steam void in the reactor vessel upper head (RVUH) reaches a maximum value.

i. The operator uses the RVUH gas vent of the RCGVS to reduce the volume of an existing steam void and draw colder RCS fluid into the RVUH, thus cooling the RVUH region and depressurizing the RCS.

j. The operator performs the cooldown process until the RCS subcooling margin reaches the maximum value after the steam void is collapsed by the RVUH gas vent.

5.4.7.3.1.2 Natural Circulation Cooldown Analysis Results

Immediately following the loss of offsite power with the assumed loss of power to the RCPs, flow through the core decreases as the RCPs coast down. This immediately results in a core protection calculator reactor trip on low reactor coolant pump shaft speed. Full natural circulation flow is then established in the RCS in less than 10 minutes.

Shortly after the reactor trip, the operator utilizes the ADVs to stabilize the NSSS at hot standby conditions. Pressurizer level stabilizes at approximately 40 percent. The auxiliary feedwater flow to the steam generators is manually controlled to slowly refill the steam generators without overcooling the RCS. The plant is maintained at hot standby for 4 hours consistent with the BTP 5-4 requirements.

At 4.0 hours after the initiating event (the loss of offsite power), the operator opens pressurizer gas vent valve of the RCGVS to depressurize the RCS to the point where the RCS subcooling margin decreases to 15 °C (27 °F). The use of the pressurizer gas vent valve serves to decreases the RCS pressure to slightly below the minimum shutoff head of the safety injection pumps. The operator then manually starts the safety injection pumps to provide inventory control with borated water during the subsequent RCS cooldown.

After the pressure reaches 113.6 kg/cm^{2}A (1,616 psia), which corresponds to the RCS subcooling margin of 15 °C (27 °F), the operator begins RCS cooldown with a 27.8 °C/hr (50 °F/hr) cooldown rate by increasing steam flow through ADVs.

At 6.8 hours, the RCS subcooling margin exceeds the maximum limit of 83.3 °C (150 °F), at which the operator stops cooldown to prevent RCS overcooling and opens the pressurizer gas vent valve.

As the RCS pressure decreases, the coolant in the RVUH reaches saturation condition and a steam void forms. The steam void in the RVUH continues to increase in size as long as the pressurizer gas vent valves remain open because the RVUH fluid temperature exceeds the saturation temperature and the fluid vaporizes. When the steam void in the RVUH increases to 21.2 m^3 (750 ft^3), the operator closes the pressurizer gas vent valves and then opens the RVUH gas vent valves to collapse the steam void.

At 7.5 hours, the steam void in the RVUH is reduced to the smallest measurable value, and the operator closes the RVUH gas vent valve and resumes the RCS cooldown with a cooldown rate of 27.8 °C/hr (50 °F/hr).

At 9.1 hours, the RCS subcooling is again reached at the maximum limit of 83.3 °C (150 °F), and the operator stops cooldown to relieve the RCS overcooling and opens the pressurizer gas vent valve. This avoids exceeding the hot leg subcooling limit and reduces the RCS pressure. The steam void in the RVUH rapidly grows again to 21.2 m^3 (750 ft^3) where the operator closes the pressurizer gas vent valve and re-opens the RVUH gas vent valve in order to collapse the void. After the RVUH void decreases, the operator resumes the RCS cooldown.

At 10.5 hours, the RCS pressure and temperature reach the shutdown cooling entry conditions of 31.6 kg/cm^2A (450 psia) and 176.7 °C (350 °F), respectively. The final collapsing of the steam void that still exists in the RVUH is accomplished using a combination of the RVUH gas vent and the SIS.

The amount of safety-grade auxiliary feedwater used is 1,140 m^3 (300,000 gal). This demonstrates that the NCC to the shutdown cooling system entry conditions, according to the BTP 5-4 requirements, can be performed well within the limit of auxiliary feedwater storage tanks capacity (i.e., minimum capacity of 1,514 m^3 (400,000 gal)). The NCC analysis results are provided in Figures 5.4.7-6 through 5.4.7-12.

5.4.7.3.1.3 Conclusions

The NSSS NCC analysis results demonstrate that a cooldown and depressurization to the SCS entry conditions are achievable within the BTP 5-4 requirements. The total auxiliary feedwater usage is well within the minimum available capacity. It is concluded that the NSSS can be cooled and depressurized to the SCS entry conditions in conformance with the restrictive assumptions of the BTP 5-4.

5.4.7.4 Tests and Inspections

5.4.7.4.1 Preoperational Testing

Preoperational tests are conducted to verify the proper operation of the SCS. The preoperational tests include calibration of instrumentation, verification of adequate cooling flow, and verification of the operability of all associated valves. In addition, a preoperational hot functional performance test is made on the installed SDCHXs as a part of the pre-core hot functional test programs. Refer to Chapter 14 for further details on these tests.

For the preoperational test for the SCP miniflow rate, ultrasonic flow meters are temporarily installed in the SCP miniflow heat exchanger inlet and outlet lines. The SCS also undergoes a series of preoperational hydrostatic tests conducted in accordance with ASME Section III.

Preoperational test results are used to perform analyses to confirm that the as-built SCS fulfills operability and provides a level of performance that satisfies design analyses for a safe cold shutdown.

The LTOP relief valves, SI-179 and SI-189, located on the SCS suction line are shop-tested. Relieving capacity of the valves is certified in accordance with ASME Section III, NC-7000. Valve set pressure is verified by actual shop testing with water.

An inspection of the as-built piping is conducted to verify that the SCP has no loop seals and is oriented downward or horizontal except for an upward section connecting to the pump suction flange.

5.4.7.4.1.1 Flow Testing

Each installed SCS train is tested to measure SCP developed differential pressure at a flow rate equal to or greater than the pump design flow rate. An analysis is performed to convert the measured differential pressure at the test temperature to a pump head. Tests and analyses are performed with SCS suction and return aligned to the RCS, and aligned to the IRWST.

Functional tests are performed with flow aligned to the RCS to confirm that the maximum flow rate does not exceed the limit. These tests are performed at an RCS temperature of 176.7°C (350°F).

Testing is performed to confirm that the SCP return line to the IRWST allows each SCP to be operated at a flow equal to or greater than design flow during inservice testing. Analyses are performed to convert the measured pump differential pressure to a pump head.

The available NPSH to the SCPs is determined based on as-built elevations, piping arrangements and system performance parameters measured during testing. Measured pump suction pressures and flow rates are used in an analysis to demonstrate that adequate NPSH is provided to each SCP assuming the following conditions:

a. Minimum RCS water level and maximum RCS water temperature for reduced inventory operation

b. Maximum design basis RCS water temperature for SCS operation (not at reduced inventory)

c. Maximum design basis IRWST water temperature and minimum IRWST water level for the SCS operating in the IRWST cooling mode

d. As-built pressure losses for pump inlet piping and components

The calculated minimum available NPSH meets or exceeds the NPSH required by the pump vendor.

Testing is performed to demonstrate that a containment spray pump (CSP) performs the pumping function of the designated SCP. The CSP is tested by aligning its suction to the SCP suction piping and its discharge to the SCP discharge piping. The CSP provides a flow rate through the SDCHX that is greater than or equal to the SCP design flow rate.

5.4.7.4.1.2 Heat Removal Capability

Tests are performed to measure shutdown cooling flow rates at the combined discharge of the SDCHX and heat exchanger bypass line using permanently installed instrumentation.

A test is performed on SDCHXs during pre-core hot functional testing. Shutdown cooling and component cooling water flow rates to each heat exchanger, reactor coolant temperatures at SDCHX inlet and outlet, and component cooling water temperature at SDCHX inlet are measured. Test results are used to verify that the heat removal capability of each SDCHX meets the heat removal capability required to achieve a safe cold shutdown. The performance demonstrates a heat removal capacity to remove heat from the reactor coolant and transfer heat to the component cooling water system.

5.4.7.4.2 Inservice Testing and Inspection

Inservice testing of SCPs and safety-related valves is addressed in Subsection 3.9.6.

Inservice inspection of SCS piping and components is addressed in Section 6.6 and Subsection 5.2.4.

5.4.7.4.3 Design Features for Minimization of Contamination

The APR1400 is designed with specific features to meet the requirements of 10 CFR 20.1406 and Regulatory Guide 4.21. The basic principles of NRC RG 4.21 and the methods of control suggested in the regulations are grouped into four design objectives and two operational objectives, as described in Subsection 12.4.2. The following evaluation summarizes the primary features that address the design and operational objectives for the shutdown cooling system.

The SCS has been evaluated for leakage identification from the SSCs that contain radioactive or potentially radioactive materials, the areas and pathways where probable leakage may occur, and the methods of leakage control incorporated in the design of the system. The leak identification evaluation indicated that the SCS is designed to facilitate early leak detection and the prompt assessment and response to manage collected fluids. Thus, unintended contamination to the facility and the environment is minimized and/or prevented by the SSC design, operational procedures and programs, and inspection and maintenance activities.

Prevention/Minimization of Unintended Contamination

a. The SCS components are located in individual cubicles inside the auxiliary building. The floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs and sumps. This design approach prevents unintended contamination of the facility and the environment.

b. The SCS components (pumps, heat exchangers, piping) are fabricated from stainless steel and are of welded construction, thus minimizing leakage and unintended contamination of the facility and the environment.

Adequate and Early Leak Detection

a. The SCS is used only in post shutdown periods. Any leakage is drained to the floor and is collected in the local sump, which is equipped with a liquid level switch. If leakage exceeds a predetermined liquid level within the sump, the level switch initiates an alarm in the MCR for operator action to investigate the source of leakage.

Reduction of Cross-Contamination, Decontamination, and Waste Generation

a. The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation.

b. The pumps are equipped with drains directly routed to the radioactive drain system. This design approach minimizes contamination to the facility and the environment.

c. The SCS is isolated from the RCS when the RCS is at high pressure (during normal power operation). The interlocks associated with six valves on the two SCS suction lines are provided to prevent the valves from opening in the event that RCS pressure exceeds SCS operating pressure, thus minimizing cross-contamination between the systems.

Decommissioning Planning

a. The SSCs are designed for extended service life and are fabricated as individual assemblies for easy removal.

b. The SSCs are designed with decontamination capabilities. Design features, such as welding techniques and surface finishes, are included to minimize the need for decontamination and the resultant waste generation.

c. The SCS is designed with minimal embedded piping for contaminated or potentially contaminated fluid, which minimizes the potential for unintended contamination of the environment.

Operations and Documentation

a. The COL applicant is to prepare operational procedures and maintenance programs related to leak detection and contamination control of SCS(COL 5.4(3)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.

b. The COL applicant is to maintain complete documentation of the system design, construction, design modifications, field changes, and operations of SCS (COL 5.4(4)). Documentation requirements are included a COL Information Item.

Site Radiological Environmental Monitoring

The SCS is on standby mode during normal power operation and is designed to have low levels of contamination. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the SCS is maintained, resulting in a very low level of contamination of the facility. Hence, the SCS is not required to be part of the site radiological environmental monitoring program.

5.4.8 Reactor Water Cleanup System (Boiling Water Reactors Only)

This section is not applicable to the APR1400.

5.4.9 [Reserved]

5.4.10 Pressurizer

5.4.10.1 Design Bases

The pressurizer is designed to:

a. Maintain RCS operating pressure so that the minimum pressure during operating transients is above the setpoint for the safety injection actuation signal (SIAS) and low pressure reactor trip and so that the maximum pressure is below the high pressure reactor trip setpoint.

b. Withstand the consequences of the design transients of Table 3.9-1 without failure or malfunction.

c. Provide sufficient water volume in the pressurizer to prevent uncovering the heaters as a result of a reactor trip.

d. Provide sufficient water volume to prevent pressurizer heaters from being uncovered by the outsurge following step load decreases of 10 percent starting within the range of 100 percent to 15 percent of full-rated power or a 5 percent per minute ramp decrease from 100 percent to 15 percent of full-rated power.

e. Provide sufficient steam volume to allow acceptance of the insurge resulting from any loss of load transient without liquid or two-phase flow reaching the pressurizer pilot-operated safety relief valve nozzles.

f. Minimize the total reactor coolant mass change and associated charging and letdown flow rates in order to reduce the quantity of wastes generated by load follow operations.

g. Provide sufficient pressurizer heater capacity to heat up the pressurizer, filled with water at the zero power level, at a rate that provides reasonable assurance of a pressurizer temperature (and thus pressure) that will maintain an adequate degree of subcooling of the water in the reactor coolant loop as it is heated by core decay heat and/or pump work from the reactor coolant pumps.

h. Contain a total water volume that does not adversely affect the total mass and energy released to the containment during the maximum hypothetical accident.

i. Provide reasonable assurance that, in addition to being specified as seismic Category I, the pressurizer vessel, including heaters, baffles, and supports is designed so that no damage to the equipment is caused by the frequency ranges of 19-24 cps and 118-143 cps. The lower frequency is produced by vibratory excitations associated with RCP rotating speed. The design basis for the higher frequency consists of a pressure pulse of 0.56 kg/cm^2 (8 psi), which diminishes internally within the vessel.

j. Maintain the pressurizer at normal operating pressure during hot standby conditions by taking into account the energy balance of maximum heat loss from the pressurizer and the pressurizer heater capacity. This capability is provided by redundant trains of heaters powered from off-site power and Class 1E emergency power. The Class 1E emergency power is used if off-site power is not available.

k. Maintain sufficient spray flows to keep the pressure below the reactor trip setpoint during maneuvering and load follow operations and loss of load transients.

l. Provide the adequate pressurizer size and spray capacity so the pressurizer pilot-operated safety relief valves are not actuated by overpressure events initiated by normal operation transients.

5.4.10.2 System Description

5.4.10.2.1 Pressurizer

The pressurizer, as shown in Figure 5.4.10-1, is a vertically mounted, bottom supported, cylindrical pressure vessel. Replaceable direct immersion electric heaters are vertically mounted in the bottom head. The pressurizer is furnished with nozzles for the spray, surge, and pilot-operated safety relief valves, and with pressure, temperature, and level instrumentation. A manway is provided in the top head for access for inspection of the pressurizer internals. The pressurizer surge line is connected to one of the reactor coolant hot legs and the spray lines are connected to two of the cold legs at the reactor coolant pump discharge. Heaters are supported inside the pressurizer to preclude damage from vibration and seismic loadings. Principal design parameters are listed in Table 5.4.10-1.

The pressurizer heaters are single-unit, direct immersion heaters that protrude vertically into the pressurizer through sleeves welded in the lower head. Each heater is internally restrained from high amplitude vibrations and can be individually removed for maintenance during plant shutdown.

The pressurizer and surge line are located entirely above the reactor coolant loops. The surge line is continuously rising from the hot leg nozzle to the pressurizer, thus providing reasonable assurance that the line contains no water traps. The pressurizer surge line is sized and arranged to minimize the flow resistance.

Each pressurizer spray line includes an isolation valve (RC-442, RC-443), which can be remotely shut from the MCR to prevent RCS depressurization in the event the pressurizer spray valve in the line fails to close.

The maximum allowable pressure drop through the pressurizer spray line piping only is 1.62 kg/cm^{2}D (23 psid) at a total flow rate of 28.4 L/s (450 gpm) and at a water temperature of 290.6 °C (555 °F).

The pressurizer is designed and fabricated in accordance with the ASME Code listed in Table 5.2-1. The interior surface is clad with weld deposited stainless steel or NiCrFe alloy.

5.4.10.2.2 Instrumentation

The pressurizer is equipped with nine nozzles for the pressurizer level, pressure and temperature measurements. Seven of the nine nozzles are installed on the top head of the pressurizer, and the other two are installed in the bottom head. Two temperature nozzles are installed in the lower shell portion of the pressurizer to monitor water temperature.

The bottom head level nozzles are provided with the internal nozzle extensions to minimize crud buildup in the nozzles. The temperature nozzles are compatible with the thermowells.

The instrument nozzles that require socket-welded end preparations meet the requirements of ASME B16.11.

See Section 7.7 for the instrumentation and control systems that are associated with the pressurizer pressure and water level.

5.4.10.2.3 Operation

The total volume of the pressurizer is established by consideration of the factors given in Subsection 5.4.10.1. To account for these factors and to provide adequate margin at all power levels, the water level in the pressurizer is programmed as a function of average coolant temperature as shown in Figure 5.4.10-2, in conjunction with Figure 5.4.10-3. High or low water level error signals result in the control actions shown in Figure 5.4.10-4. The pressurizer surge line is sized to accommodate the flow rates associated with the RCS expansion and contraction due to the transients specified in Subsection 3.9.1.

The pressurizer maintains RCS operating pressure and, in conjunction with the chemical and volume control system (CVCS), Subsection 9.3.4, compensates for changes in reactor coolant volume during load changes, heatup, and cooldown. During full-power operation, the pressurizer is about half full of saturated steam.

RCS pressure may be controlled automatically or manually by maintaining the temperature of the pressurizer fluid at the saturation temperature corresponding to the desired system pressure. A small continuous spray flow is maintained to the pressurizer to avoid stratification of pressurizer boron concentration and to maintain the temperature in the surge and spray lines, thereby reducing thermal shock as the spray control valves open. An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow cooling if the reactor coolant pumps are shut down.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. The average reactor coolant temperature is programmed to vary as a function of load as shown in Figure 5.4.10-3. A reduction in load is followed by a decrease in the average reactor coolant temperature to the programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level, causing the reactor system pressure to decrease. This pressure reduction is partially compensated by flashing of pressurizer water into steam. In the event of a pressure level decrease, the two CVCS letdown orifice isolation valves close, and the throttling openings of the CVCS charging control valve are automatically controlled to add coolant to the RCS and restore pressurizer water level.

When the main steam demand is increased, the average reactor coolant temperature is raised in accordance with the coolant temperature program. The expanding coolant from the reactor coolant piping hot leg enters the bottom of the pressurizer through the surge line, compressing the steam and raising RCS pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. If the pressure increase is large enough, the pressurizer spray valves open and spray coolant from the reactor coolant pump discharges (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase. The programmed pressurizer water level is a RCS average temperature dependent function. A high level error signal, produced by an in-surge, causes the charging control valve modulated to a closing direction, thus restoring the pressurizer to the programmed level. Small pressure and primary coolant volume variations are accommodated by the steam volume that absorbs flow into the pressurizer, and by the water volume that allows flow out of the pressurizer.

A number of the heaters are connected to proportional integral controllers, which adjust the heat input to account for steady-state losses and the continuous spray flow, and to maintain the desired steam pressure in the pressurizer. The remaining heaters are connected to on-off controllers. These heaters are normally de-energized but are automatically turned on by a low pressurizer pressure signal or a high level error signal. This latter feature is provided because load increases result in an in-surge of relatively cold coolant into the pressurizer, thereby decreasing the bulk water temperature. The CVCS acts to restore level, resulting in a transient pressure below normal operating pressure. To minimize the extent of this transient, the on-off controlled backup heaters are energized, contributing more heat to the water. A low-low pressurizer water level signal de-energizes all heaters to prevent heater damage before they are uncovered. The pressure control program is shown in Figure 5.4.10-5.

5.4.10.3 Design Evaluation

It is demonstrated by analysis in accordance with requirements for ASME Section III, Class 1 vessels that the pressurizer is adequate for all normal operating and transient conditions expected during the life of the facility. Following completion of fabrication, the pressurizer is subjected to the required ASME Section III hydrostatic test and post-hydrostatic test nondestructive testing.

During hot functional testing, the transient performance of the pressurizer is checked by determining its normal heat losses and maximum depressurization rate. This information is used in setting the pressure controllers. Reasonable assurance of the structural integrity of the pressurizer is further obtained from the inservice inspections performed in accordance with ASME Section XI and described in Section 5.2.

Overpressure protection of the RCS is provided by four pilot-operated safety-relief valves. See Subsection 5.4.14.

5.4.10.4 Test and Inspection

Prior to and during fabrication of the pressurizer, nondestructive testing is performed in accordance with the requirements of ASME Section III. Table 5.4.10-2 summarizes the pressurizer inspection program, which also includes tests not required by the Code. See Subsection 5.2.4 for inservice inspections of the pressurizer.

5.4.11 Pressurizer Relief Tank

The in-containment refueling water storage tank (IRWST) is used as the pressurizer relief tank. The IRWST collects and condenses the steam discharged from the pressurizer through the pilot-operated safety-relief valves (POSRVs) and the reactor coolant gas vent system (RCGVS).

The POSRVs on the pressurizer are described in Subsections 5.2.2.4.1 and 5.4.14. The RCGVS is described in Subsection 5.4.12.

5.4.11.1 Design Bases

The IRWST is an integral part of the containment building internal structure and is a reinforced concrete structure with a stainless steel liner on the surfaces that are expected to be in direct contact with borated water. The IRWST is located below El. 100 ft in the floor slab between the secondary shield wall and the containment wall.

The IRWST and the discharge piping connected to the IRWST are designed to seismic Category I requirements to remain functional in the event of an SSE and are designed to prevent adverse effects on the performance of safety-related SSCs.

The IRWST is equipped with the spargers located at the end of the discharging piping and designed to effectively condense the steam and minimize the loads on the structure.

Protection from the overpressurization of and a vacuum in the IRWST is provided by swing panels on the side walls of four vent stacks. The vent stacks are located on the concrete slab at El. 100 ft, the top of the IRWST. The swing panels are installed on three side walls of each vent stack as pressure relief devices. The swing panels are closed to minimize the release of vaporized IRWST water into the containment atmosphere during normal plant operation. The swing panels are described in Subsection 6.8.2.2.5.

The IRWST is designed to accommodate the hydrodynamic loads and the thermal effects in the IRWST due to the steam discharged from the pressurizer.

5.4.11.2 System Description

The IRWST is a component of the in-containment water storage system (IWSS), which is described in Section 6.8. The flow diagram of the IRWST is given in Figure 6.8-3. The elevation and plan views of the IRWST are shown in Figures 1.2-2 and 6.8-1, respectively.

Four pressurizer POSRVs are connected to the top of the pressurizer by separate inlet lines. There are two main discharge lines to the IRWST. The steam from two POSRVs is combined into a common discharge line. The flow diagram, showing the pressurizer POSRVs and their discharge lines, is provided as Figure 5.1.2-3.

The RCGVS piping allows the operator to direct the RCGV discharge to the IRWST. The RCGVS flow diagram is shown in Figure 5.4.12-1.

5.4.11.3 Performance Evaluation

The IRWST is designed as seismic Category I with a design temperature of 143.3 °C (290 °F). Pressure in the IRWST air space is relieved to the containment atmosphere through the vent stacks.

The evaluation of the hydrodynamic loads on the IRWST and the pool temperature of the IRWST are described in Subsections 6.8.4.3 and 6.8.4.4, respectively.

5.4.11.4 Instrumentation

The IRWST is equipped with instrumentation that indicates water level, temperature, and pressure. The instrumentation is provided in the MCR and the RSR to allow the operator to monitor the status of the IRWST. The high and low alarms for level and temperature are provided in the MCR and the RSR.

The IRWST instrumentation is described in Subsection 6.8.3.

5.4.12 Reactor Coolant System High Point Vents

The reactor coolant gas vent system (RCGVS) is used to discharge noncondensable gases and steam from the high point of the RCS for venting or pressure control during post-accident conditions.

5.4.12.1 Design Bases

The RCGVS is designed to provide a safety-grade means of remotely venting noncondensable gases from the reactor vessel closure head and the pressurizer steam space during post-accident conditions. The RCGVS is also designed to provide a safety-grade means of remotely and selectively removing steam from the pressurizer steam space and/or the reactor vessel for RCS pressure control purposes in the event that pressurizer main spray and auxiliary spray are unavailable during non-LOCA design basis events. In addition, the RCGVS is used for the noncondensable gases vent path during plant startup to fill the RCS.

The reactor vessel closure head vent portion of the RCGVS is designed to provide sufficient venting capacity to vent a steam bubble formed in the reactor vessel closure head during a natural circulation cooldown analysis, assuming a single failure. Reactor vessel closure head vent flow isolation is possible, assuming a single failure.

The pressurizer vent portion of the RCGVS is designed to provide sufficient venting capacity to reduce pressurizer pressure consistent with plant cooldown requirements, assuming a single failure. Pressurizer vent flow isolation is possible, assuming a single failure.

The RCGVS equipment and piping from the reactor vessel closure head vent up to and including the second vent valve, and from the pressurizer up to and including second vent valve are designed as seismic Category I, Class 1E and designed, fabricated, erected, tested and maintained to high quality standards in accordance with ASME Section III, Class 1 requirements.

Each active RCGV valve is designed to be powered from the normal or the emergency power source. Power connections are through two independent power divisions so that in the event of an accident, in conjunction with a loss of normal power and a single failure of one emergency DC power division, a vent path from the pressurizer and the reactor vessel head can be established. The RCGV valves are remotely operated from the main control room and remote shutdown room.

Vent areas provide for mixing of the containment air. Swing panels at the top of the IRWST allow circulation of air for adequate mixing of any combustible gases with the containment atmosphere.

Venting does not adversely affect the performance of safety-related SSCs and does not aggravate the challenge to containment or the course of an accident.

The RCGVS is designed in accordance with the quality assurance acceptance criteria provided in Chapter 17.

The RCGVS satisfies applicable requirements and industry standards, including ASME Code classifications; 10 CFR 50.34(f)(2)(vi); 10 CFR 50.44; 10 CFR 50.46; 10 CFR 50.46a; 10 CFR 50.49; 10 CFR 50.55a; GDC 1, 14, 17, 19, 30, 34, and 36; and safety classifications and environmental qualifications.

5.4.12.2 System Design

RCGVS provides a means of venting noncondensable gases and steam from the pressurizer and the reactor vessel closure head to the in-containment refueling water storage tank (IRWST). The functions are as follows:

a. RCGVS provides a safety-grade means of venting noncondensable gases and steam from the pressurizer during post-accident conditions for non-LOCA design basis events.

b. Safety-grade means to depressurize the RCS in the event that pressurizer main spray and auxiliary spray systems are unavailable.

RCGV connections to the RCS are located at the reactor vessel closure head vent pipe and at the steam sample/vent line off the pressurizer upper head. The RCGVS flow diagram is shown in Figure 5.4.12-1.

5.4.12.2.1 Reactor Vessel Closure Head Vent

Piping from the reactor vessel is routed directly to the RCGV piping. This piping allows the operator to direct the RCGV discharge, through parallel valve divisions, to the IRWST, which is designed as seismic Category I.

The solenoid-operated valves of the parallel valve divisions are controlled from the MCR or RSR. Open and closed indications of the valves are provided and monitored from the MCR or RSR. Each valve is powered by the independent Class 1E power supply, and the valves are supplied with power by alternate alternating current power during an SBO. The valves are qualified using the ANSI/IEEE Std. 344 (Reference 23) as endorsed by NRC RG.1.100 (Reference 24).

The following information is available to the operator for initiating and terminating the reactor vessel closure head venting operation during post-accident conditions:

a. For initiating system operation: reactor vessel water level

b. For terminating system operation: reactor vessel water level

If voids form in the reactor vessel closure head, the operator may open the RCGV valves on the top of the reactor vessel (RG-415, RG-414, RG-417, and RG-416) to vent steam from the reactor vessel closure head, allowing the reactor vessel to be refilled. The reactor vessel closure head vent has a capacity of 4,797 kg/hr (10,576 lb/hr) at 175.8 kg/cm^2A (2,500 psia).

The noncondensable gases accumulating in the U-tubes of the steam generators are transferred to the reactor vessel closure head by using a procedure in which one or more RCPs per loop operate for short periods to force noncondensable gases out of the U-tubes.

5.4.12.2.2 Pressurizer Vent

The RCGV piping allows the operator to direct the RCGV discharge, through parallel valve divisions, to the IRWST, which is designed as seismic Category I.

The solenoid-operated valves of the parallel valve divisions are controlled from the MCR or RSR. Open and closed indications of the valves are provided and monitored from the MCR or RSR. Each valve is powered by the independent Class 1E power supply, and the valves are supplied power by alternate alternating current power during an SBO. The valves are qualified using the ANSI/IEEE Std. 344 as endorsed by NRC RG.1.100.

The following information is available to the operator for initiating and terminating the pressurizer venting operation during post-accident conditions:

a. For initiating system operation: pressurizer pressure and cold leg temperatures

b. For terminating system operation: pressurizer pressure and cold leg temperatures

The operator may use the RCGV function to cool down and depressurize the plant in the event the pressurizer main spray and auxiliary spray systems are not operable. The operator manually opens the RCGV valves (RG-410, RG-411, RG-412, and RG-413) on the top of the pressurizer, releasing steam to the IRWST through valve RG-0419/RG-0420. This pressurizer vent has the capacity of 14,023 kg/hr (30,915 lb/hr) at the pressurizer condition of 175.8 kg/cm^{2}A (2,500 psia). The RCGV flow and the depressurization rate are controlled by valves RG-410, RG-411, RG-412, and RG-413 in the vent lines from the top of the pressurizer, and by opening and closing the RCGV valves (RG-414, RG-415, RG-416, and RG-417) from the top of the reactor vessel closure head.

The noncondensable gases accumulating in the U-tubes of the steam generators are transferred to the pressurizer by using a procedure in which one or more RCPs per loop operate for short periods to force noncondensable gases out of the U-tubes.

5.4.12.2.3 Design Features for Minimization of Contamination

The RCGVS is designed with features that meet the requirements of 10 CFR 20.1406 and NRC RG 4.21. The basic principles of NRC RG 4.21, and the methods of control suggested in the regulations, are delineated in four design objectives and two operational objectives, as described in Subsection 12.4.2.

The reactor coolant gas vent system consists of piping and valves that are located inside the containment. The RCGVS vents noncondensable gases from the pressurizer and the reactor vessel upper head and depressurizes the RCS in the event that the pressurizer main spray or auxiliary spray systems are unavailable during plant cool down. The piping directs the vented gases to the IRWST or the reactor drain tank (RDT) and is sloped to facilitate the drainage of condensation. As the system is located entirely within containment, any leakage from the system components will be collected in the RDT or IRWST inside containment. Hence, the RCGVS has low potential to contaminate other areas of the plant or the environment. This design is in conformance with the requirements of NRC RG 4.21.

Prevention/Minimization of Unintended Contamination

The RCGVS is designed to vent non-condensable gases from the pressurizer and the reactor vessel upper head and depressurizes the reactor coolant system in the event that the pressurizer main spray or auxiliary spray systems are unavailable during plant cooldown. The piping directs the vented gases to the IRWST or the RDT and is sloped to facilitate the drainage of condensation, thus minimizing leakage and unintended contamination of the facility and the environment.

Adequate and Early Leak Detection

The RCGVS is designed not to be used during normal operation, and the piping is designed to slope downward to drain to the RDT and the IRWST. The potential for leakage is very low and a leak detection system is not required.

Reduction of Cross-Contamination, Decontamination, and Waste Generation

The RCGVS piping is made of stainless steel material for life-cycle planning. The material is compatible with the chemical, physical, and radiological environment, thus minimizing waste generation.

Decommissioning Planning

The RCGVS piping is designed for the full service life of the plant and is accessible for easy removal during decommissioning.

Operations and Documentation

The RCGVS piping is used during refueling or post-accidental conditions. Adequate instrumentation is provided to indicate the venting operations to prevent undue interruption.

Site Radiological Environmental Monitoring

The RCGVS piping is part of the overall plant and is designed to vent noncondensable gases from the reactor vessel and the pressurizer. The RCGVS piping does not generate any radioactive materials and thus, site radiological environmental monitoring is not required.

5.4.12.3 Performance Evaluation

The RCGVS is designed to provide remote noncondensable gas venting from the reactor vessel closure head and the pressurizer steam space during post-accident conditions.

Redundant flow paths are provided in the vent paths from the reactor vessel closure head and pressurizer, respectively, to provide reasonable assurance of flow under single failure conditions. A redundant flow path contains two valves in series, in each flow path, to preclude spurious flow path initiation upon single failures and inadvertent open flow path. To achieve this, the two valves are powered by a different train of the Class 1E power source.

A break in the vent line on the reactor vessel closure head (RVCH) is categorized as a small break LOCA of not greater than NPS of 2.54 cm (1 in) in diameter. A break phenomenon (or behavior) of the RVCH vent line is similar to the breaks that are analyzed in Subsection 15.6.5. Hence, the results presented in Subsection 15.6.5 conservatively envelop the RVCH vent line break case.

The evaluation of the reactor coolant gas vent system operation is as follows:

a. The operation is needed when venting the noncondensable in the upper reactor vessel is necessary.

b. The size of a noncondensable bubble is estimated from reading the reactor vessel water level indication.

c. Initiating and terminating system operation are manually performed in accordance with the above described conditions.

d. The temperature and pressure instrumentation is provided to detect RCS leakage and pressure buildup, respectively.

e. The operator action to open the RCGV valve may be needed to vent steam in the reactor vessel closure head or to release steam to the IRWST.

The venting operation is performed in accordance with system operating procedure for the RCGVS to discharge noncondensable gases and steam from the high point of the RCS during post-accident conditions. The system operating procedure for the RCGVS is described in Section 13.4.

5.4.12.4 Inspection and Testing Requirements

Subsection 3.9.6 describes inservice testing and inspection of valves. Subsection 5.2.4 describes inservice inspection and testing of ASME Code, Class 1 components that are part of the reactor coolant pressure boundary.

5.4.12.5 Instrumentation Requirements

Temperature instrumentation is provided downstream of the RCGV valves (RG-418, RG-419, and RG-420) to detect RCS leakage during normal operation when the valves are closed. These instruments are not required for post-accident operation. Temperature readouts are provided in the MCR and RSR.

A pressure instrument is provided downstream of the dual RCGV valves but before the isolation valves to the IRWST to detect pressure buildup in this region. Pressure readout is provided at the MCR and RSR.

All RCGV valves are operated in the MCR and the RSR.

5.4.13 Main Steamline Flow Restrictor

5.4.13.1 Design Basis

The steam 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 steamline. The flow rate is decreased by the small flow area and is limited to sonic velocity. 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 steamline piping are reduced, and stresses on internal steam generator components, particularly the tubesheet and tubes, are limited. The restrictor is also designed to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.13.2 Design Description

The steam generator steam nozzles are one-piece forgings with an integral venturi-type flow restrictor. The flow restrictor with throat diameter of 38.96 cm (15.34 in) is installed at 72.66 cm (28.607 in) inner diameter steam generator outlet nozzle.

5.4.13.3 Design Evaluation

The flow restrictor design has been sufficiently analyzed to provide reasonable assurance of its structural adequacy. The maximum pressure drop through the restrictor at the full power flow rate of 2.04×10^6 kg/hr (4.49×10^6 lb/hr) is approximately 0.42 kg/cm^{2}D (6 psid) in the nozzle divergent section, and 0.56 kg/cm^{2}D (8 psid) from the steam generator steam dome to the steam nozzle outlet. Construction materials and manufacturing of the flow restrictor are in accordance with ASME Section III, Class 1.

5.4.13.4 Tests and Inspections

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

5.4.14 Safety and Relief Valves

5.4.14.1 Design Basis

The POSRVs on the pressurizer are designed to protect the primary system, as required by ASME Section III.

The design basis for establishing the relieving capacity of the pressurizer POSRVs is presented in Subsection 5.2.2. For the postulated transients presented in Chapter 15, the results indicate that relieving capacity of the POSRVs is sufficient to provide overpressure protection.

Safety valves on the steam side of each SG are designed to protect the steam system, as required by ASME Section III. They are conservatively sized to pass a steady-flow equivalent to the maximum expected power level at the design pressure of the steam system.

5.4.14.2 Description

The RCS has four pressurizer POSRVs to provide overpressure protection. A typical POSRV is illustrated in Figure 5.4.14-1. The design parameters are given in Table 5.4.14-1. The valves are connected to the top of the pressurizer and meet ASME Section

III requirements. Each valve has a body that comprises a nozzle, spindle, and disc. The pressurizer POSRVs pass sufficient pressurizer steam to limit the RCS pressure to 110 percent of design pressure (193.3 kg/cm^{2}A (2,750 psia)) following a loss of load with a delayed reactor trip, which is assumed to be initiated by the secondly generated safety grade signal from the RPS. A delayed reactor trip is assumed on a high-pressurizer pressure signal. To determine maximum steam flow through the pressurizer POSRVs, the MSSVs are assumed to be operational.

Overpressure protection for the shell side of the SGs and the main steam lines up to the inlet of the turbine stop valve is provided by the MSSVs. These valves are each sized to pass a steam flow of 430,913 kg/hr (950,000 lb/hr) at 92.83 kg/cm^{2}A (1,320 psia). This limits SG pressure to less than 110 percent of SG design pressure during worst-case transients. The MSSVs consist of 20 valves with staggered set pressures. The valves are spring-loaded safety valves fabricated in accordance with ASME Section III. Parameters for the MSSVs are given in Table 5.4.14-2.

The manual actuation of pressurizer POSRVs can be used for rapid depressurization for feed-and-bleed operation in an total loss of feedwater event.

5.4.14.3 Evaluation

Overpressure protection is discussed in Subsection 5.2.2.

5.4.14.4 Tests and Inspections

The valves are inspected during fabrication in accordance with ASME Section III requirements.

The inlet and outlet portions of the POSRVs are hydrostatically tested with water at the appropriate pressures as required by the applicable section of the ASME Section III. Set pressure and seat leakage tests can be performed with steam. Final seat leakage tests are performed before shipment using either hot air or hot nitrogen.

5.4.15 Component Supports

5.4.15.1 Design Basis

The criteria applied in the design of the RCS supports are that the specific function of the supported equipment be achieved during the conditions of normal, earthquake, POSRV actuation, IRWST discharge, and branch line pipe break (BLPB) conditions. BLPB includes feedwater line breaks, main steam line breaks and all LOCA conditions resulting from breaks not eliminated by leak-before-break analysis in piping to branch nozzles of the RCS. Specifically, the supports are designed to support the RCS components and to restrain the components in accordance with the stress and deflection limits of ASME Section III under the combined SSE, IRWST discharge and BLPB loadings.

5.4.15.2 Description

Figure 5.4.15-1 illustrates the RCS support points. A description of the supports for each supported component is as follows:

a. Reactor vessel supports

   The RV is supported by four vertical columns located under the vessel inlet nozzles. These columns are designed to be flexible to the horizontal direction to allow horizontal thermal expansion during heatup and cooldown. They also support the RV in the vertical direction.

   Horizontal keyways along the upper portion of the column guide the RV during thermal expansion and contraction of the RCS and maintain the vessel centerline.

   Four horizontal keys welded to the vessel bottom head and keyways of the column base plates allow free thermal expansion and contraction of the RV. The column base plate acts as a keyway to restrain the bottom of the RV for dynamic load conditions.

The supports are designed to accommodate normal, seismic, IRWST discharge, and BLPB loads. In addition to the design basis loads, irradiation effects are also considered in the fracture mechanics analysis of the columns.

Reactor vessel supports are shown in Figure 5.4.15-2.

b. Steam generator supports

The steam generator supports are shown in Figure 5.4.15-3. The steam generator is supported by a conical skirt welded to the steam generator lower head. The skirt provides a bolting surface for a heavy steel sliding base. Preloaded studs transfer loads from the skirt to the sliding base. Four low friction spherical head bearings under the plate are the sliding interface, which allows horizontal motion parallel to the hot leg due to thermal expansion. Machined cutouts in the sliding base act as keyways for embedded keys that support the generator horizontally during earthquakes, IRWST discharge, and a postulated pipe break. The keys are designed to resist dynamic event loads in a direction perpendicular to the hot leg piping. In the keyways, lower expansion plates (low friction bearings) are used to minimize resistance to thermal motion. The clearances between the lower expansion plate (low friction bearings) and the keys are shimmed and verified during hot functional testing.

Horizontal support at the top of the steam generator is provided by two keys and two hydraulic snubber assemblies. The keys and assemblies act as horizontal supports for the steam generator during earthquakes, IRWST discharge, and a postulated pipe break while allowing motion parallel to the hot leg due to thermal expansion. Upper expansion plates (low friction bearings) are bolted to the sides of the keyway. The clearances between the key and the upper horizontal supporting structures are shimmed and verified during hot functional testing.

Each snubber assembly consists of a lug welded to the steam generator, a lever, two links, a snubber, one clevis bracket pinned to the lever, and one clevis bracket pinned to the snubber as shown in Figure 5.4.15-3. Each clevis bracket is drilled to accept anchor bolts. Preloaded anchor bolts and the clevis brackets are the mechanism by which loads are transmitted to the concrete structure.

c. RCP supports

The supports control movement of the reactor coolant pump in the horizontal and vertical planes during earthquakes, IRWST discharge, and postulated pipe breaks, but accommodate motion due to thermal expansion.

The reactor coolant pump supports are shown in Figure 5.4.15-4. The reactor coolant pump and motor assembly is supported by four vertical columns pinned to the pump mounting skirt. It is supported for seismic, IRWST discharge, and postulated pipe break loads by two horizontal columns pinned to the top of the motor mount, two horizontal columns pinned to the pump support skirt, and a horizontal snubber system attached to the top of the motor mount.

Each column, horizontal and vertical, and the snubber assembly end in a clevis bracket drilled to accept anchor bolts. The loads are transmitted to the concrete structure through the clevis brackets and preloaded anchor bolts.

d. Pressurizer supports

The pressurizer is supported by a cylindrical skirt welded to the bottom head of the pressurizer. The skirt ends in a flange that is drilled to accept anchor bolts. Support loads are transmitted to the foundation through the skirt flange. Although most thermal growth is in the vertical direction, the pressurizer skirt design accommodates radial growth without bolt slippage.

Four keys welded to the upper portion of the pressurizer shell give additional support to the pressurizer during an earthquake, POSRV actuation, IRWST discharge, and a postulated pipe break. The clearance between the key and the supporting structure is shimmed and verified during hot functional testing.

5.4.15.3 Evaluation

Reasonable assurance of the structural integrity of the RCS support components is provided by quality assurance inspections in accordance with ASME Section III during fabrication of the supports. The non-integral supports are procured by individual equipment specifications that impose appropriate quality assurance requirements commensurate with the functions of the component.

During pre-operational testing of the RCS, the support displacements are measured to check the consistency with the calculated displacements and/or clearances. Subsequent inspections of supports that support RCS components are conducted in accordance with the ASME Section XI.

The COL applicant is to verify that the as-built RV support material properties and 60-year neutron fluence (COL 5.4(5)) are consistent with the following:

a. Maximum phosphorous, 0.015 percent per heat, and 0.018 percent per product analysis

b. Maximum copper, 0.15 percent per heat and per product analysis

c. Other chemical compositions consistent with SA508 chemistry

d. 60-year neutron fluence of 2.0×10^{18} n/cm^2 (E > 1.0 MeV)

If the requirements above are not met, a revised fracture mechanics analysis is performed.

5.4.16 Combined License Information

COL 5.4(1) The COL applicant is to prepare operational procedures and maintenance programs related to leak detection and contamination control of RCS.

COL 5.4(2) The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of RCS.

COL 5.4(3) The COL applicant is to prepare operational procedures and maintenance programs related to leak detection and contamination control of SCS.

COL 5.4(4) The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of SCS.

COL 5.4(5) The COL applicant is to verify the as-built RV support material properties and 60-year neutron fluence.

5.4.17 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." U.S. Nuclear Regulatory Commission.

3. Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Rev. 1, U.S. Nuclear Regulatory Commission, August 1975.

4. APR1400-A-M-NR-14001-P, "KHNP APR1400 Flywheel Integrity Report," KHNP, November 2014.

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components, The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

6. ASME PTC 8.2, "Centrifugal Pumps," The American Society of Mechanical Engineers, 1990.

7. NEMA MG-1, "Motors and Generators," National Electrical Manufacturers Association, 2009 (with 2010 Revision 1).

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9. ASME Section III, Appendix N, "Dynamic Analysis Methods," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

10. Bulletin 79-13, "Cracking in Feedwater System Piping," Rev. 1, U.S. Nuclear Regulatory Commission, August 30, 1979.

11. NEI 97-06, "Steam Generator Program Guidelines," Rev. 3, Nuclear Energy Institute, January 2011.

12. EPRI Report 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," Rev. 7, Electric Power Research Institue, October 2007.

13. EPRI Report 1022832, "PWR Primary-to-Secondary Leak Guidelines," Rev. 4, Electric Power Research Institue, September 2011.

14. EPRI Report 1019038, "Steam Generator Integrity Assessment Guidelines," Rev. 3, Electric Power Research Institue, November 2009.

15. EPRI Report 1016555, "Pressurized Water Reactor Secondary Water Chemistry Guidelines," Rev. 7, Electric Power Research Institue, February 2009.

16. EPRI Report 1014986, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Rev. 6, Electric Power Research Institue, December 2007.

17. 10 CFR 20.1406, "Minimization of Contamination," U.S. Nuclear Regulatory Commission.

18. Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation-Life Cycle Planning," Rev. 0, U.S. Nuclear Regulatory Commission, June 2008.

19. NUREG-0800, Standard Review Plan, BTP 5-4, "Design Requirements of the Residual Heat Removal System," Rev. 4, U.S. Nuclear Regulatory Commission, March 2007.

20. Bulletin 88-04, "Potential Safety-Related Pump Loss," U.S. Nuclear Regulatory Commission, May 5, 1988.

21. GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," U.S. Nuclear Regulatory Commission, June 28, 1989.

22. APR1400-E-N-NR-14005-P, "Shutdown Evaluation Report," KHNP, December 2014.

23. IEEE Std. 344-2004 (R2009), "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers (IEEE), 2005.

24. Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Rev. 3, U.S. Nuclear Regulatory Commission, September 2009.

Table 5.4.1-1

Reactor Coolant Pump Parameters

Parameter	Value
Number of units	4
Type	Vertical, single-stage centrifugal
Rated total dynamic head, (1) m (ft)	109.7 (360)
Rated flow, L/min(2) (gpm)	460,256 (121,600)
Design pressure, kg/cm^2A (psia)	175.8 (2,500)
Design temperature, °C (°F)	343.3 (650)
Normal operating pressure, kg/cm^2A (psia)	158.2 (2,250)
Normal operating temperature, (1) °C (°F)	290.6 (555)
NPSH available (at rated flow), m (ft)	152.4 (500)
Suction temperature, (1) °C (°F)	290.6 (555)
Water volume, each, m^3 (ft^3)	3.26 (115)
Weight (including motor), dry, kg (lb)	144,515 (318,600)
Rotating inertia, pump, and motor: Assembly, minimum, kg-m^2 (lbs-ft^2)	6,717 (159,400)
Shaft seals	Mechanical face seals
Pump speed, (1) rpm	1,190
Motor synchronous speed, rpm	1,200
Motor type	AC induction
Horsepower, hot, (1) kW (hp)	7,457 (10,000)
cold, kW (hp)	10,067 (13,500)
Rated brake horsepower, kW (hp)	7,457 (10,000)
Voltage, V	13,200
Phase	3
Frequency, Hz	60
Insulation class	F
Starting current, at 100 % voltage, amps	3,426

(1) Parameters are related to four-pump, full-power operating conditions.

(2) RCP rated flow exceeds RCS design minimum flow in Sections 4.4 and 5.1 to provide reasonable assurance that the minimum flow is achieved.

Table 5.4.2-1 (1 of 2)

Steam Generator Design Parameters

Parameter[1]	Value
Number of units	2
Heat transfer rate per SG, kcal/hr (Btu/hr)	1.721×10^9 (6.830×10^9)
Number of tubes per SG	13,102
Average active tube length per SG, m (ft)	19.391 (63.62)
Heat transfer area per SG (approximate), m^2 (ft^2)	15,205 (163,670)
Primary Side	
Design pressure/temperature (kg/cm^2A/°C) (psia/°F)	175.76/343.33 (2,500/650)
Coolant inlet temperature, °C (°F)	323.88 (615)
Coolant outlet temperature, °C (°F)	290.55 (555)
Coolant flow rate, each, kg/hr (lb/hr)	37.78×10^6 (83.3×10^6)
Coolant volume at 68 °F each, m^3 (ft^3)	86.84 (3,066.99)
Tube size, OD, mm (in)	19.05 (0.75)
Tube thickness, nominal, mm (in)	1.0668 (0.042)
Primary inlet nozzle, number/inside diameter (ID), mm (in)	1/1,066.8 (42)
Primary outlet nozzle, number/ID, mm (in)	2/762 (30)
Secondary Side	
Design pressure/temperature, kg/cm^2A/°C (psia/°F)	84.36/298.88 (1,200/570)
Steam pressure, kg/cm^2A (psia)	70.3 (1,000)[2]
Steam flow rate (at 0.25 % moisture) per SG, kg/hr (lb/hr)	4.070×10^6 (8.975×10^6)
Feedwater temperature at full power, °C (°F)	232.22 (450)
Moisture carryover, weight maximum, %	0.25
Steam nozzle, number/ID, mm (in)	2/711.2 (28)
Feedwater nozzles, number/size/schedule (economizer)	2/14/120
Feedwater nozzles, number/size/schedule (downcomer)	1/6/(Special)

Table 5.4.2-1 (2 of 2)

Parameter[1]	Value
Overall heat transfer coefficient: Evaporator, kcal/hr-m^2 °C (Btu/hr-ft^2 °F) Economizer, kcal/hr-m^2 °C (Btu/hr-ft^2 °F)	 6,483.864 (1,328) 2,875.750 (589)
Inventory per SG at normal water level and full power, kg (lbm)	98,895 (218,027)
Boil dry time from normal water level of full power (approximate), min	26.67
Boil dry time from nominal low SG level of reactor trip (approximate), min	26.30

(1) Includes operation with continuous SG blowdown flow of 0.2 % of the steam flow rate

(2) Measured at the SG steam dome area

Table 5.4.7-1 (1 of 3)

Shutdown Cooling System Design Parameters

Parameter	Value
System Design Parameters	
Shutdown cooling system initiation	Approximately 3.5 hours after reactor shutdown
RCS maximum cooldown rate (at initiation of shutdown cooling), °C/hr (°F/hr)	41.7 (75)
Maximum shutdown cooling flow, L/min (gpm)/HX	23,217 (6,134)
Component Design Parameters	
Shutdown cooling heat exchanger: Quantity Type Service transfer rate, kcal/hr-m^2-°C (Btu/hr-ft^2-°F) Effective heat transfer area, m^2/HX(ft^2/HX)	 2 Shell and tube, horizontal U-tube 1,872.9 (383.6) 776.9 (8,362.5)
Tube side: Fluid Design pressure, kg/cm^2 (psig) Design temperature, °C (°F) Material Code Fouling resistance, m^2-K/W (hr-ft^2-°F/Btu)	 Reactor coolant 63.2 (900) 204.4 (400) Austenitic stainless steel ASME Section III, NC (1) 0.000088(0.0005)
Shell side: Fluid Design pressure, kg/cm^2 (psig) Design temperature, °C (°F) Material Code Fouling resistance, m^2-K/W (hr-ft^2-°F/Btu)	 Component cooling water 14.06 (200) 93.3 (200) Carbon steel ASME Section III, ND (2) 0.000088(0.0005)

Table 5.4.7-1 (2 of 3)

Parameter	Value
Component Design Parameters (cont.)	
At 96 hours after shutdown	
Tube side:	
Flow, kg/hr (lb/hr)	1.08×10^6 (2.38×10^6)
Inlet temperature, °C (°F)	48.9 (120)
Outlet temperature, °C (°F)	40.2 (104.4)
Shell side:	
Flow, million kg/hr (lb/hr)	2.49×10^6 (5.48×10^6)
Inlet temperature, °C (°F)	35 (95)
Outlet temperature, °C (°F)	38.8 (101.8)
Heat transfer rate, W (Btu/hr)	11.0×10^6 (37.4×10^6)
Shutdown cooling pump:	
Quantity	2
Type	Single stage, vertical, centrifugal
Safety classification	2
Code	ASME Section III, NC
Design pressure, kg/cm^2 (psig)	63.2 (900)
Design temperature, °C (°F)	204.4 (400)
Design flow rate, L/min (gpm)	20,536 (5,425) (3)
Design head, m (ft)	140.2 (460)
Materials	Stainless steel type 304,316
Seals	Mechanical
Brake power, kW (HP)	746 (1,000)
NPSH Available	5.79 m at 20,536 L/min (19 ft at 5,425 gpm) (3)
NPSH Required	5.49 m at 20,536 L/min (18 ft at 5,425 gpm) (3)
Shutdown cooling miniflow heat exchanger:	
Quantity	2
Type	Shell and tube, horizontal U-tube
Service transfer rate, kcal/hr-m^2-°C (Btu/hr-ft^2-°F)	1831.4 (375.1)
Effective heat transfer area, m^2/HX (ft^2/HX)	14.0 (150.2)

Table 5.4.7-1 (3 of 3)

Parameter	Value
Component Design Parameters (cont.)	
Tube side: Fluid Design pressure, kg/cm^2 (psig) Design temperature, °C (°F) Material Code	 Reactor coolant 63.3 (900) 204.4 (400) Austenitic stainless steel ASME Section III, NC
Shell side: Fluid Design pressure, kg/cm^2 (psig) Design temperature, °C (°F) Material Code	 Component cooling water 14.1 (200) 149 (300) Carbon steel ASME Section III, ND

(1) ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” Division 1, Subsection NC: “Class 2 Components,” ASME, the 2007 Edition and the 2008 Addenda.

(2) ASME Boiler and Pressure Vessel Code, Section III,” Rules for Construction of Nuclear Power Plant Components,” Division 1, Subsection ND: “Class 3 Components,” ASME, the 2007 Edition and the 2008 Addenda.

(3) Including the minimum bypass flow

Table 5.4.7-2 (1 of 13)

SCS Failure Modes and Effects Analysis

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
1.	SCS Suction Line Isolation Valve SI-651 SI-652 SI-653 SI-654	a) Fails closed	• Mechanical binding • Electrical malfunction • Corrosion	Prevention of decay heat removal from core via one SCS train during normal shutdown cooling or long term cooling following LOCA	• Periodic testing; • Valve position indication in control room	Redundant SCS train	
		b) Fails open	Same as 1a	None, LTOP transient alarm while SI-651/653 or SI-652/654 are not full closed with high RCS pressure transients and LTOP inhibited alarm while either valve not full open during a low RCS temperature	Same as 1a	The redundant series valve ensures that SCS is protected from normal RCS pressure during power operation	Interlocks associated with the valves prevent overpressurization. These interlocks prevent the valves in the suction line of the SCS from being opened if RCS pressure exceeds 31.6 kg/cm^2A (450 psia).
2.	SCS Suction Line Isolation Valve SI-655 SI-656	a) Fails closed	• Mechanical binding • Electrical malfunction • Corrosion	Inability to align one shutdown cooling subsystem for shutdown cooling	• Periodic testing • Valve position indication in control room	Redundant SCS train	
		b) Fails open	Same as 2a	None	Same as 2a	None required	Valve is normally locked closed in control room

Table 5.4.7-2 (2 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
3.	SCP Suction Isolation Valve SI-106 SI-107	a) Fails closed	• Mechanical binding • Corrosion • Operator error	Effective loss of one SCS train	• Periodic testing • Low flow indication (F-302A, 305B)	Redundant SCS train	
		b) Fails open	Same as 3a	None	Periodic testing	None	
4.	Shutdown Cooling Pump 1, 2	a) Fails to start	• Mechanical failure • Electrical malfunction	Effective loss of one SCS train	• Low flow indication (F-302A, 305B) • Periodic testing • Pump “run” light	Redundant SCP and CSP might permit shutdown cooling although the cooling time will be extended	When used in a containment spray configuration, the SCPs are capable of being automatically started by an SIAS or CSAS
5.	SCP MFHX Isolation Valve SI-265 SI-269	a) Fails closed	• Mechanical binding • Corrosion	Possible damage to associated SCP	Periodic testing	Redundant SCS train	Valve is normally locked open
		b) Fails open	Same as 5a	None	Periodic testing	None	
6.	SCP Miniflow Heat Exchanger 1, 2	a) Loss of cooling	Single failure in component cooling water system (CCWS)	Possible damage to associated SCP	Periodic testing	Redundant SCS train	
		b) Cross leakage	Corrosion	Leakage from SCP miniflow to the CCWS	Radiation and/or level indication in CCWS	Redundant SCS train	

Table 5.4.7-2 (3 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
7.	SCP Discharge Isolation Valve SI-578 SI-579	a) Fails closed	• Mechanical binding • Corrosion	Effective loss of one shutdown cooling pump	• Periodic testing • Low flow indication (F-302A, 305B)	Redundant SCS train	Valve is normally locked open; min. flow line will provide the min. flow required to protect the pump
		b) Fails open	Same as 7a	None	Periodic testing	None	
8.	Shutdown Cooling Heat Exchanger 1, 2	a) Loss of cooling water	• Insufficient component cooling water flow • Excessive fouling	Diminished ability of one subsystem to provide RCS heat removal	• Periodic testing • High temperature indication from T-302A, T-305B	Redundant SCS train	The SDCHXs are used during long term cooling following a LOCA
		b) Cross leakage	Corrosion	Leakage from SCS system to the CCWS	Radiation and/or level indication in CCWS	Redundant SCS train	Same as 8.a)
		c) Plugged Tube	• Corrosion • Foreign material in RCS	• No flow through one SDCHX • Reduced cooling capability	Position indication in control room, no delta temperature across SDCHX as indicator by T-302A, 305B	Redundant SCS train	

Table 5.4.7-2 (4 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
9.	SDCHX Bypass Flow Control Valve SI-312 SI-313	a) Fails closed	• Mechanical binding • Electrical malfunction • Corrosion	• During initial stage of shutdown cooling, one shutdown cooling subsystem would pump excessively cooled water to the reactor core. • Reduction in shutdown cooling margin	• Periodic testing • Valve position • Low temperature indication from T-302A, 305B	Operator can turn off the SCS subsystem. The RCS is pre-borated to provide sufficient shutdown margin	
		b) Fails open	Same as 9a	Reduced reactor coolant flow through one SDCHX	• Periodic testing • Valve position • High temperature indication from T-302A, 305B	Redundant SCS subsystem will assure shutdown cooling although cooling time will be extended	
10.	SDCHX Outlet Flow Control Valve SI-310 SI-311	a) Fails closed	• Mechanical binding • Electrical malfunction • Corrosion	• No flow through one SDCHX • Reduced cooling capability	Position indication in control room, no delta temperature across SDCHX as indicator by T-302A, 305B	Redundant SCS train	
		b) Fails open	Same as 10a	Inability to regulate and maintain cooldown rate	Valve position indication in control room; Low temperature indication from T-302A, 305B	Operator can turn off the SCS subsystem. The RCS is pre-borated to provide sufficient shutdown cooling margin	

Table 5.4.7-2 (5 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
11.	SCS Flow Indicator F-302A F-305B	False Indication	Electrical malfunction	• Inability to control cooldown rate in affected train • Possible isolation of functional SCS train	• Periodic testing • Comparison with redundant indicator, with all other process instrumentation and valve position indications	• Redundant SCS train • Redundant flow indicators	
12.	SCS Temperature Indicator T-302A T-305B T-300A T-303B T-301A T-304B	False Indication	Electrical malfunction	• Inability to control cooldown rate in affected train • Inconsistent reading with other temperature indicators	• Periodic testing • Comparison with redundant indicator, with all other process instrumentation and valve position indications	• Redundant SCS train • Redundant temperature indicators	
13.	SCS Pressure Indicator P-302 P-305 P-300 P-301	False Indication	Electrical malfunction	Inconsistent reading with other pressure indicators	• Periodic testing • Comparison with redundant indicator	• Redundant SCS train • Redundant pressure indicators	

Table 5.4.7-2 (6 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
14.	Shutdown Purification Isolation Valve SI-421 SI-420	a) Fails closed	• Mechanical binding • Corrosion	Inability to remove contaminants from one SCS flow path during shutdown cooling	• Periodic testing • The failure to purify would be detected by periodic sampling	Redundant purification connections to other SCS subsystem	
		b) Fails open	Same as 14a	None	Periodic testing	Series isolation valves in CVCS valve	Valve is normally locked closed and only open for shutdown purification
15.	SCS Test Return Line Isolation Valve SI-314 SI-315 SI-688 SI-693	a) Fails closed	• Mechanical binding • Corrosion	• No effect on SCS operation • Inability to perform IRWST cooling operation (SCP flow testing) or reactor cavity fill operation (SI-314/688 only)	• Periodic testing • Valve position indication in control room	None	
		b) Fails open	Same as 15a	None	• High temperature indication from T-301A, 304B • Periodic testing	Series isolation valves (SI-300/301) in IRWST return line	Valve is normally locked closed in control room

Table 5.4.7-2 (7 of 12)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
16.	IRWST Return Line Isolation Valve SI-300 SI-301	a) Fails closed	• Mechanical binding • Corrosion	• No effect on SCS operation • Inability to perform IRWST cooling operation (SCP flow testing) or reactor cavity fill operation (SI-300 only)	Periodic testing	None	
		b) Fails open	Same as 16a	None	• High temperature indication from T-301A, 304B • Periodic testing	Series isolation valves (SI-688/693) in SCS test return line	
17.	Reactor Cavity Isolation Valve SI-393 SI-391	a) Fails closed	• Mechanical binding • Corrosion	Inability to perform reactor cavity fill operation.	Periodic testing	None	
		b) Fails open	Same as 17a	None	Periodic testing	Series in isolation valves SI-393/391 are closed	Valve is normally locked closed in control room

Table 5.4.7-2 (8 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
18.	IRWST Return Line Isolation Valve SI-395 SI-959	a) Fails closed	• Mechanical binding • Corrosion	Inability to perform IRWST cooling operation (SCP flow testing)	Periodic testing	None	Valve is normally locked open
		b) Fails open	Same as 18a	No effect on IRWST cooling operation. Inability to perform reactor cavity fill operation (SI-395 only)	Periodic testing	None required	
19.	SCS Warmup Line Flow Control Valve SI-691 SI-690	a) Fails closed	• Mechanical binding • Electrical malfunction • Corrosion	Inability to gradually warm up the shutdown cooling line during the shutdown cooling alignment procedure	• Periodic testing • Valve position indication in control room	Redundant shutdown cooling subsystem train will not be affected	The safety injection piping and nozzles are designed for a limited number of thermal cycles that could result from operating the SCS without prior warmup
		b) Fails open	Same as 19a	Diversion of flow from discharge leg to suction leg of SCS without passing through the reactor core during shutdown cooling operation	• Periodic testing • Valve position indication in control room	Redundant shutdown cooling subsystem train will not be affected	These valves are gradually closed once warmup and flowrate stability have been reached

Table 5.4.7-2 (9 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
20.	SCS Line Isolation Valve SI-601 SI-600	a) Fails closed	• Mechanical binding • Electrical malfunction • Corrosion	Inability to inject cooled coolant into one of the DVI injection line	• Periodic testing • Valve position indication in the control room	Redundant SCS train	
		b) Fails open	Same as 20a	Inability to gradually warm up the shutdown cooling line during the shutdown cooling alignment procedure	Same as 20a	Redundant shutdown cooling subsystem will not be affected	The safety injection piping and nozzles are designed for a limited number of thermal cycles which could result from operating the SCS without prior warmup
21.	SCS/CSS Pump Suction Cross Connect Valve SI-340 SI-342	a) Fails closed	• Mechanical binding • Electrical malfunction	• No effect on SCS operation • Inability to perform SCS operation using CSP or containment spray operation using SCP	• Periodic testing • Valve position indication in the control room	None	These valves can be aligned for SCS/CSS interconnection
		b) Fails open	Same as 21a	Loss of one SCS train	• Low flow in SCS • Periodic testing • Valve position indication in the control room	Redundant SCS train	Valve is normally locked closed in control room

Table 5.4.7-2 (10 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
22.	SCS/CSS Pump Discharge Cross Connect Valve SI-341 SI-343	a) Fails closed	• Mechanical binding • Electrical malfunction	• No effect on SCS operation • Inability to perform SCS operation (or IRWST cooling operation) using CSP or containment spray operation using SCP	• Periodic testing • Valve position indication in the control room	None	These valves can be aligned for SCS/CSS interconnection
		b) Fails open	Same as 22a	Loss of one SCS train	• Low flow in SCS • Periodic testing, Valve position indication in the control room	Redundant SCS train	Valve is normally locked closed in control room
23.	SCP Suction Isolation Valve SI-344 SI-346	a) Fails closed	• Mechanical binding • Electrical malfunction	• No effect on SCS operation • Inability to perform IRWST cooling operation (SCP flow testing) or reactor cavity fill operation	• Periodic testing • Valve position indication in the control room	None	
		b) Fails open	Same as 23a	None	Same as 23a	Series in check valve SI-159/160 are closed	Valve is normally locked closed in control room

Table 5.4.7-2 (11 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
24.	CSP Suction Isolation Valve SI-347 SI-348	a) Fails closed	• Mechanical binding • Electrical malfunction	• No effect on SCS operation • Inability to perform IRWST cooling operation using CSP or containment spray operation using SCP	• Periodic testing • Valve position indication in the control room	Redundant SCS train	Valve is normally locked open in control room
		b) Fails open	Same as 24a	None	Same as 24.a)	None	
25.	Test Connection Isolation Valve SI-967 SI-977	a) Fails closed	• Mechanical binding • Corrosion	Inability to perform leakage test of containment isolation valve	None	None required for shutdown cooling	
		b) Fails open	Same as 25a	None	Operator	Series isolation valves SI-968/978 are closed	Valve is normally closed
26.	Test Connection Isolation Valve SI-965 SI-975	a) Fails closed	• Mechanical binding • Corrosion	Inability to perform hydrostatic test of SI-653/654 downstream piping	None	None required for shutdown cooling	
		b) Fails open	Same as 26a	None	Operator	End of downstream line is capped stub	Valve is normally closed

Table 5.4.7-2 (12 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
27.	Sampling System Isolation Valve SI-252 SI-253	a) Fails closed	• Mechanical binding • Corrosion	Unable to sample shutdown cooling water at this location	Operator	Other SCS sample locations	
		b) Fails open	Same as 27a	None	None	Series isolation valve is closed	
28.	Shutdown Cooling Line	a) One line clogs	Contaminants	Effective loss of one shutdown cooling subsystem	Low flow indications form F-302A or F-305B;	Redundant shutdown subsystem	Periodic sampling and flow check will monitor buildup of contaminants
		b) Limited leakage in one train	Seal failure	Release of coolant and radio- activity outside of containment	Local leak detection	The leak can be isolated without affecting the redundant subsystem	
29.	SDCHX Vent Valve SI-170 SI-180	a) Fails closed	• Mechanical binding • Corrosion	Unable to vent SDCHX	Periodic testing	None required for shutdown cooling	
		b) Fails open	Same as 29a	None	Periodic testing	None	Valve is normally closed
30.	SDCHX Drain Valve SI-172 SI-182	a) Fails closed	• Mechanical binding • Corrosion	Unable to drain SDCHX	Periodic testing	None required for shutdown cooling	
		b) Fails open	Same as 30a	None	Periodic testing	End of downstream line is flanged	Valve is normally closed

Table 5.4.7-2 (13 of 13)

No.	Name	Failure Modes	Cause	Symptoms and Local Effects Including Dependent Failure	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
31.	SCP MFHX Vent Valve SI-570 SI-571	a) Fails closed	• Mechanical binding • Corrosion	Unable to vent SCP MFHX	Periodic testing	None required for shutdown cooling	
		b) Fails open	Same as 31a	None	Periodic testing	None	Valve is normally closed
32.	SCP MFHX Drain Valve SI-576 SI-577	a) Fails closed	Mechanical binding Corrosion	Unable to vent SCP MFHX	Periodic testing	None required for shutdown cooling	
		b) Fails open	Same as 32a	None	Periodic testing	End of downstream line is flanged	Valve is normally closed

Table 5.4.10-1

Pressurizer Design Parameter

Property	Parameter
Design pressure, kg/cm^2A (psia)	175.8 (2,500)
Design temperature, °C (°F)	371.1 (700)
Normal operating pressure, kg/cm^2A (psia)	158.2 (2,250)
Normal operating temperature, °C (°F)	344.8 (652.7)
Internal free volume, m^3 (ft^3)	68.0 (2,400)
Normal (full power) operating water volume, m^3 (ft^3)	33.2 (1,171)
Normal (full power) steam volume, m^3 (ft^3)	35.7 (1,260)
Installed heater capacity, kW	2,400
Heater type	Immersion
Spray flow, minimum design capacity, L/min (gpm)	1,703.4 (450)
Bypass spray flow, continuous, L/min (gpm)	1.9~22.7 (0.5~6)
Nozzles: Surge, in (nominal) Spray, in (nominal) POSRV, in	 12, schedule 160 4, schedule 160 6, Liner ID 7.75, nozzle ID
Instrument: Level, in (nominal) Temperature, in (nominal) Pressure, in (nominal)	 3/4, schedule 160 1, schedule 160 3/4, schedule 160
Heater, OD, mm (in)	31.75 (1-1/4)

Table 5.4.10-2

Pressurizer Tests

Component	Test
Heads Plates Cladding	 UT, MT UT, PT
Shell Plates Cladding	 UT, MT UT, PT
Heaters Tubing Centering of elements End plug	 UT, PT RT UT, PT
Nozzle (forgings)	UT, MT
Studs	UT, MT
Welds Shell, longitudinal Shell, circumferential Cladding Nozzles Nozzle safe ends	 RT, MT RT, MT UT, PT RT, MT RT, PT
Instrument connections	PT
Support skirt	MT, RT
Temporary attachment after removal	MT
All welds after hydrostatic test	MT or PT
Heater assembly, end plug weld	PT

UT = Ultrasonic testing
MT = Magnetic particle testing
PT = Dye-penetrant testing
RT = Radiographic testing

Table 5.4.12-1

Reactor Coolant Gas Vent System – Active Valve List

Valve Number	Type	Line Size – Schedule	Power Source 125V DC Bus	Actuator	Safety Class
RG-410	Globe	50 mm (2 in) - 160	A	Solenoid	1
RG-411	Globe	50 mm (2 in) -160	B	Solenoid	I
RG-412	Globe	50 mm (2 in) - 160	C	Solenoid	I
RG-413	Globe	50 mm (2 in) - 160	D	Solenoid	I
RG-414	Globe	50 mm (2 in) - 160	A	Solenoid	I
RG-415	Globe	25 mm (1 in) - 160	B	Solenoid	I
RG-416	Globe	25 mm (1 in) - 160	C	Solenoid	I
RG-417	Globe	25 mm (1 in) - 160	D	Solenoid	I
RG-419	Globe	80 mm (3 in) - 160	B	Solenoid	I
RG-420	Globe	80 mm (3 in) - 160	A	Solenoid	I

Table 5.4.14-1

Pilot-Operated Safety Relief Valve Parameters

Parameter	Value
Type	Pilot operated
Design pressure, kg/cm^2A (psia)	175.8 (2,500)
Design temperature, °C (°F)	371.1 (700)
Fluid	Saturated steam, 4,400 ppm boron, pH = 4.5 to 10.6
Set pressure, kg/cm^2A (psia)	173.7 (2,470)
Set pressure uncertainty, %	± 0.75
Minimum capacity, kg/hr (lb/hr)	244,900 (540,000)
Maximum capacity, kg/hr (lb/hr)	285,700 (630,000)
Maximum opening/closing dead time, sec	0.2/0.4 (hydraulic actuation)
Maximum opening/closing time (including dead time), sec	Open (hydraulic): 0.5 Close(hydraulic): 0.9
Closing pressure, kg/cm^2A (psia)	87 % of opening pressure
Typical materials: Body Spindle Inlet/outlet flange	 Type 304 or 316, austenitic stainless steel Type 304 or 316, austenitic stainless steel Type 304 or 316, austenitic stainless steel

Table 5.4.14-2

Main Steam Safety Valve Parameters

Parameter	Value
Design pressure, kg/cm^2 (psig)	86.48 (1,230)
Design temperature, °C (°F)	299 (570)
Fluid	Saturated steam
Set pressure, kg/cm^2 g (psig)	82.5 (1,174) 84.7 (1,205) 86.5 (1,230) 86.5 (1,230) 86.5 (1,230)
Set pressure uncertainty, manufacture/operation, %	±1/±3
Minimum capacity, kg/sec (lbm/hr) at 110 % of steam generator design pressure	2,394 (19 × 10^6) total (20 valves)
Type	Spring loaded
Orifice area, cm^2 (in^2)	106.45 (16.5)
Accumulation, %	3
Back pressure:	
Maximum buildup/maximum superimposed, kg/cm^{2}G (psig)	10.19/0 (145/0)
Approximate dry weight, kg (lbs)	771 (1,700)
Blowdown, %	5
Typical materials:	
Body	ASME SA 105
Disc	ASME SB 637 UNS N07718
Nozzle	ASME SA 182, GR. F316

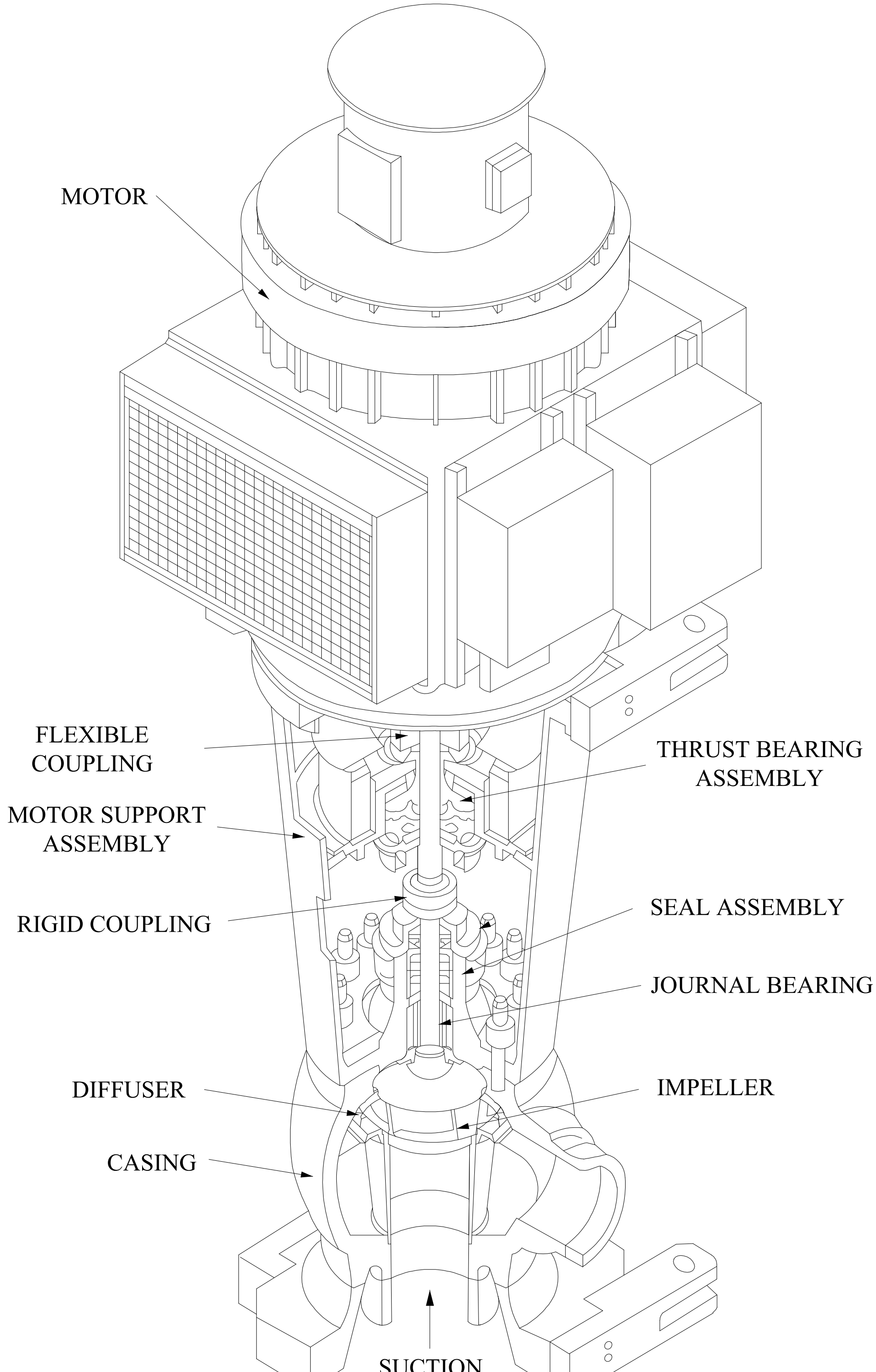
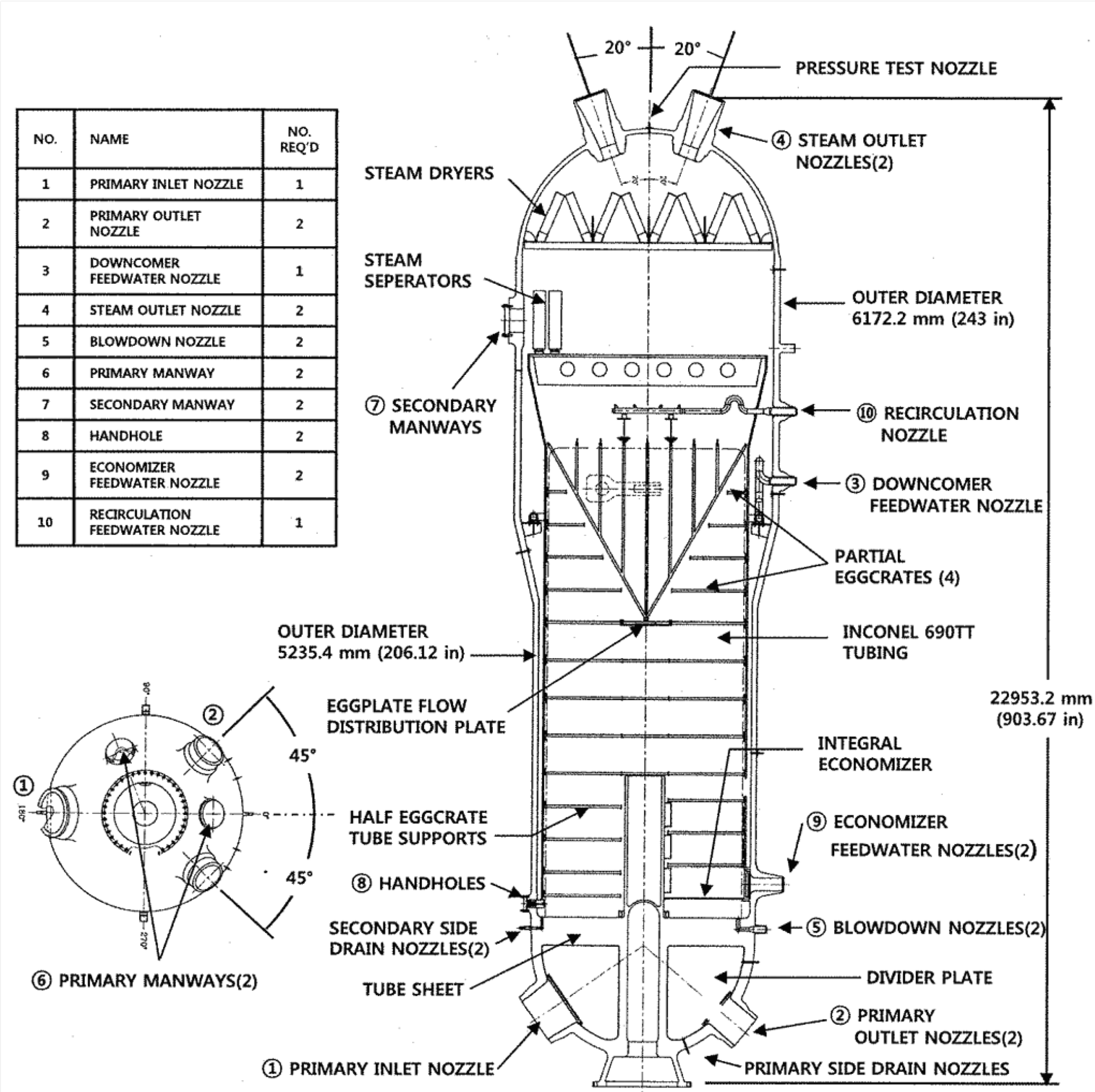


Figure 5.4.1-1 Reactor Coolant Pump



NO.	NAME	NO. REQ'D
1	PRIMARY INLET NOZZLE	1
2	PRIMARY OUTLET NOZZLE	2
3	DOWNCOMER FEEDWATER NOZZLE	1
4	STEAM OUTLET NOZZLE	2
5	BLOWDOWN NOZZLE	2
6	PRIMARY MANWAY	2
7	SECONDARY MANWAY	2
8	HANDHOLE	2
9	ECONOMIZER FEEDWATER NOZZLE	2
10	RECIRCULATION FEEDWATER NOZZLE	1

Figure 5.4.2-1 Steam Generator

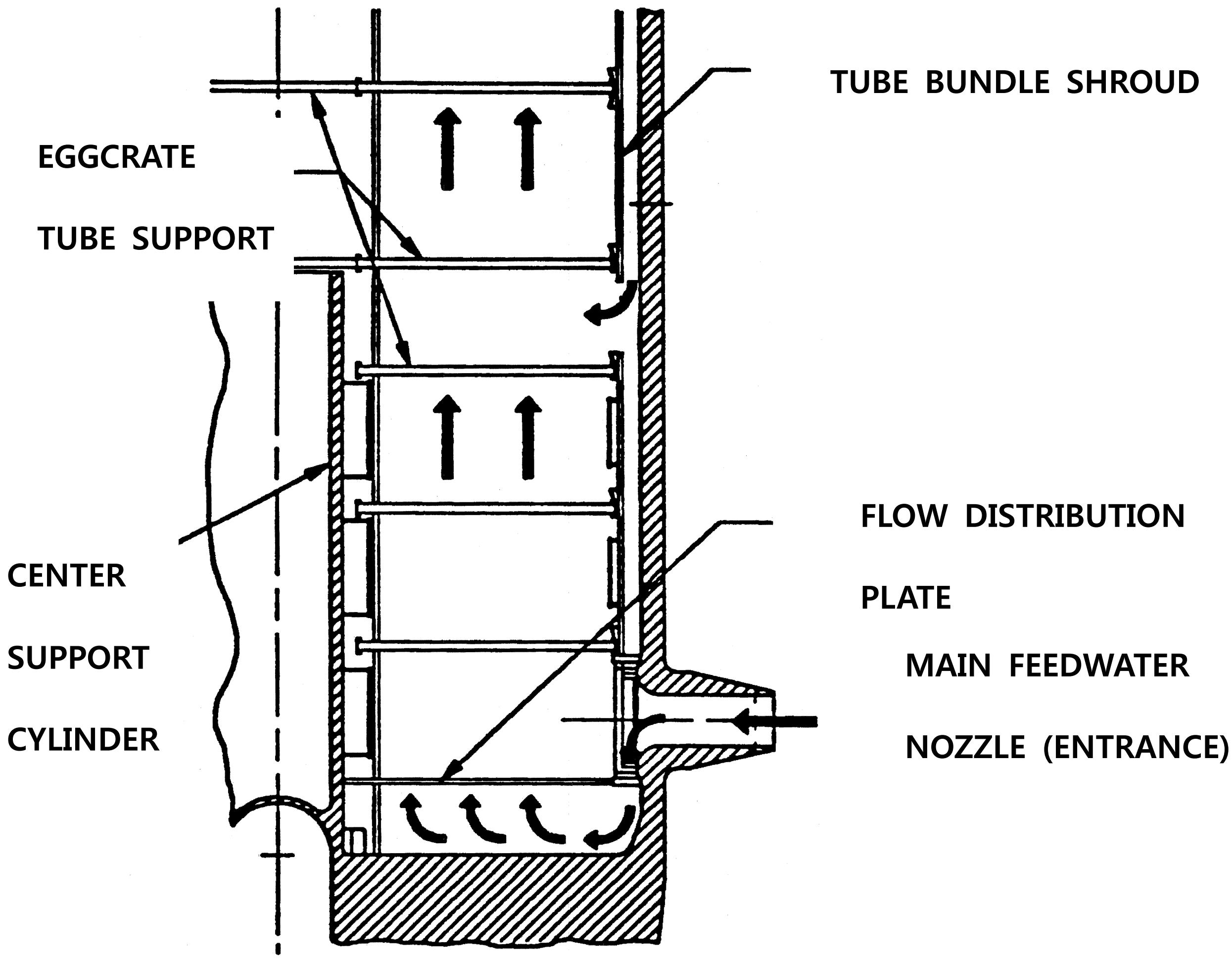


Figure 5.4.2-2 Steam Generator Economizer and Lower Tube Bundle Region

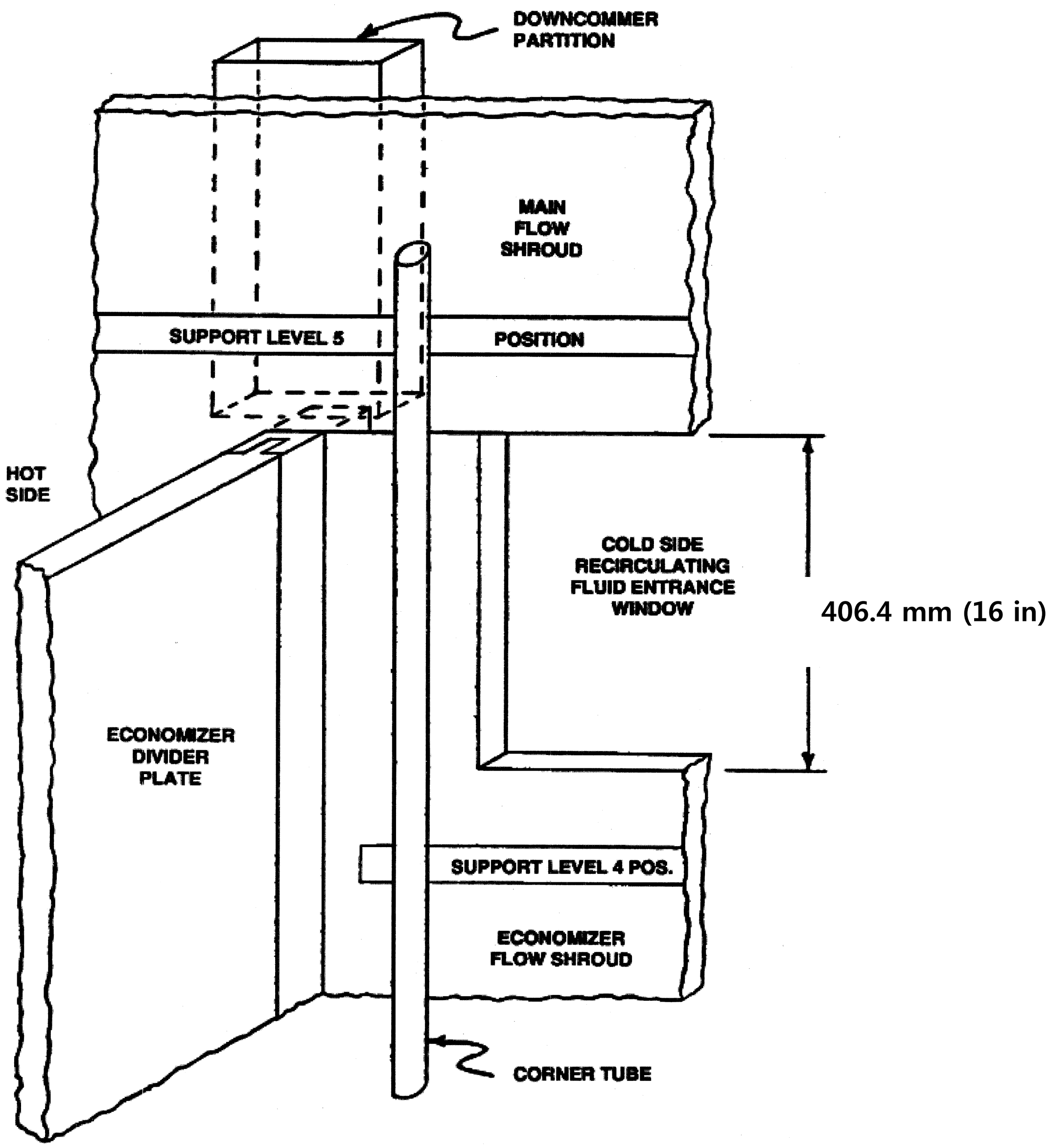


Figure 5.4.2-3 Steam Generator Cold Side Recirculating Fluid Entrance Region

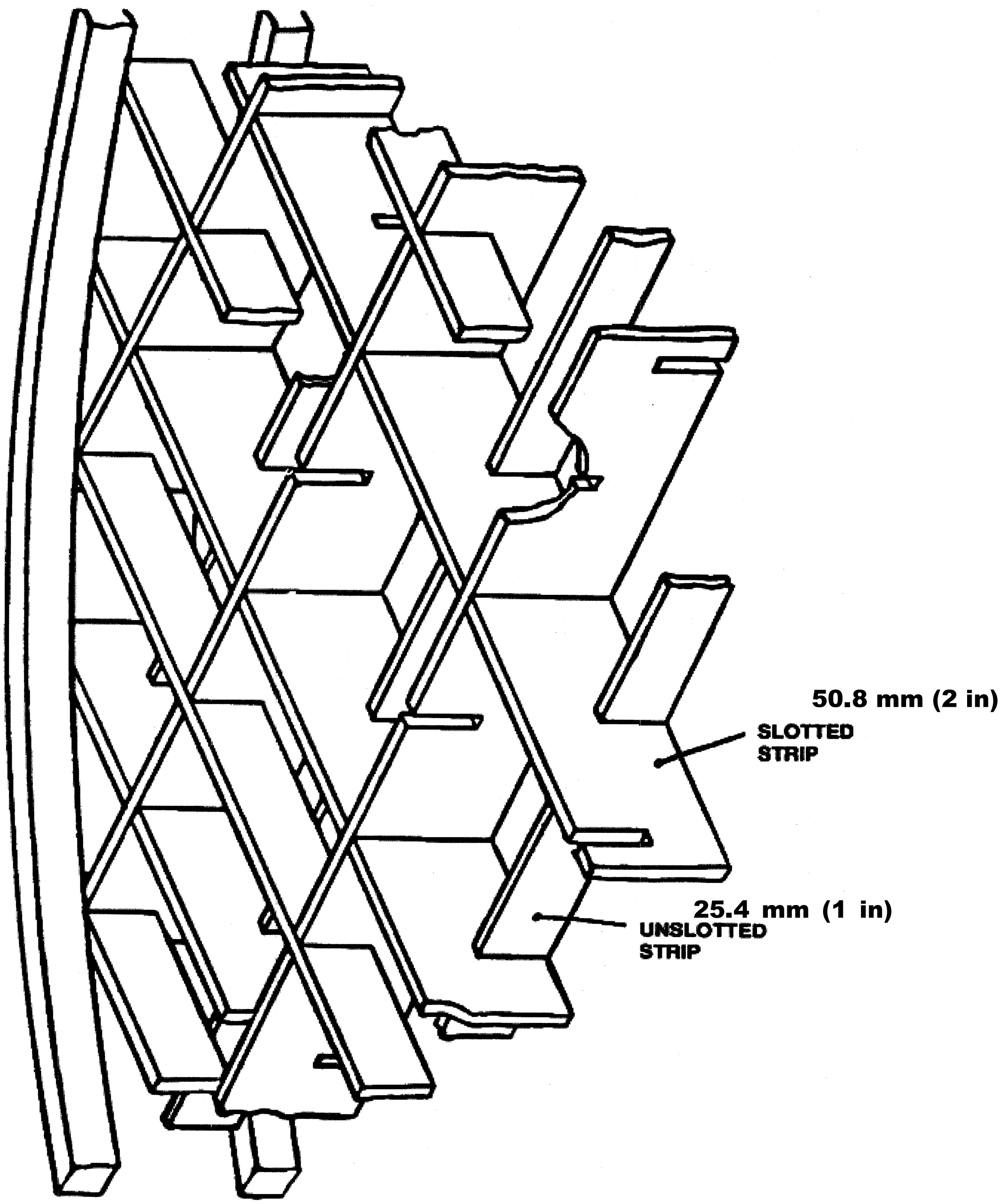


Figure 5.4.2-4 Steam Generator Tube Eggcrate Support

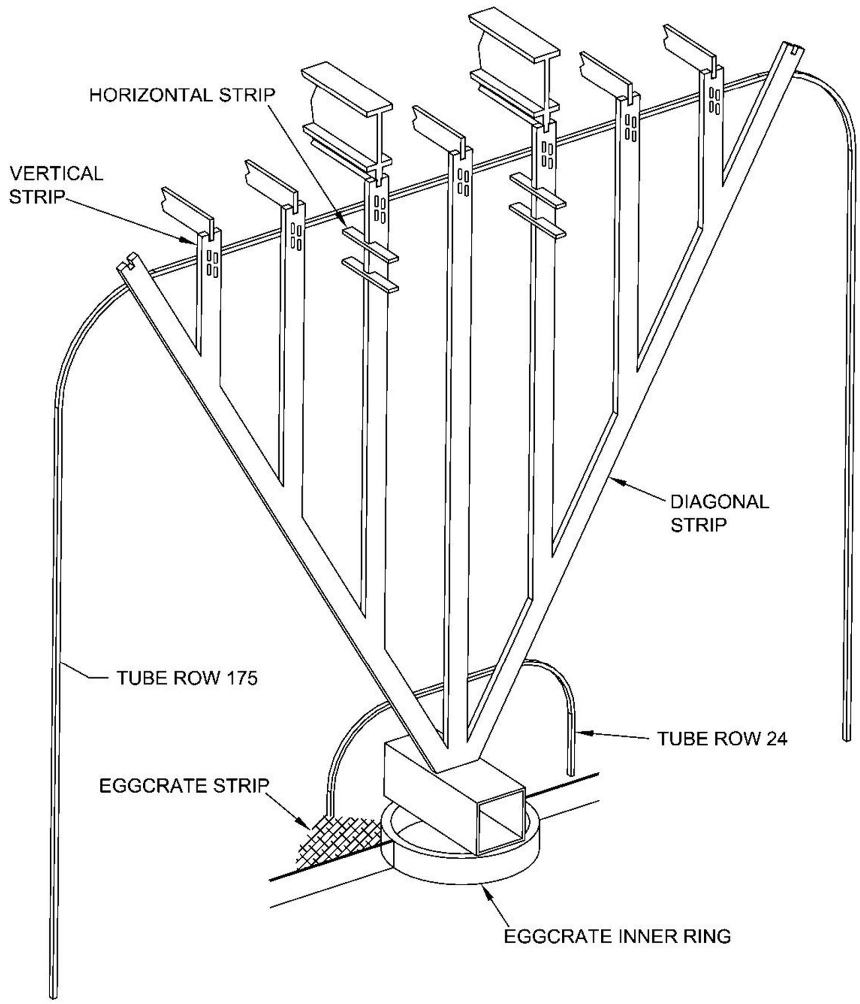


Figure 5.4.2-5 Steam Generator Tube Vertical Supports

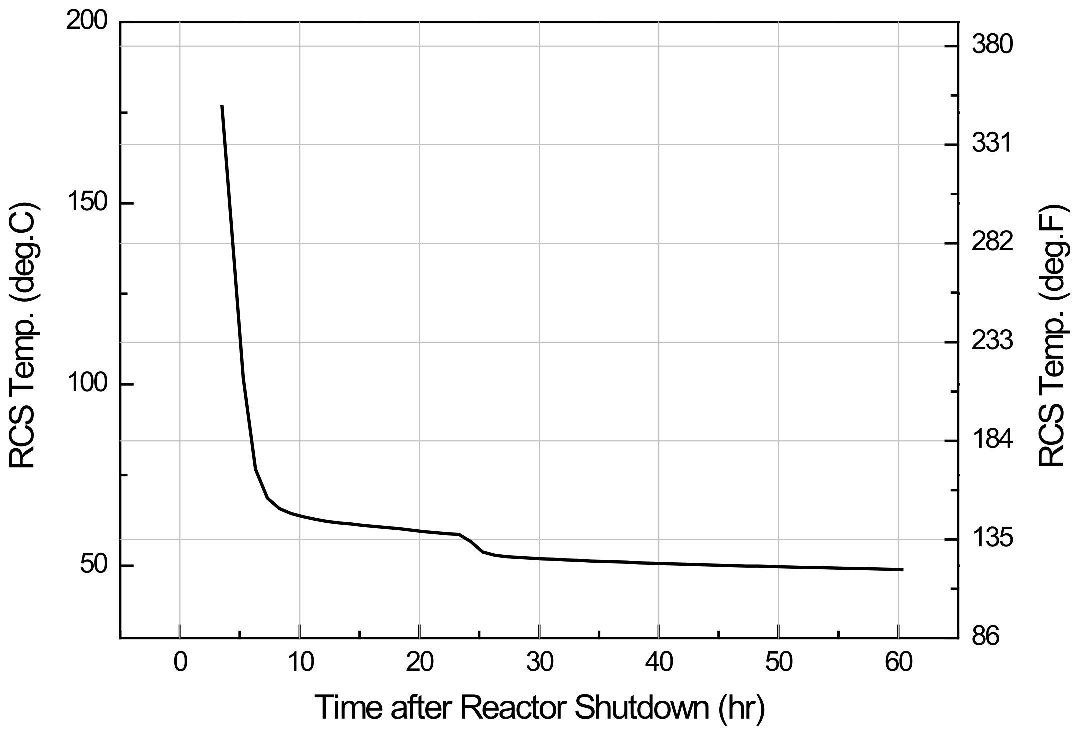


Figure 5.4.7-1 Shutdown Cooling System; Two Train Cooldown

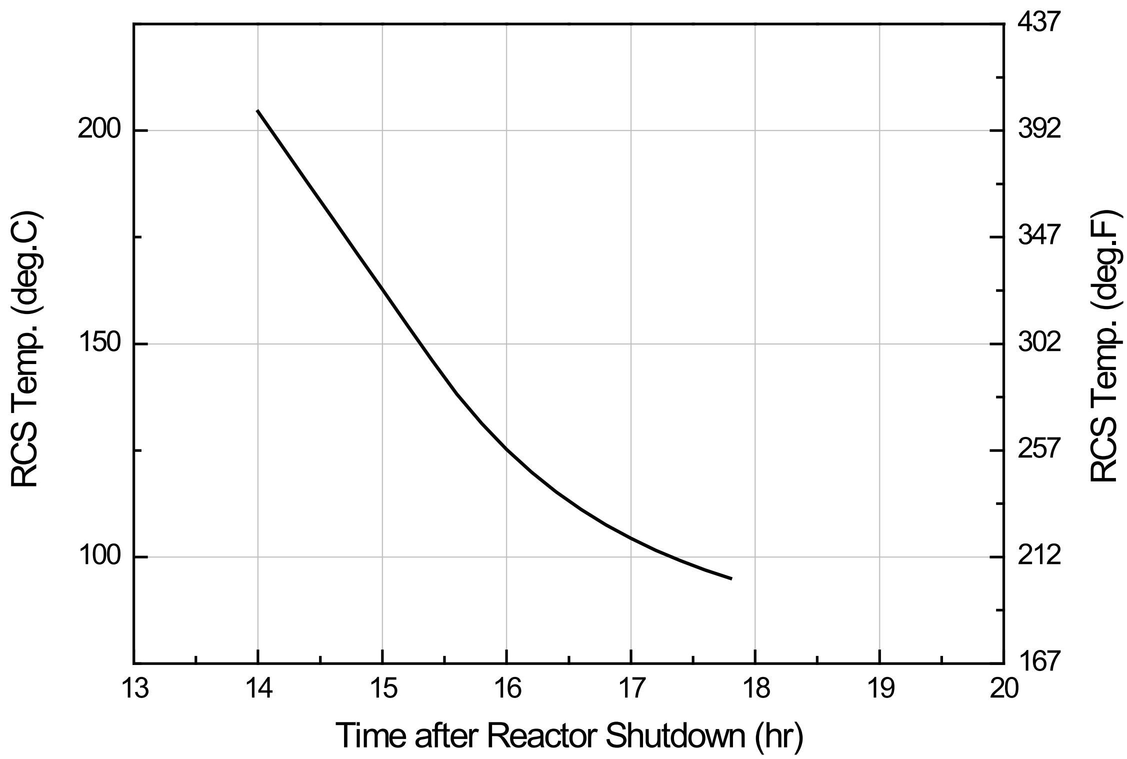


Figure 5.4.7-2 Shutdown Cooling System; One Train Cooldown

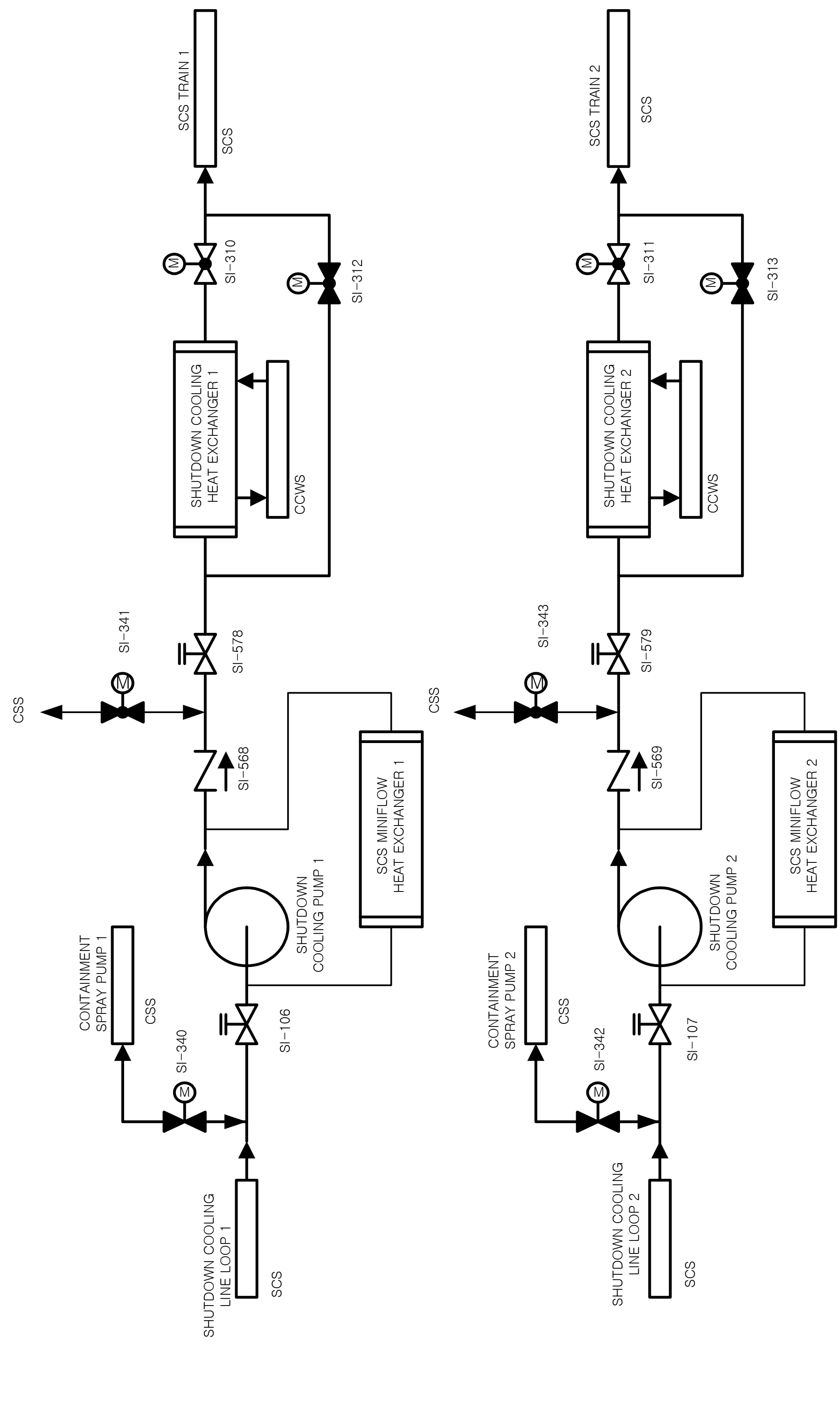


Figure 5.4.7-3 Shutdown Cooling System Flow Diagram; Shutdown Cooling Mode (1 of 2)

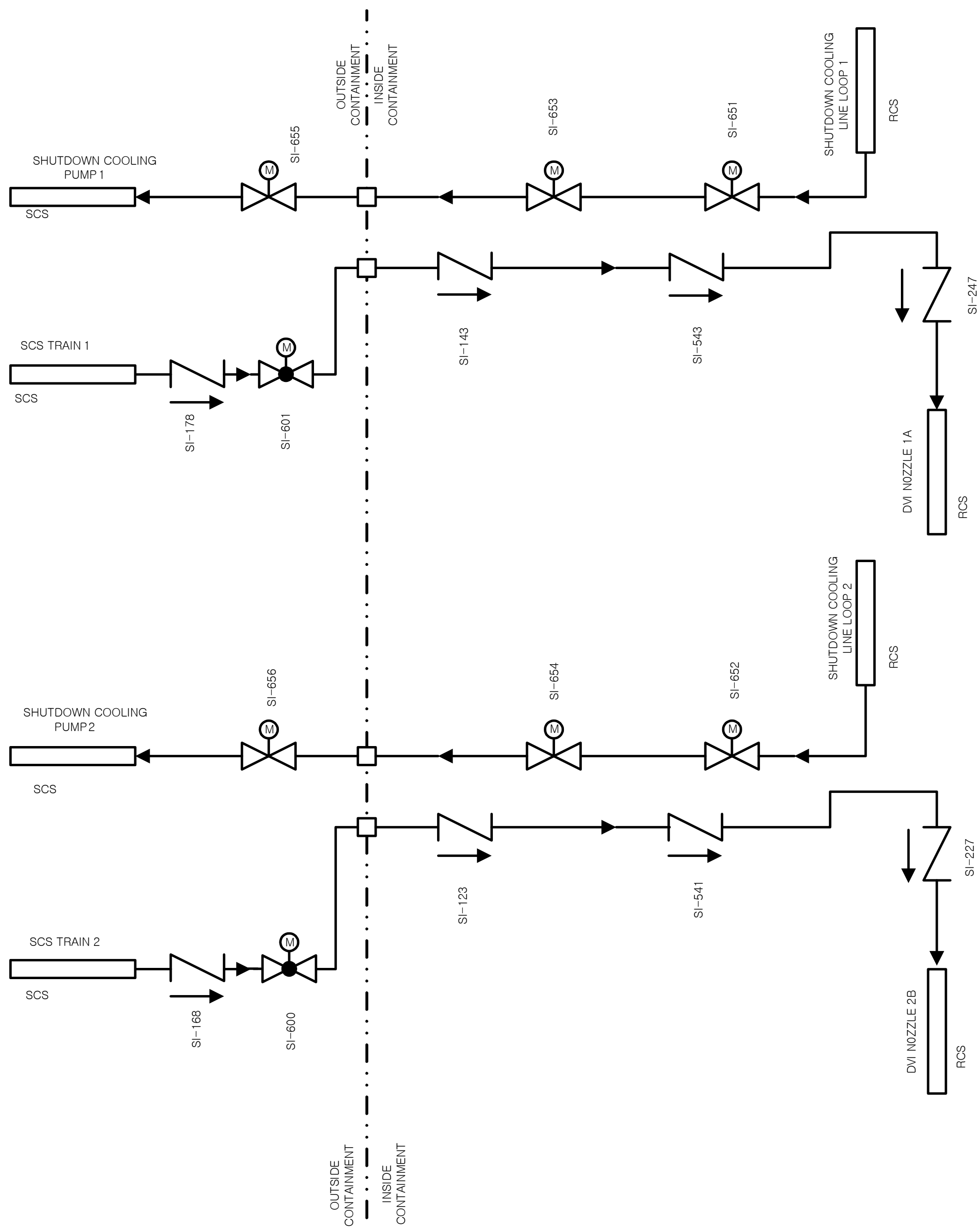


Figure 5.4.7-3 Shutdown Cooling System Flow Diagram; Shutdown Cooling Mode (2 of 2)

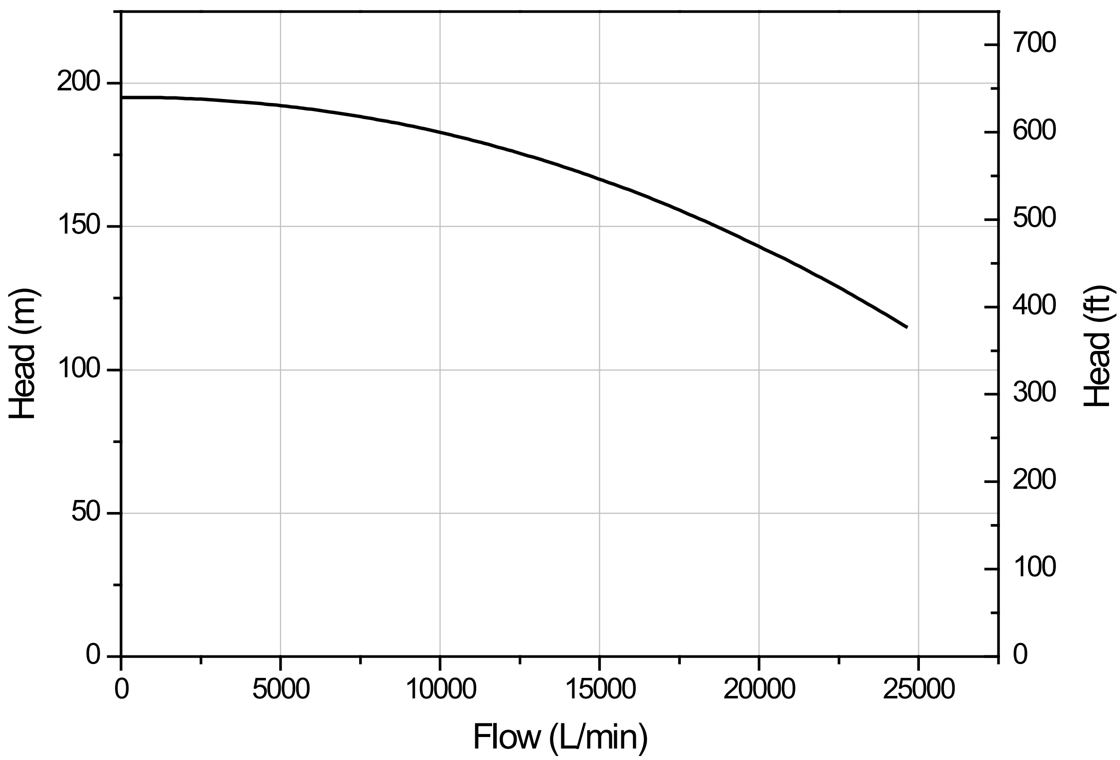


Figure 5.4.7-4 Shutdown Cooling Pump Characteristic Curve (Typical)

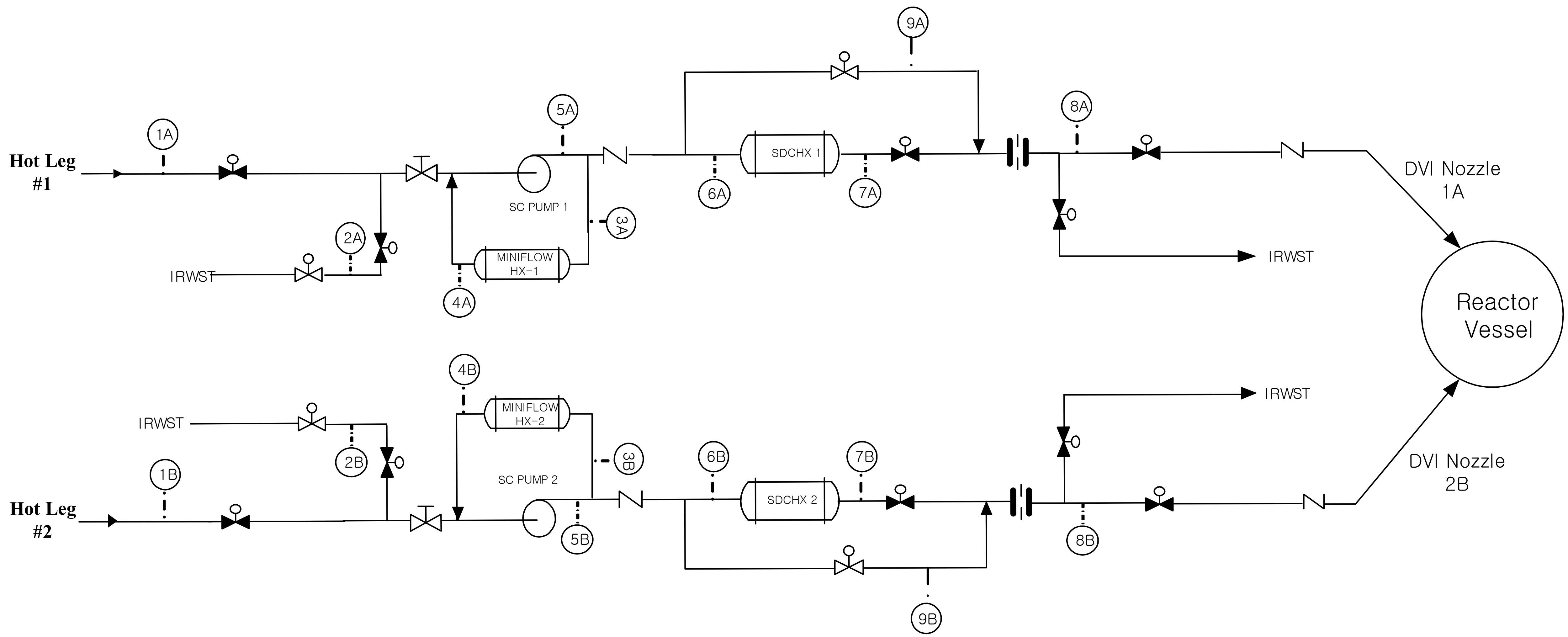


Figure 5.4.7-5   Shutdown Cooling System Mode Diagram (1 of 2)

Location		1A	2A	3A	4A	5A	6A	7A	8A	9A	1B	2B	3B	4B	5B	6B	7B	8B	9B
Normal Shutdown	Pressure (kg/cm^{2}A)	31.6	1.3	44.2	44.2	44.2	44.2	44.2	44.2	44.2	31.6	1.3	44.2	44.2	44.2	44.2	44.2	44.2	44.2
	Temperature (°C)	~176.7	~48.9	~176.7	~151.1	<176.7	~176.7	~53.3	~53.3	~53.3	~176.7	~48.9	~176.7	~151.1	<176.7	~176.7	~53.3	~53.3	~53.3
	Flow Rate (L/min)	18,927	0	1,609	1,609	20,536	18,927	18,927	18,927	0	18,927	0	1,609	1,609	20,536	18,927	18,927	18,927	0
Safety Shutdown	Pressure (kg/cm^{2}A)	28.1	1.3	44.2	44.2	44.2	44.2	44.2	44.2	44.2	28.1	1.3	44.2	44.2	44.2	44.2	44.2	44.2	44.2
	Temperature (°C)	~193.3	~48.9	~193.3	~151.1	~193.3	~193.3	~53.3	~53.3	~53.3	~193.3	~48.9	-	-	-	-	-	-	-
	Flow Rate (L/min)	18,927	0	1,609	1,609	20,536	18,927	18,927	18,927	0	0	0	0	0	0	0	0	0	0
Refueling	Pressure (kg/cm^{2}A)	1.3	1.3	19.8	19.8	19.8	19.8	19.8	19.8	19.8	1.3	1.3	19.8	19.8	19.8	19.8	19.8	19.8	19.8
	Temperature (°C)	48.9	~48.9	48.9	~46.7	48.9	48.9	~40.0	~40.0	~40.0	48.9	~48.9	48.9	~46.7	48.9	48.9	~40.0	~40.0	~40.0
	Flow Rate (L/min)	18,927	0	1,609	1,609	20,536	18,927	18,927	18,927	0	18,927	0	1,609	1,609	20,536	18,927	18,927	18,927	0
Startup	Pressure (kg/cm^{2}A)	31.6	1.3	44.2	44.2	44.2	44.2	44.2	44.2	44.2	31.6	1.3	44.2	44.2	44.2	44.2	44.2	44.2	44.2
	Temperature (°C)	~176.7	~48.9	~176.7	~151.1	~176.7	~176.7	~176.7	~176.7	~176.7	~176.7	~48.9	~176.7	~151.1	~176.7	~176.7	~176.7	~176.7	~176.7
	Flow Rate (L/min)	18,927	0	1,609	1,609	20,536	0	0	18,927	18,927	18,927	0	1,609	1,609	20,536	0	0	18,927	18,927

Figure 5.4.7-5   Shutdown Cooling System Mode Diagram (2 of 2)

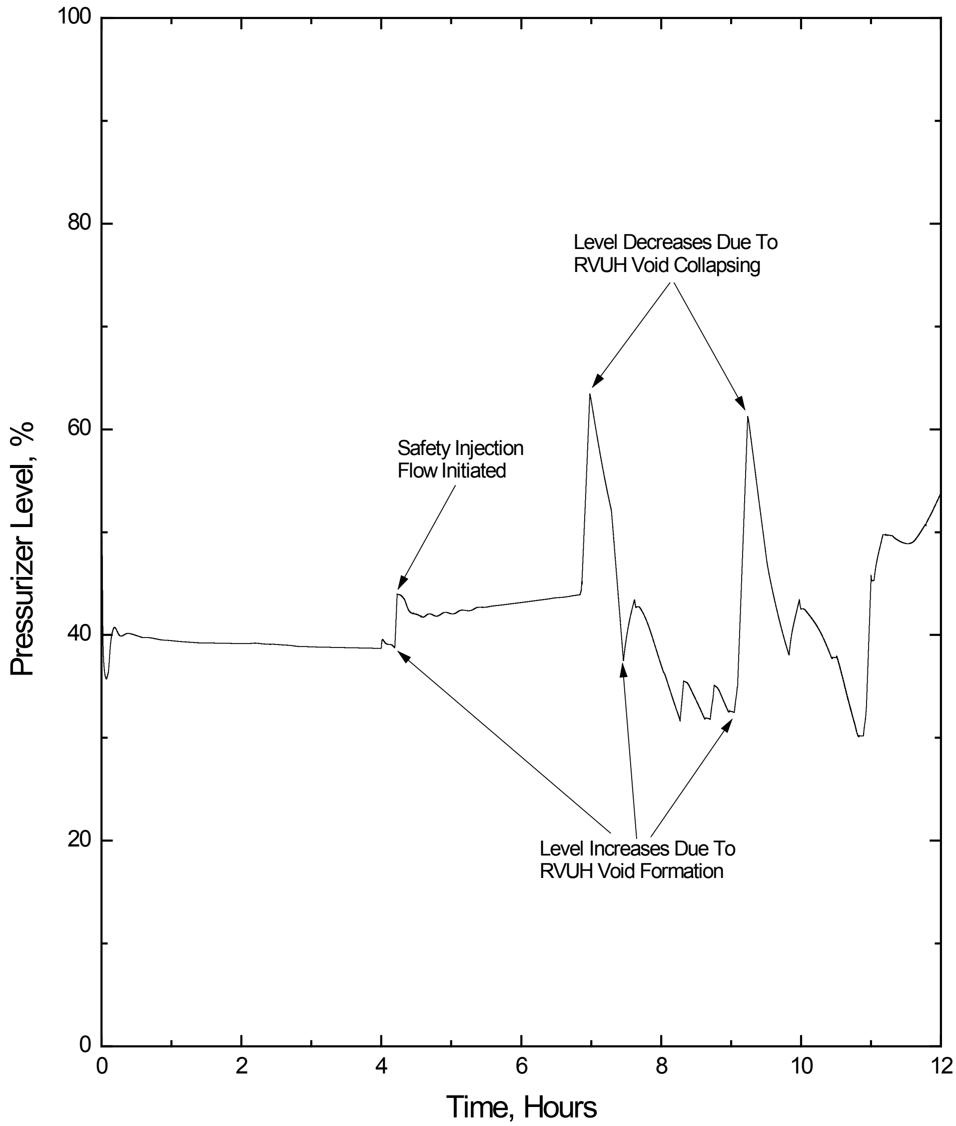


Figure 5.4.7-6 Pressurizer Level vs. Time for NCC Transient

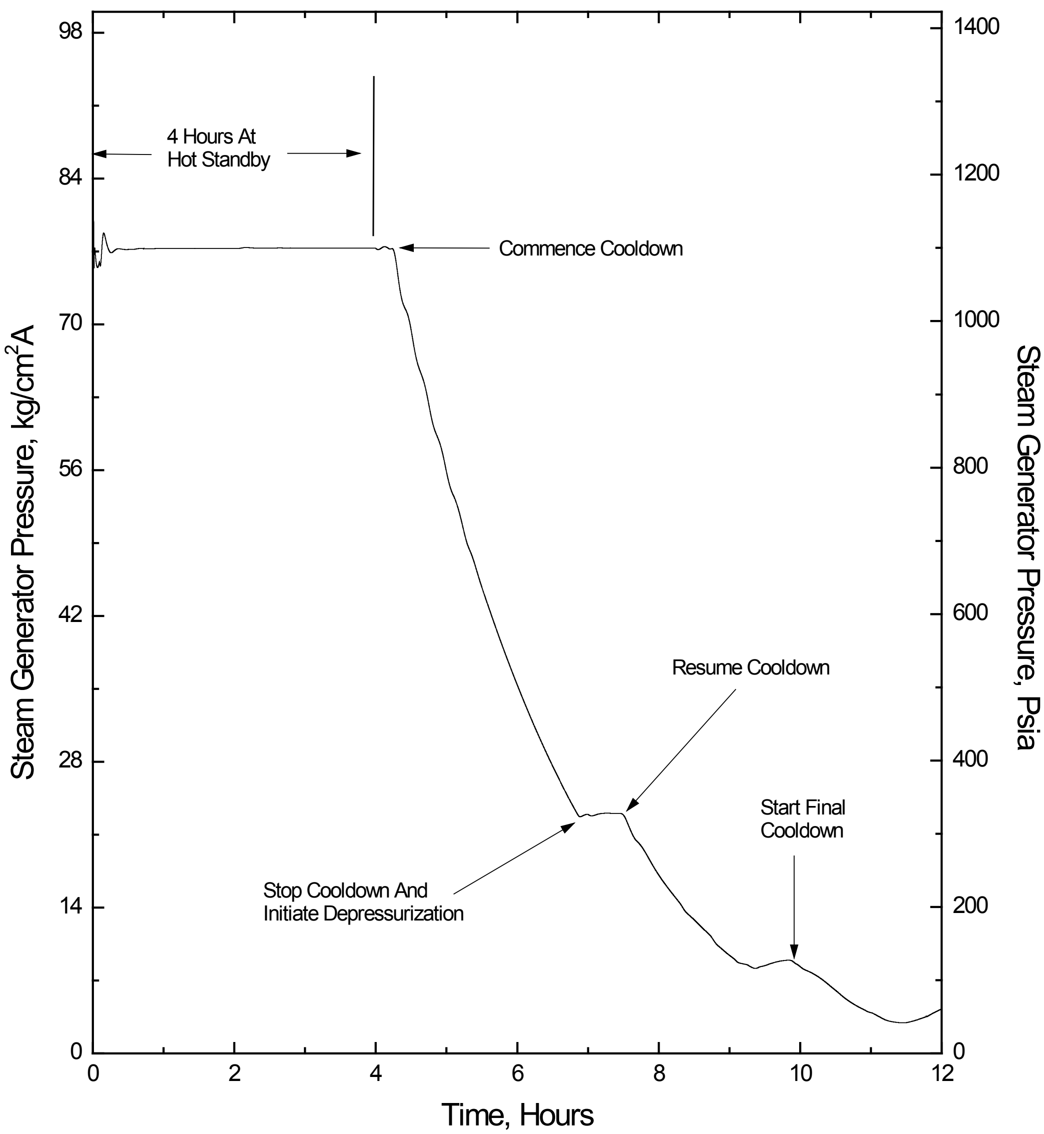


Figure 5.4.7-7 Steam Generator Pressure vs. Time for NCC Transient

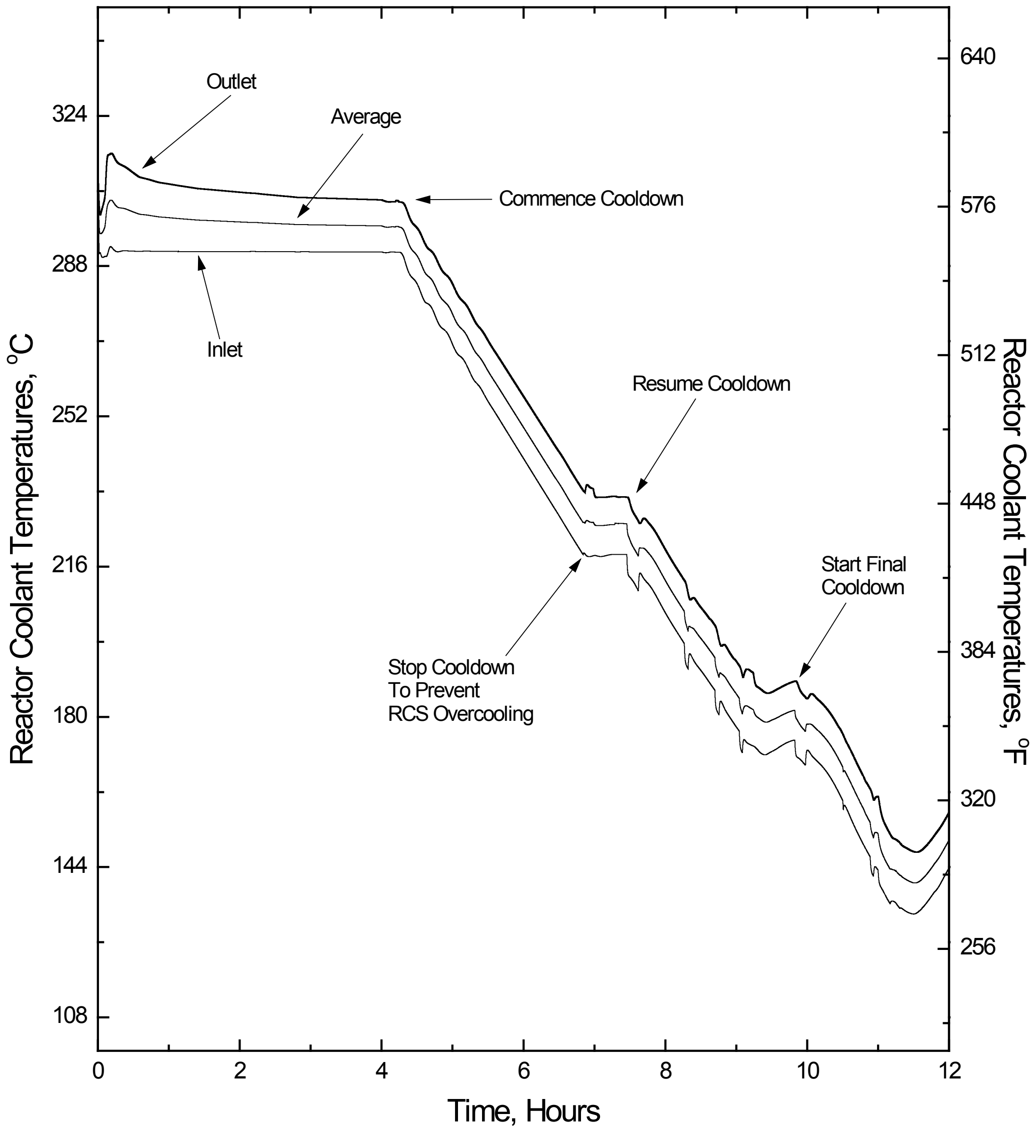


Figure 5.4.7-8 Reactor Coolant Temperatures vs. Time for NCC Transient

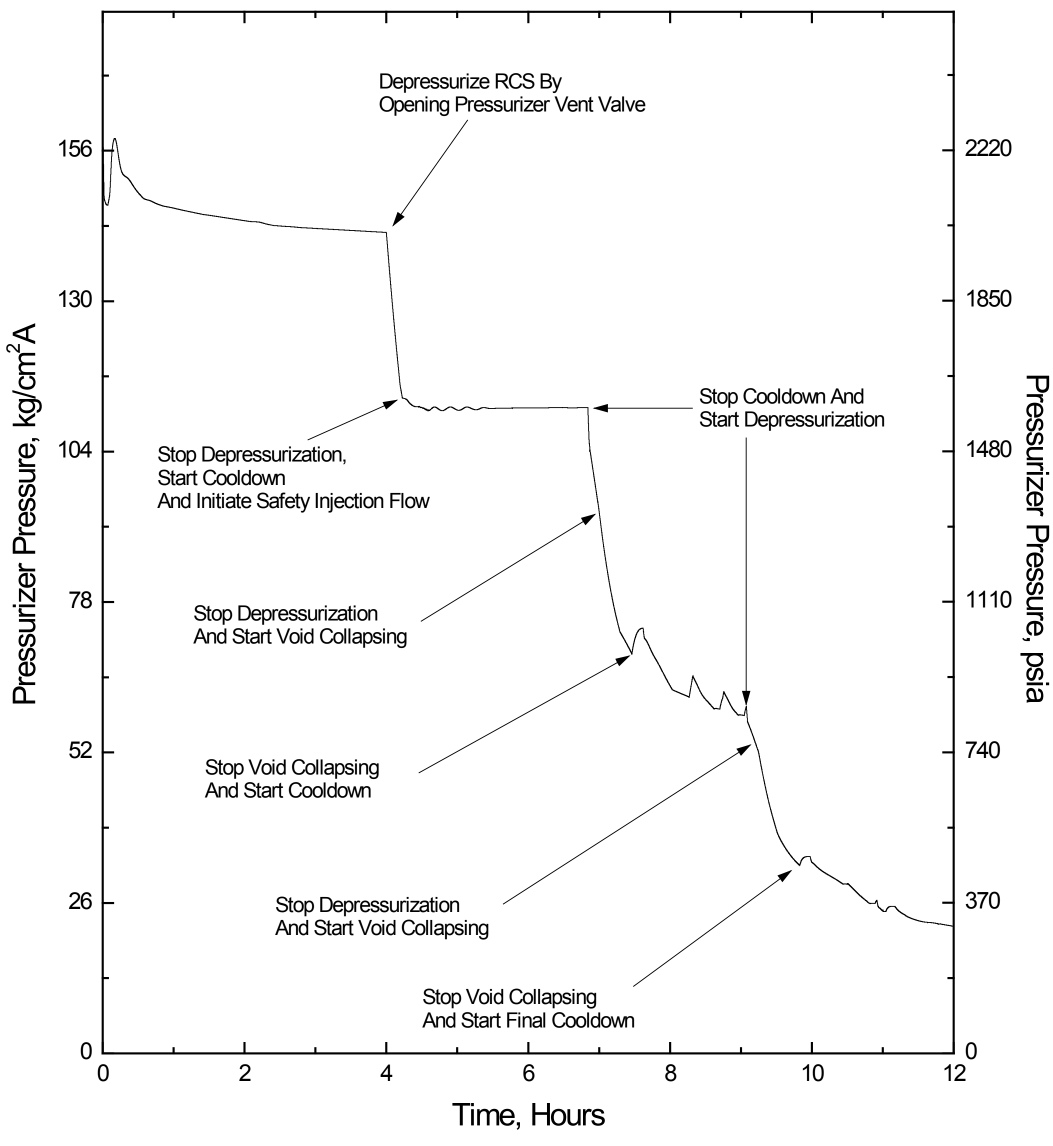


Figure 5.4.7-9 Pressurize Pressure vs. Time for NCC Transient

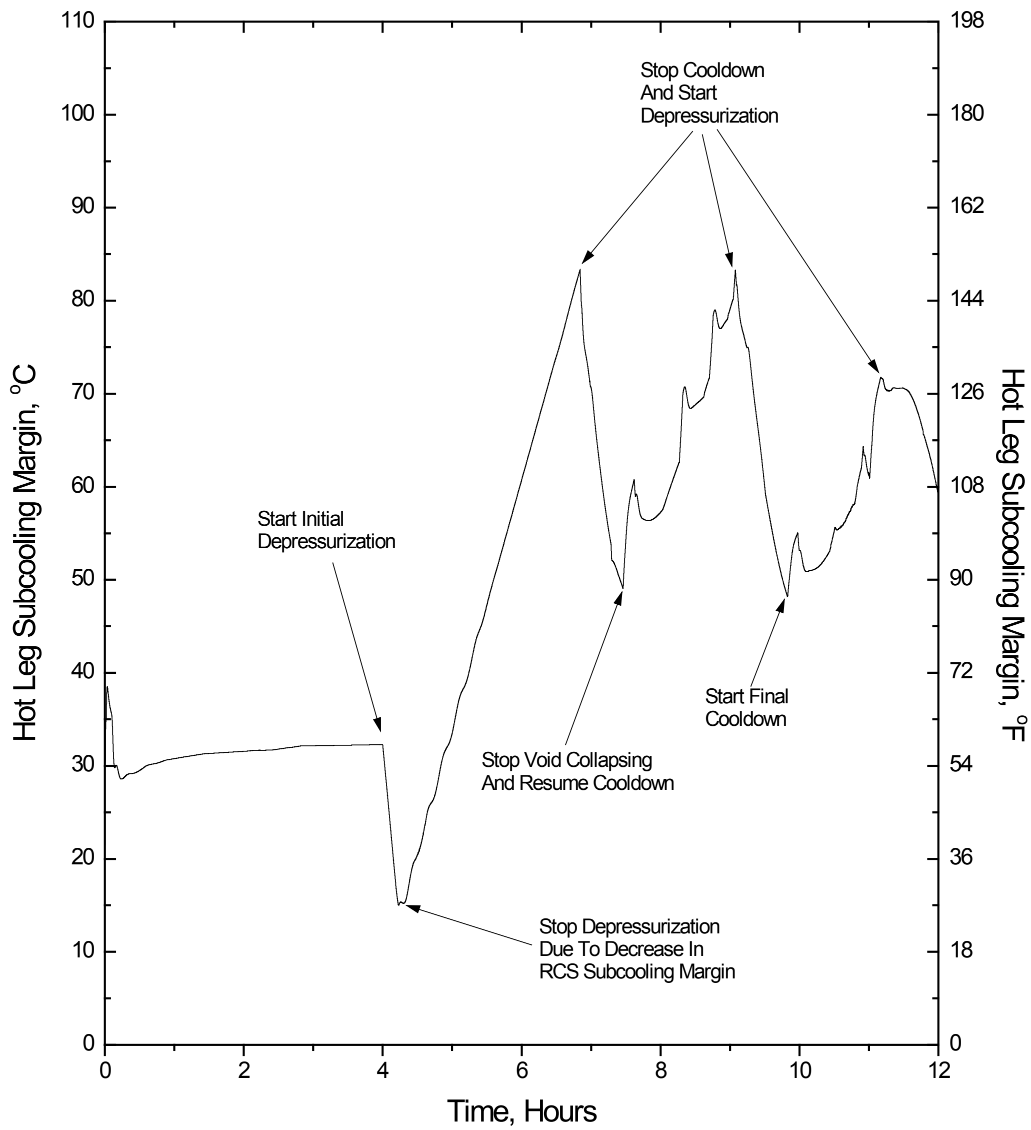


Figure 5.4.7-10 Hot Leg Subcooling Margin vs. Time for NCC Transient

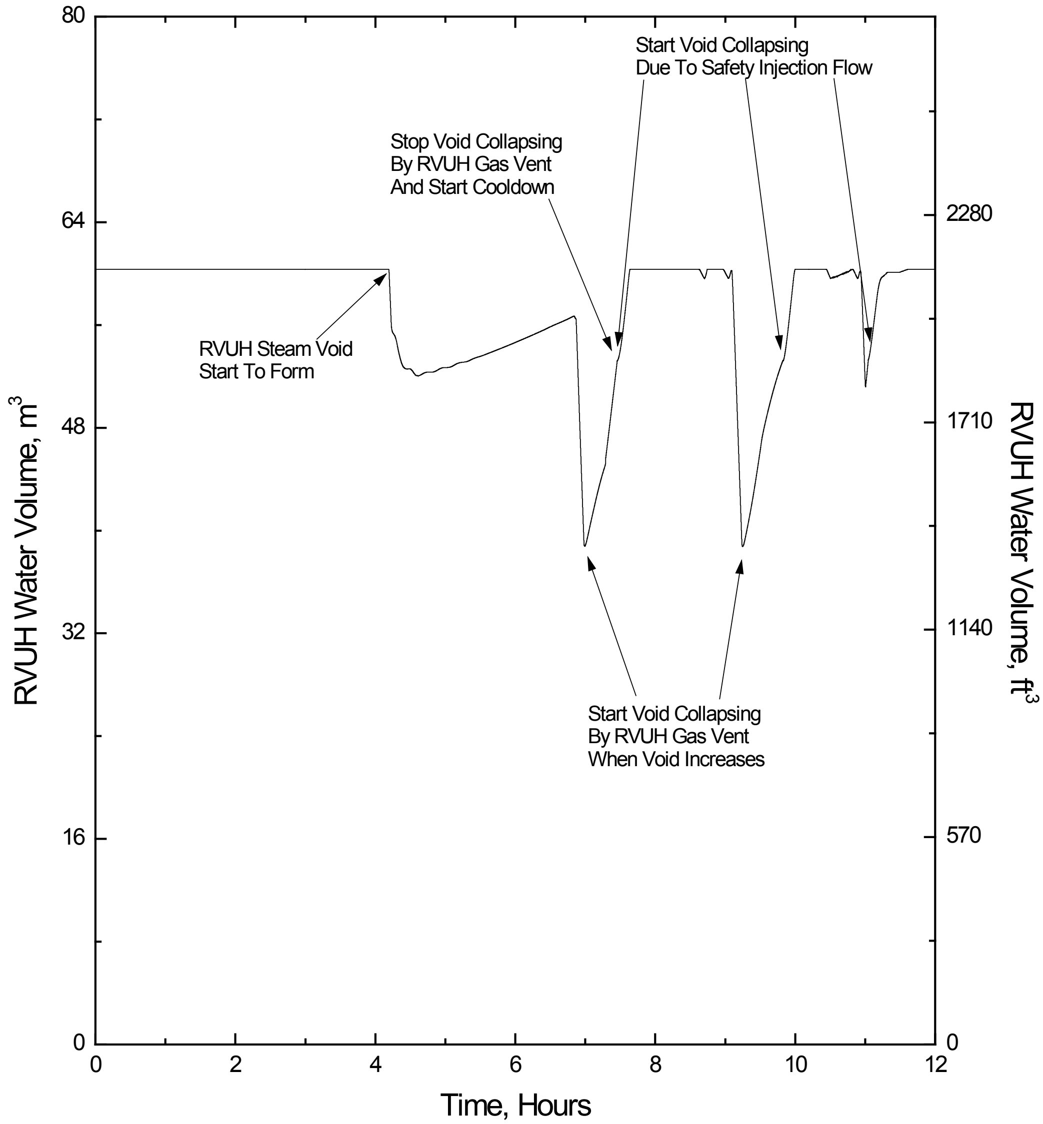


Figure 5.4.7-11 RVUH Water Volume vs. Time for NCC Transient

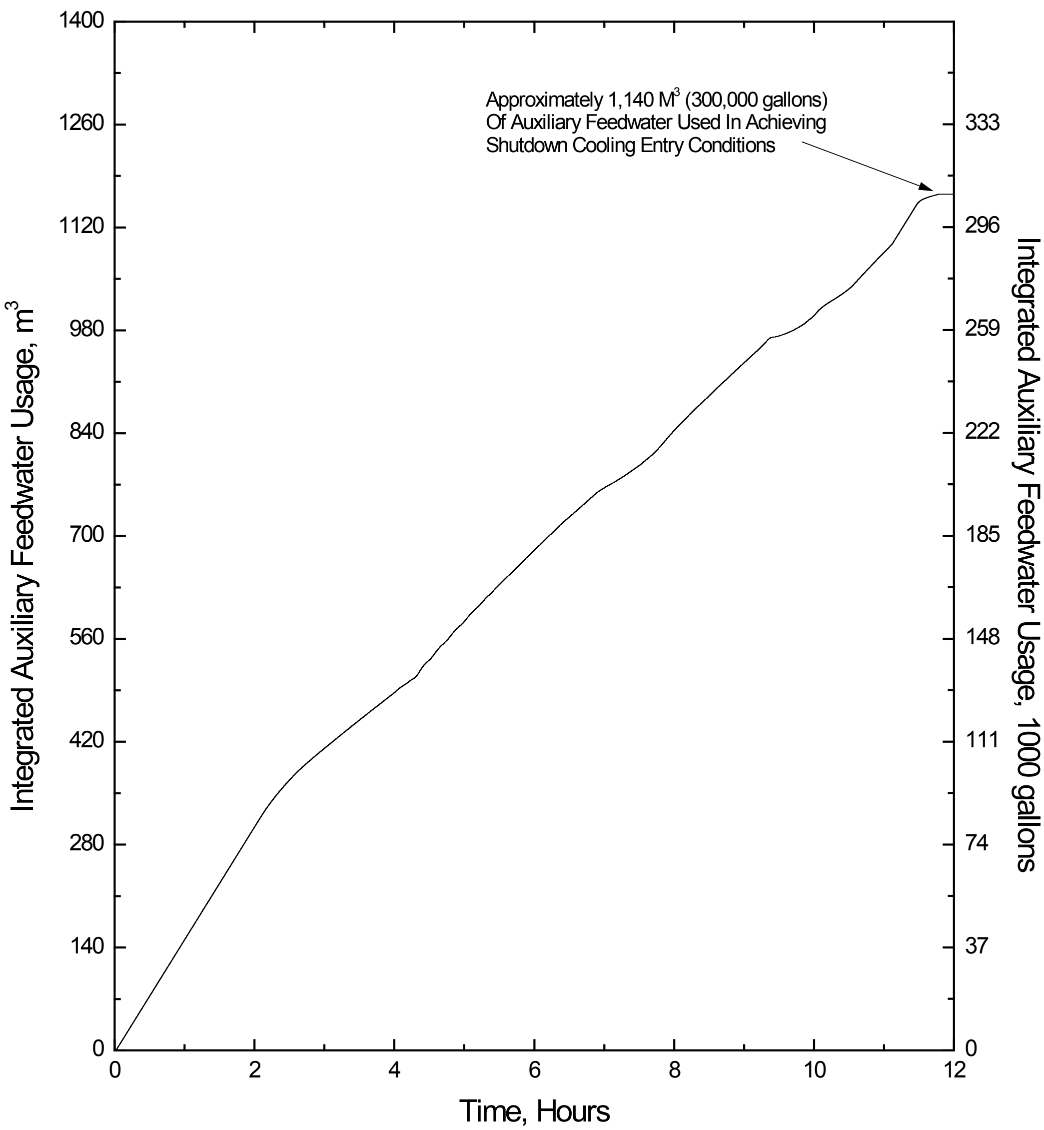
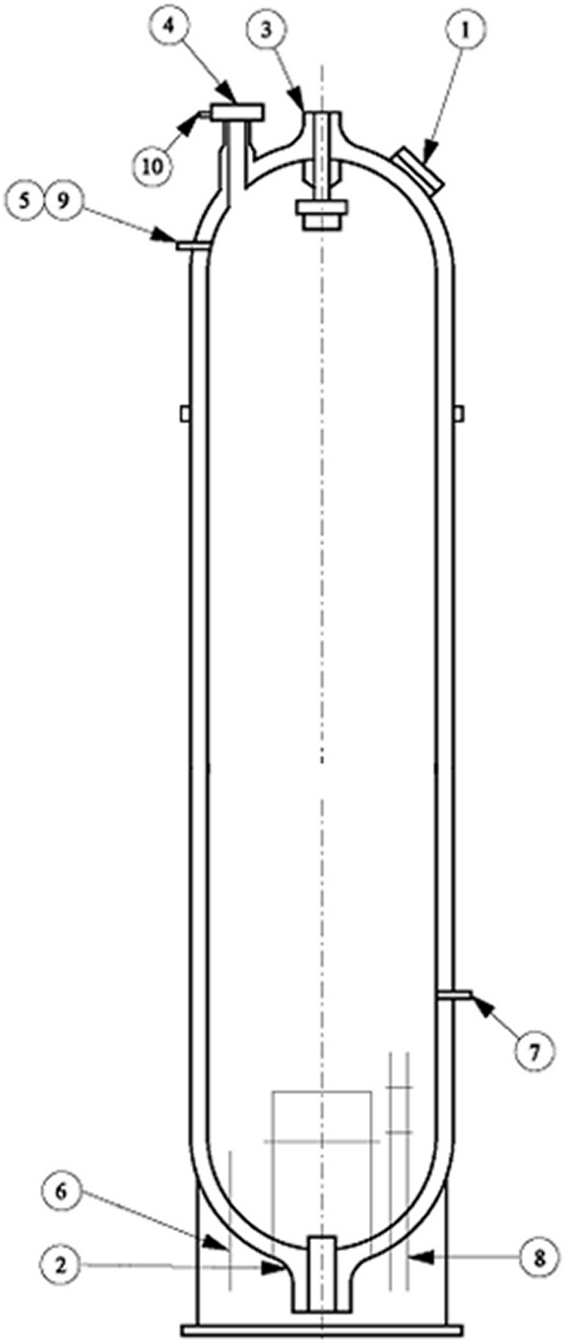


Figure 5.4.7-12 Integrated Auxiliary Feedwater Usage vs. Time for NCC Transient



NOZZLE SCHEDULE		
NO.	SERVICE	NO. REQUIRED
1	MANWAY	1
2	SURGE	1
3	SPRAY	1
4	POSRV	4
5	LEVEL-UPPER	3
6	LEVEL-LOWER	2
7	TEMPERATURE	2
8	HEATER	48
9	PRESSURE	4
10	RCGVS	4

Figure 5.4.10-1 Pressurizer

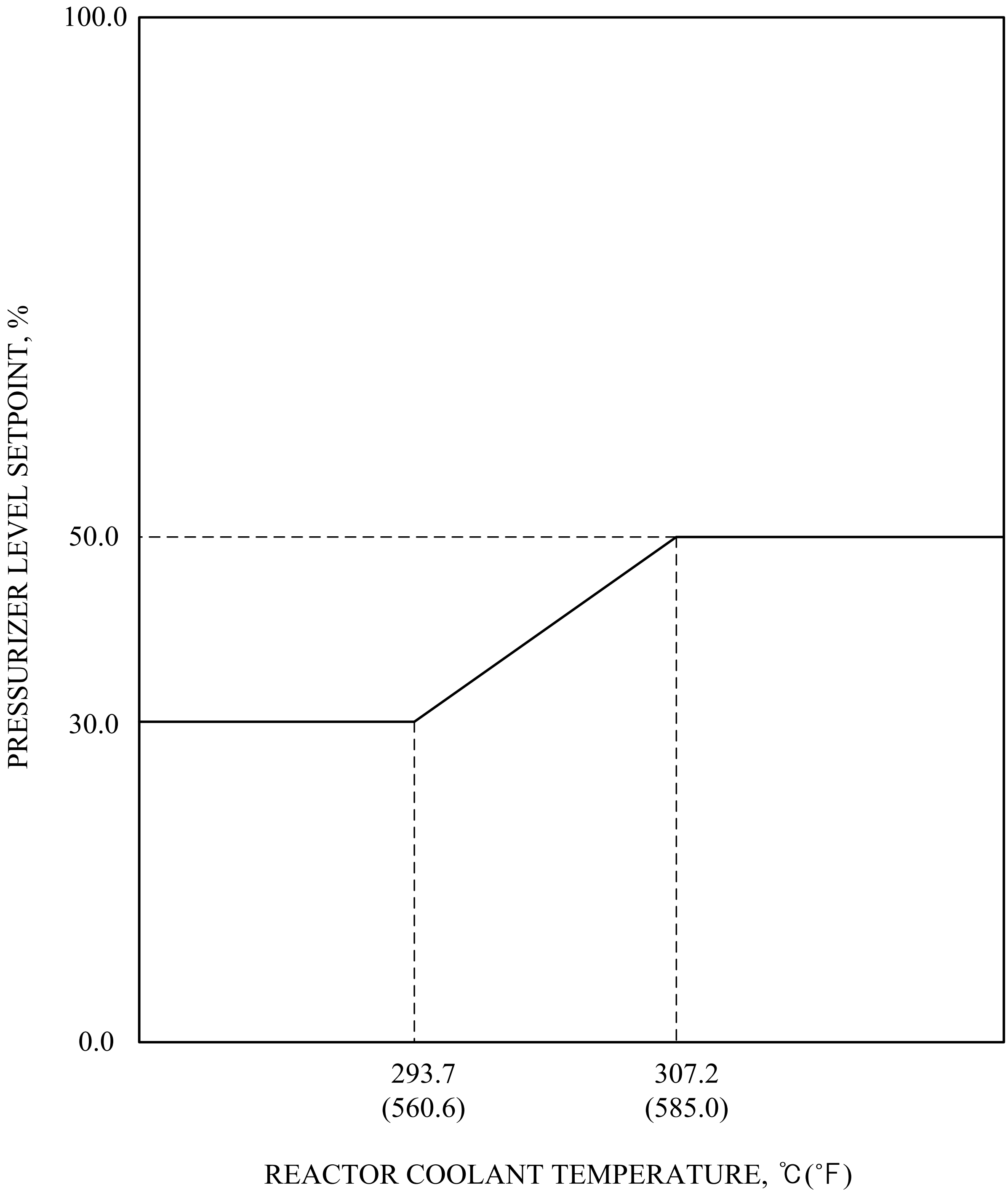


Figure 5.4.10-2 Pressurizer Level Setpoint Program

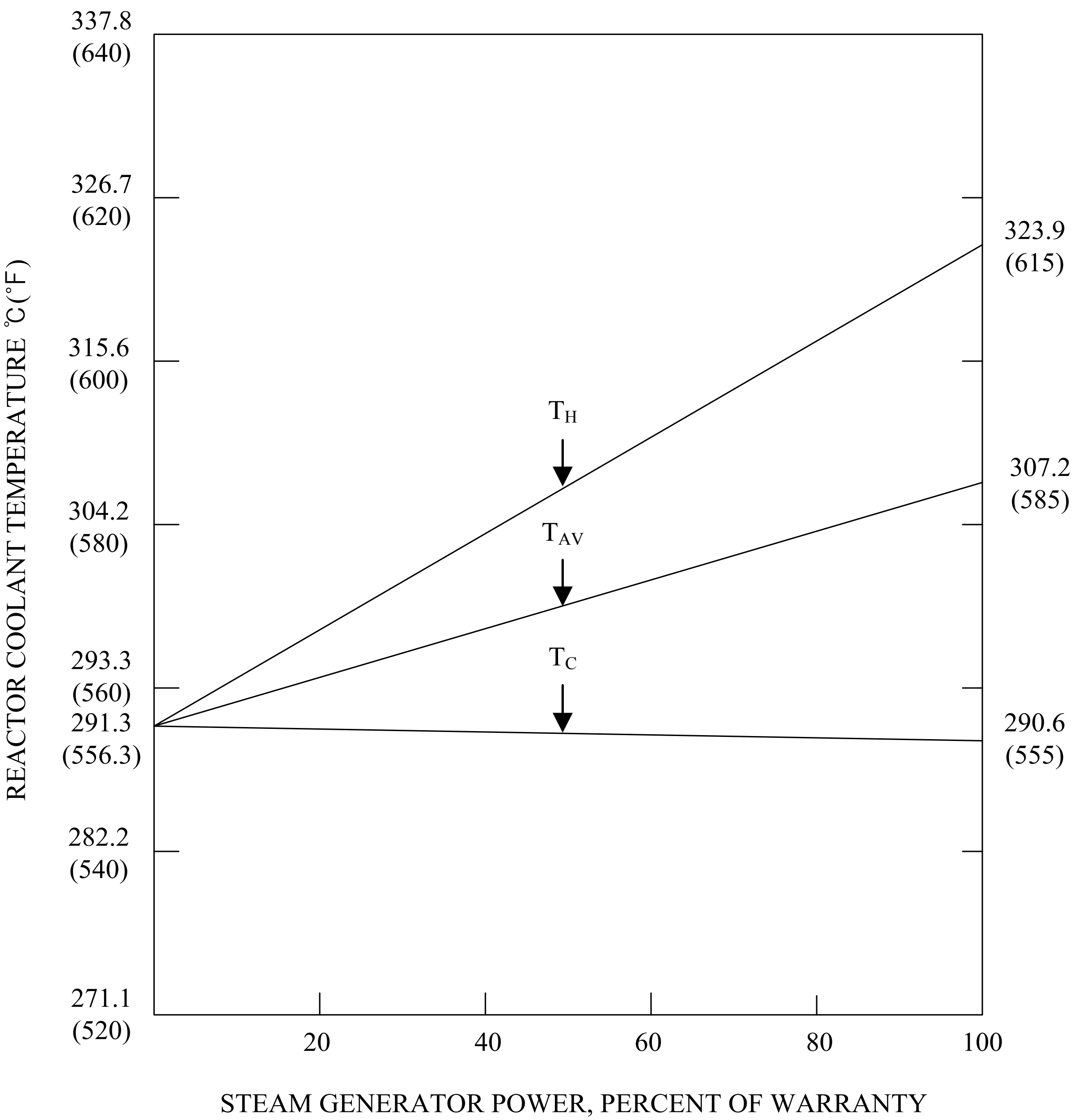


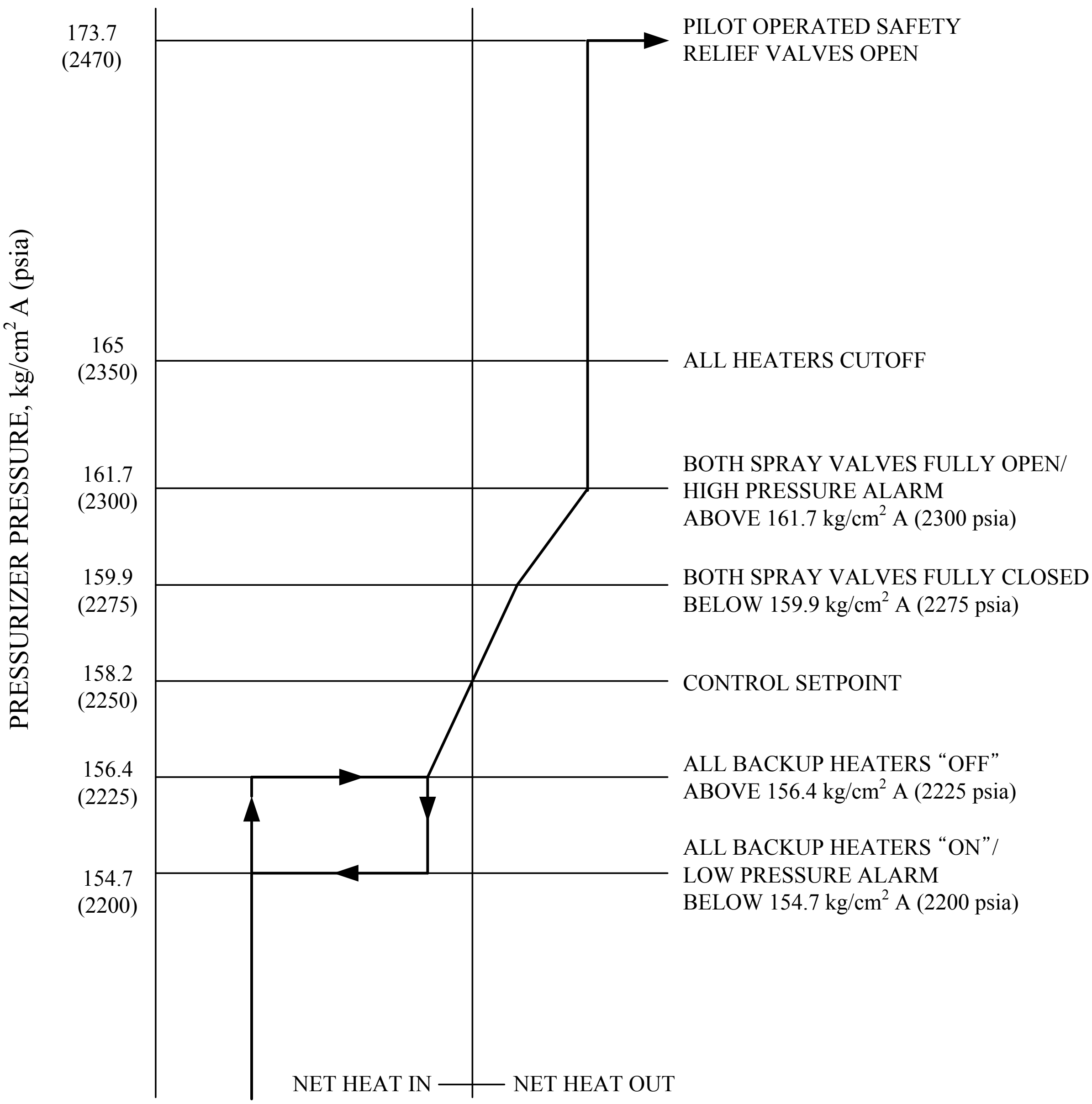
Figure 5.4.10-3   Temperature Control Program

Pressurizer Level Error, mm (in)	Action	
+1,244.6 (+49)	High-level error alarm	↑
+1,168.4 (+46)	Letdown orifice isolation value (CH-110Z) open	↑
+1,117.6 (+44)	Clear high-level error alarm	↓
+431.8 (+17)	Letdown orifice isolation valve (CH-110Z) close Energize backup heaters	↓ ↑
+355.6 (+14)	Backup heaters off	↓
-355.6 (-14)	Clear low-level error alarm	↑
-482.6 (-19)	Low-level error alarm	↓
-1,955.8 (-77)	Letdown orifice isolation valve (CH-110Y) open	↑
-3,251.2 (-128)	Letdown orifice isolation valve (CH-110Y) close	↓

↑: Level increasing relative to setpoint

↓: Level decreasing relative to setpoint

Figure 5.4.10-4 Pressurizer Level Error Program



Note : The proportional heater controls pressurizer pressure using Proportional-Integral controller.

Figure 5.4.10-5 Pressurizer Pressure Control Program

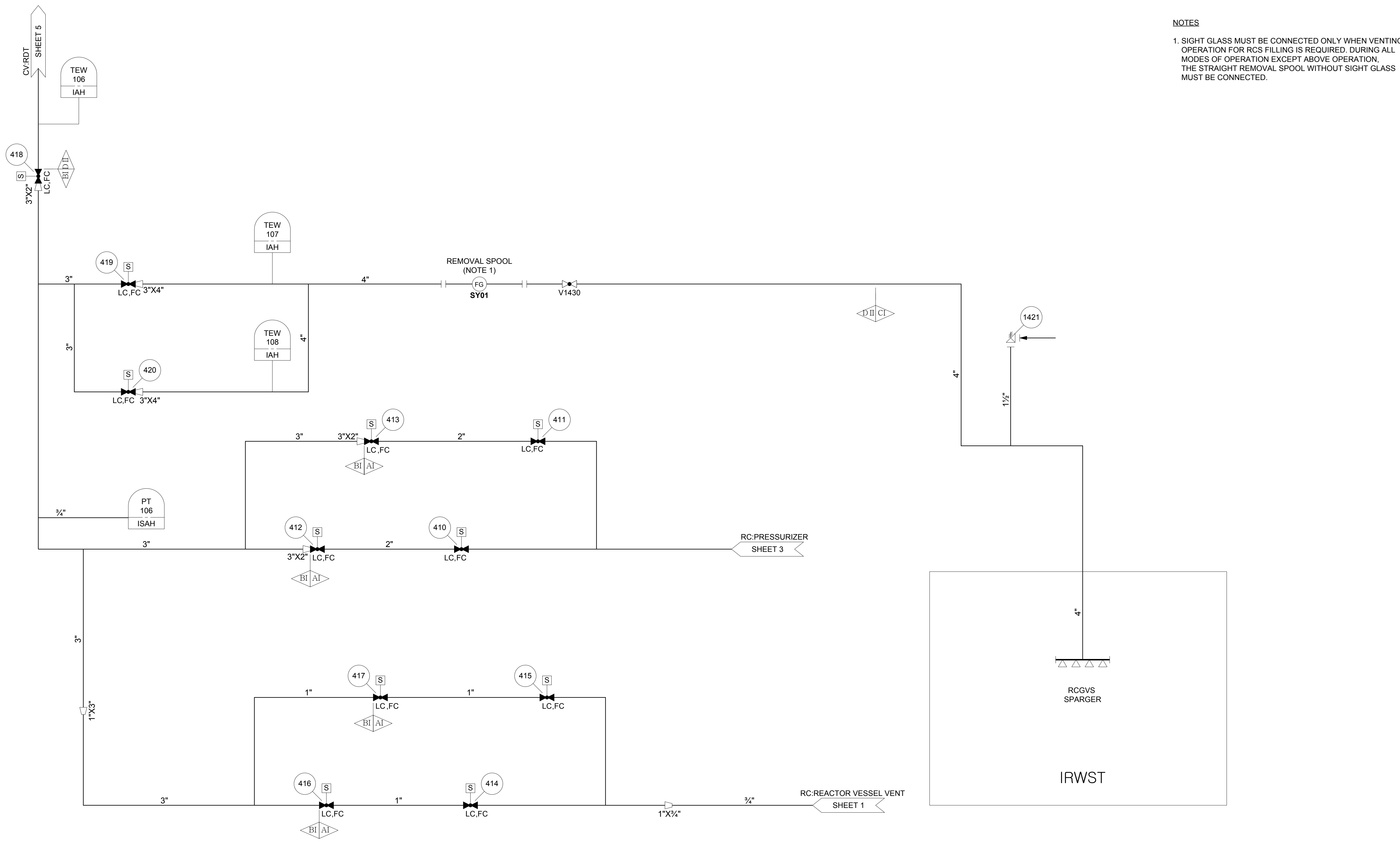
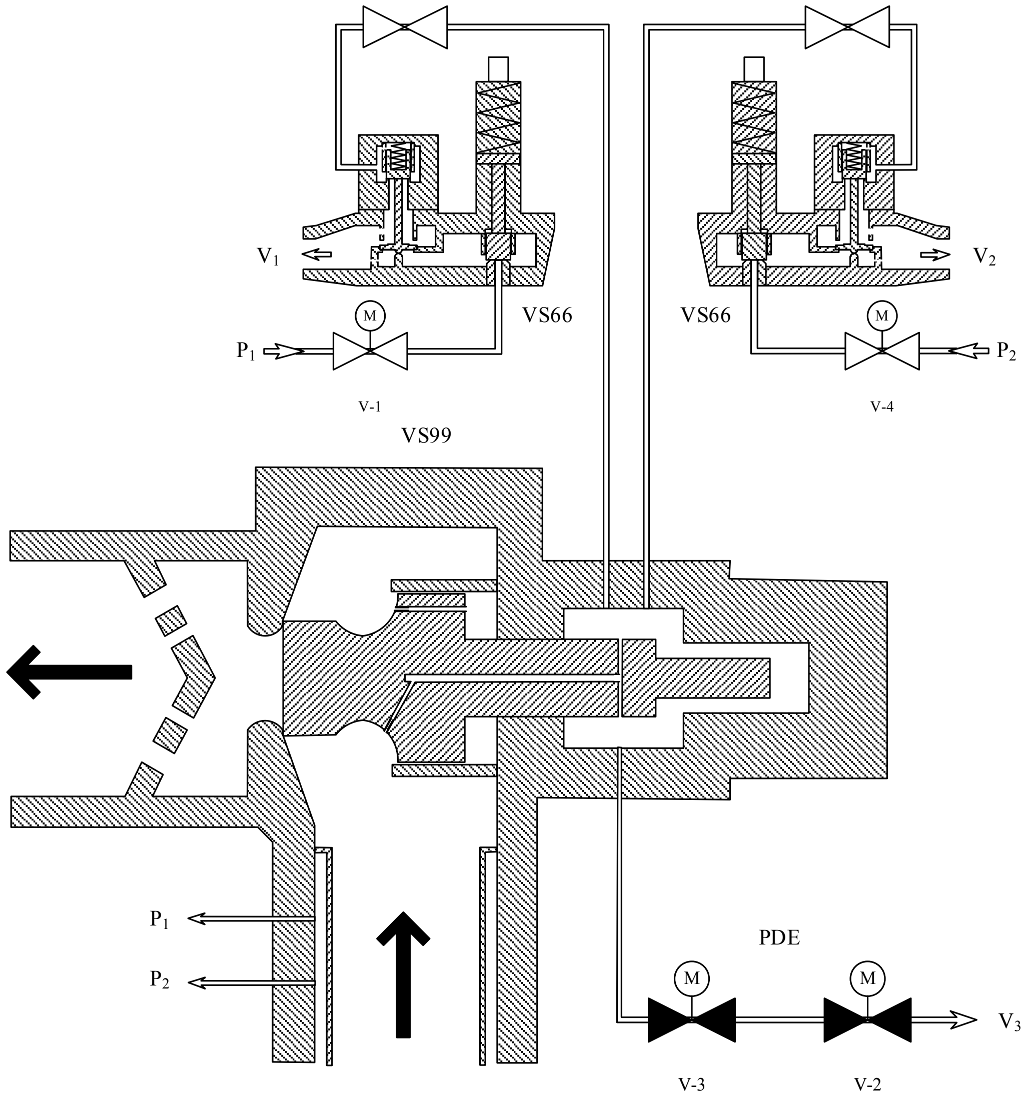


Figure 5.4.12-1 Reactor Coolant Gas Vent System Flow Diagram



VS99 : Main Valve
VS66 : Spring Loaded Pilot Valve
PDE : Motor Operated Pilot Valves
M : Motor Operated Valve
V_i : Pilot Discharge
P_i : Impulse Line

Figure 5.4.14-1 Pilot Operated Safety Relief Valve Schematic Diagram

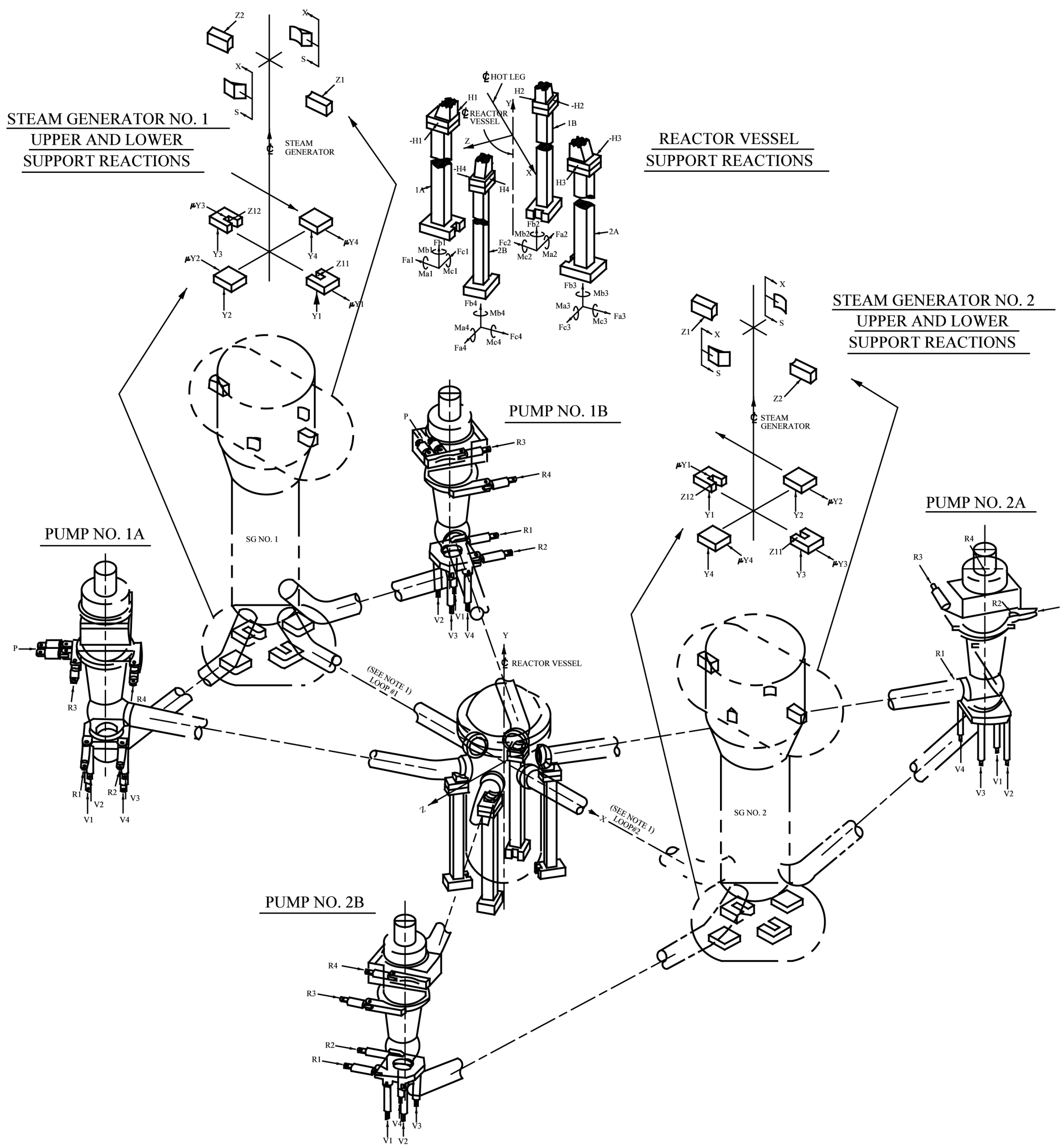


Figure 5.4.15-1 Reactor Coolant System Arrangement and Support Points

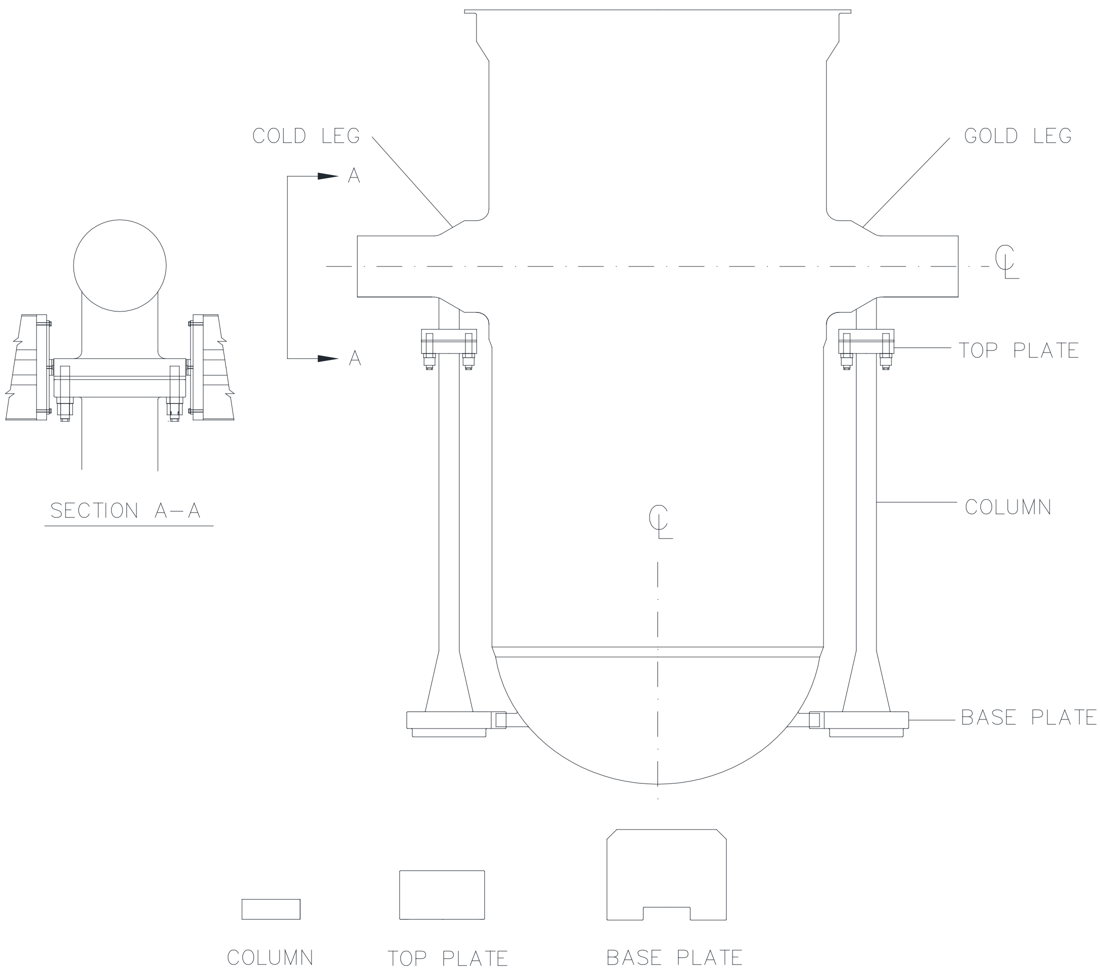


Figure 5.4.15-2 Reactor Vessel Support

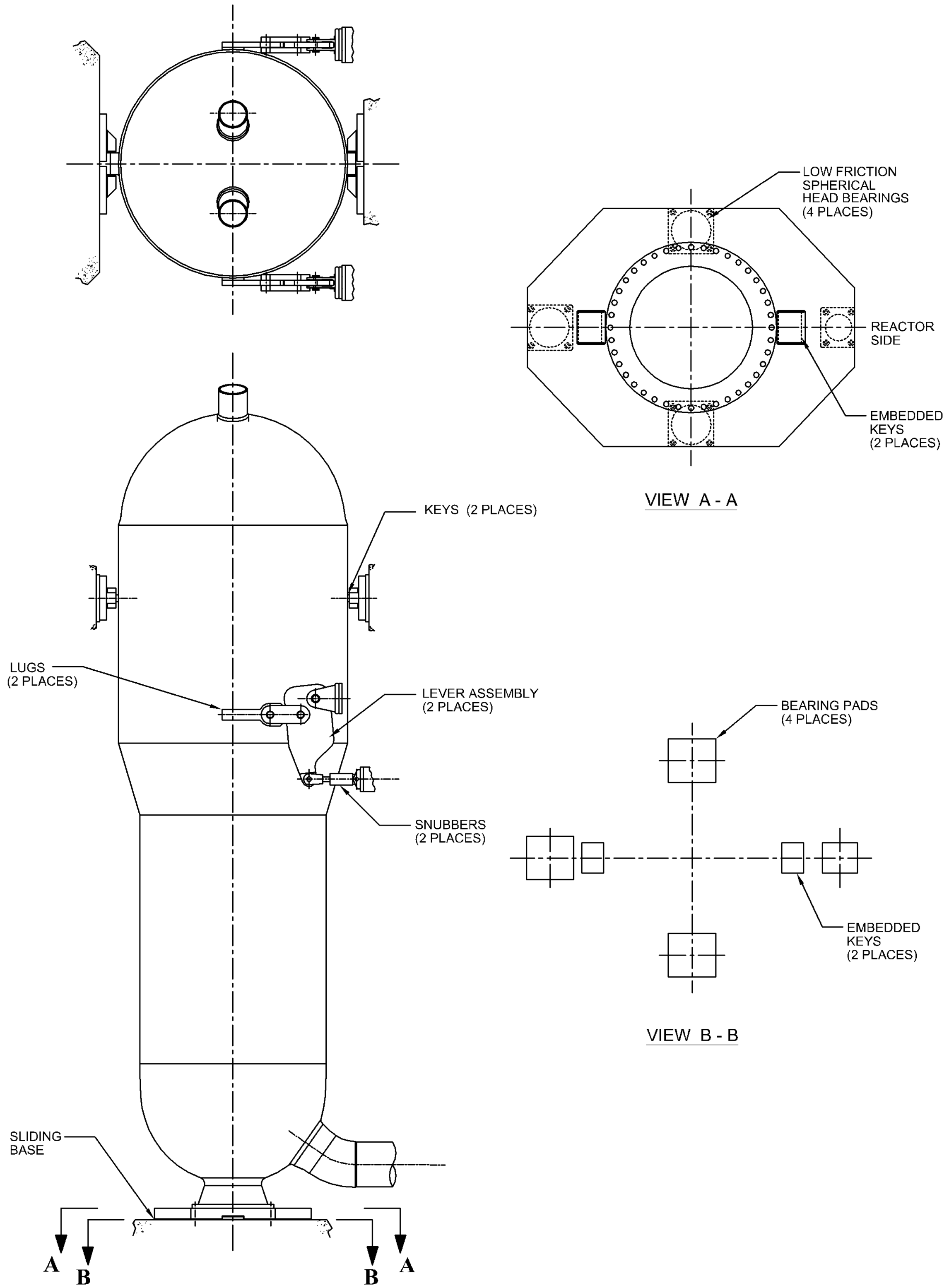


Figure 5.4.15-3 Steam Generator Supports

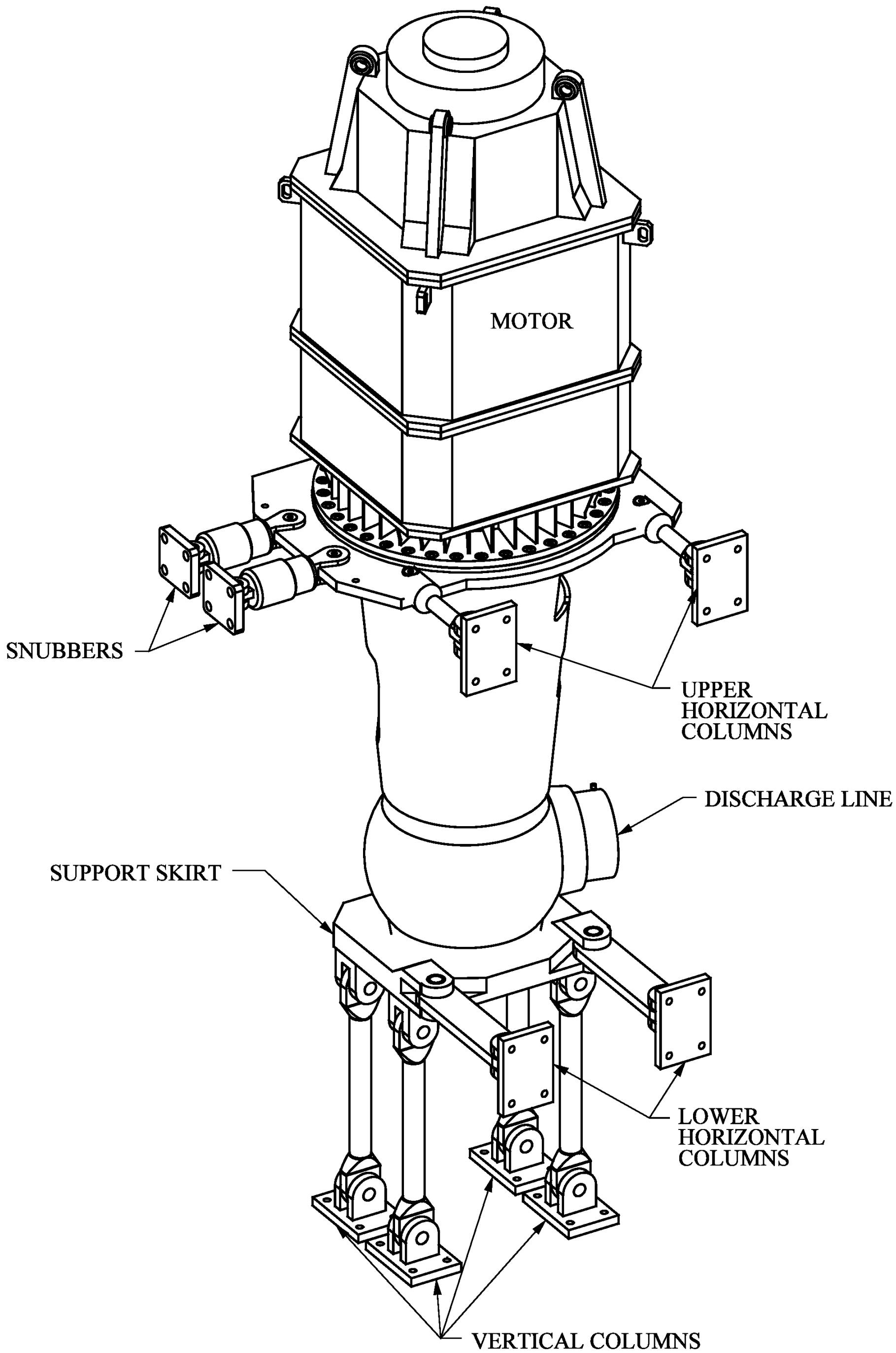


Figure 5.4.15-4 Reactor Coolant Pump Supports

APPENDIX 5A

EVALUATION OF THE APR1400 DESIGN AND INTERSYSTEM LOSS-OF-COOLANT ACCIDENT CHALLENGES

APPENDIX 5A – EVALUATION OF THE APR1400 DESIGN AND INTERSYSTEM LOSS-OF-COOLANT ACCIDENT CHALLENGES

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5A.1 Introduction

5A.1.1 Background

Studies conducted to identify the vulnerability of pressurized water reactor (PWR) plants to intersystem loss-of-coolant accidents (ISLOCAs) have concluded that ISLOCAs could cause core damage substantially more often than probabilistic safety assessments (PSAs) from some PWR plants have indicated. According to U.S. Nuclear Regulatory Commission (NRC) Information Notice 92-36 (Reference 1), these PSAs are typically limited to modeling ISLOCA sequences that include only a catastrophic failure of check valves that isolate the reactor coolant system (RCS) from low-pressure systems and give little consideration to human errors that could lead to an ISLOCA. The PSAs also include little consideration of harsh environments such as flooding that could be caused by an ISLOCA and their effects on plant equipment and recovery activities.

The results of the studies suggest that ISLOCA precursors are most likely human error and procedural deficiencies and are attributed to a lack of awareness of the possibility or consequences of an ISLOCA.

The NRC has developed a position on design requirements to minimize the potential for ISLOCAs (NRC Information Notice 92-36 (Reference 1); NRC Letter (Reference 2); SECY-90-016 (Reference 4); NUREG/CR-5102 (Reference 7)). The APR1400 design has been evaluated, and several design improvements necessary to conform with the NRC position have been identified. The evaluation and improvements are described in this appendix.

5A.1.2 Objectives

The objectives of this appendix are to (1) evaluate the design of systems that interface with the RCS, (2) provide reasonable assurance that the systems would retain structural integrity in an ISLOCA, and (3) provide reasonable assurance that any leakage would be a small fraction of the makeup system capability and that the consequential offsite dose would be a small fraction of that allowed in 10 CFR 100.11.

The evaluation goes beyond the traditional approach of assessing the ability of the systems that interface with the RCS to withstand an overpressurization event. Instead, the approach is to assume that the system design incorporates all of the requirements necessary to satisfy the General Design Criteria and then to consider the exposure of a low-pressure system to full RCS pressure.

5A.1.3 Scope

The scope of the evaluation is applicable to the APR1400 design and the systems and subsystems that interface directly or indirectly with the RCS and that are susceptible to ISLOCA challenges.

5A.2 Conclusions

The improvements to the APR1400 design that facilitate accommodating an ISLOCA and that are described in this appendix include a 63.3 kg/cm^{2}G (900 psig) design of several portions of low-pressure systems, additional pressure instrumentation to provide warning of a high-pressure condition, and relocation of the refueling water storage tank inside containment. The APR1400 design, including these improvements, conforms with all NRC requirements concerning ISLOCA events.

The improvements made to the APR1400 design resulting from this ISLOCA evaluation include:

a. Increasing the design pressure rating of equipment or systems

b. Adding equipment and instrumentation that alert the operator to an ISLOCA challenge or terminate and limit the scope of an ISLOCA event

These improvements are made together with the following:

a. Capability to leak test pressure isolation valves

b. Pressure isolation valve position indications and controls in the main control room (MCR)

c. High pressure alarms to warn the operator when rising pressure approaches the design pressure of low-pressure systems to provide reasonable assurance of conformance with the applicable NRC requirements

Table 5A-1 summarizes the design improvements made to the APR1400 design as a result of the ISLOCA evaluation.

5A.3 Design Requirements and Approach

5A.3.1 ISLOCA Definition

An ISLOCA is defined in NRC Information Notice 92-36 as “a class of accidents in which a break occurs in a system connected to the reactor coolant system (RCS), causing a loss of the primary system inventory.” An ISLOCA is interpreted as a beyond design basis event for systems that are connected directly or indirectly to the RCS. The pressurization pathway is established by an inadvertent opening of a valve or valves, a failure of containment isolation, or the postulation that valves are fully open (e.g., check valves). This interpretation is believed to address all sources that may challenge low-pressure systems. Based on this definition of ISLOCA, evaluations were performed to assess the ability of the APR1400 design to withstand an overpressurization event.

5A.3.2 ISLOCA Acceptance Criteria

The design responses to ISLOCA challenges described in this appendix are evaluated against acceptance criteria consistent with the guidance in NRC Information Notice 92-36.

The systems that are susceptible to an ISLOCA are to be designed so that the following conditions are satisfied:

a. The system retains its structural integrity throughout the event. Structural integrity is preserved if the system maintains its pressure boundary despite distortion and/or loss of function.

b. Any leakage caused by the event is limited to the makeup system capabilities.

c. Offsite doses are limited to a small fraction of the limit specified in 10 CFR 100.11, as assumed in the design bases for the Chapter 15 analyses.

5A.3.3 Conformance Methods

The design responses to ISLOCA challenges described in this appendix consist of:

a. Increasing the design pressure rating of equipment or systems

   Systems or subsystems susceptible to an ISLOCA are designed to a pressure of at least 40 percent (63.3 kg/cm^{2}G (900 psig)) of the RCS normal operating pressure. Austenitic stainless steel piping uses a minimum wall thickness corresponding to the standard weight for sizes less than 406.4 mm (16 in) nominal pipe size (NPS) and schedule 40 for 406.4 mm (16 in) NPS and larger sizes (NUREG/CR-5603, Reference 3).

b Incorporating the following design features that alert the operator to an ISLOCA challenge or terminate and limit the scope of the ISLOCA event:

   1) Locating the systems or subsystems susceptible to an ISLOCA completely within containment

   2) Isolating the systems or subsystems from the RCS during conditions when the RCS pressure exceeds its design pressure

   3) Providing pressure relief to limit the pressurization to within the design capabilities of the systems or subsystems

5A.3.4 ISLOCA Evaluation Process

The APR1400 was designed by reflecting the traditional design bases events. The evaluation of ISLOCA events performed for this appendix and the subsequent determination of the appropriate design responses consisted of the following:

a. The RCS flow diagram described in Chapter 5 was reviewed to identify all systems or subsystems that interface with the RCS. All systems and subsystems within containment or designed to full RCS operating pressure were noted to meet the acceptance criteria. The systems or subsystems located outside containment and designed for less than RCS pressure were evaluated to determine the pressurization pathways from the RCS. The pathways represent all of the possible ways to pressurize the low-pressure systems connected to the RCS.

b. The pressurization pathways were further evaluated to determine the impact of the interface of systems or subsystems with the initiating pressurization pathway. The pressurization pathways are illustrated in Figures 5A-1 to 5A-9. The figures show a pyramid structure beginning at the top with a system or subsystem that is directly connected with the RCS. A given pyramid represents a class of potential ISLOCA events characterized by a finite set of pressurization pathways. The systems that could be pressurized if all of the interfacing system valves are postulated to be open are identified in the pressurization pathways.

c. The pressurization pathways were analyzed to identify any pattern that could suggest a hierarchy of design responses to prevent or mitigate an ISLOCA event. Any hierarchical pattern would be motivated by the desire to satisfy the ISLOCA acceptance criteria with a design response commensurate with the perceived benefit and without degradation of safety.

d. Each interface in the pressurization pathways was analyzed to identify the type and location of design response to satisfy the ISLOCA acceptance criteria.

A review of the pressurization pathways in Figures 5A-1 to 5A-9 led to the following observations:

a. The following systems or subsystems are directly connected to the RCS for one or more ISLOCA events (i.e., are in one or more pressurization pathways), and all ISLOCA events (pressurization pathways) begin with these systems or subsystems:

    1) Safety injection system (SIS)

    2) Shutdown cooling system (SCS)

    3) Chemical and volume control system (CVCS)

4) Sampling system (SS)

b. The following nine classes of pressurization pathways were identified:

1) Shutdown cooling line

2) Safety injection delivery line

3) Letdown line

4) Charging line

5) RCP seal injection

6) RCP controlled bleedoff

7) Sampling – hot leg

8) Sampling – pressurizer surge line

9) Sampling – pressurizer steam space

c. The containment spray system (CSS) is not directly connected to the RCS for the ISLOCA events that are identified. For more information, see Subsection 5A.4.3.

d. A given system can occupy several levels in the interface hierarchy for any ISLOCA event.

d. The design response applied to a system should be commensurate with the highest interface level (i.e., the highest position in the pyramid) that a system may have for any ISLOCA event. The rationale is that there is a correlation between a system's position in the hierarchy and the scope of the event (number of interfacing systems affected), which is one measure of consequence of the event.

e. Increasing the design pressure rating of a system high in a pyramid, such as a design response to ISLOCA precursors, does not protect connected subsystems lower in the pyramid. These subsystems still require a design response of their own. For example, designing the SCS to 40 percent of normal RCS operating pressure does not address the protection of the systems that interface with the SCS against ISLOCAs.

f. The application of a design response that terminates and limits the scope of the event to a system high in the pyramid can protect subsystems lower in the pyramid. For example, isolation of the letdown flow terminates the event and protects downstream portions of the CVCS as well as interconnecting systems (e.g., SS, waste management system). Such downstream effects, which are characteristic of some design responses, are important considerations in selecting an appropriate design response.

5A.4 Design Responses

This section presents an evaluation of the systems and subsystems that are directly connected to the RCS during some mode of operation, which are identified in Subsection 5A.3.4 as the classes of pressurization pathways in relation to an ISLOCA. Any ISLOCA event begins with one of the systems or subsystems. The systems and subsystems are as follows:

a. Shutdown cooling line (Subsection 5A.4.1)

b. Safety injection delivery line (Subsection 5A.4.2)

c. Letdown line (Subsection 5A.4.4)

d. Charging line (Subsection 5A.4.5)

e. RCP seal injection (Subsection 5A.4.6)

f. RCP controlled bleedoff (Subsection 5A.4.7)

g. Sampling – hot leg (Subsection 5A.4.8.1)

h. Sampling – pressurizer surge line (Subsection 5A.4.8.2)

i. Sampling – pressurizer steam space (Subsection 5A.4.8.3)

The evaluation of each system and subsystem that is presented in this section is structured as follows:

a. Description of the interface and related system under evaluation to provide orientation to the system and the components affected by the design response

b. Design evaluation – description of the mechanical and electrical features of the design that have been incorporated to resolve the ISLOCA challenge; schematic drawing of the features; and description of any impact from the design response on the PSA, emergency operation guidelines (EOGs), or configuration management

5A.4.1 Shutdown Cooling System

5A.4.1.1 Description of Interface

The SCS supply line is directly connected to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the hot leg and out of containment through the containment isolation valves to the low-pressure sections of the SCS. Once pressurized, other low-pressure systems are susceptible.

Figure 5A-1 shows the various pathways an ISLOCA in this class can take and the interfacing systems it would affect, assuming all downstream interfacing valves are open. The SCS interface with the atmosphere is described in Subsection 5A.4.9.5. The interface with the SIS, CSS, and sampling system (SS) is described in Subsections 5A.4.2, 5A.4.3, and 5A.4.8, respectively.

The SCS return line is directly connected to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the direct vessel injection (DVI) nozzles and out of containment through the containment isolation valves to the low-pressure sections of the SCS. Once pressurized, other low-pressure systems are susceptible to damage from the high-pressure condition.

Figure 5A-1 shows the various pathways an ISLOCA in this class can take and the interfacing systems it would affect assuming all downstream interfacing valves are open. The SCS interface with the atmosphere is described in Subsection 5A.4.9.5. The interfaces with the SIS and SS are described in Subsections 5A.4.2 and 5A.4.8, respectively. The CVCS, SS, and SIS have a direct interface with the RCS, and the enhancements satisfy the indirect interface through the SCS.

5A.4.1.2 Design Evaluation

The design pressure for most of the SCS supply and return lines is equal to or greater than 63.3 kg/cm^{2}G (900 psig). This design response satisfies the ISLOCA acceptance criteria (i.e., at least 40 percent of normal RCS pressure or 63.3 kg/cm^{2}G [900 psig]). Figure 5A-10 shows the design responses for this class of ISLOCA. This is in accordance with the NRC's position presented in the NRC Letter (Reference 2).

The design response to an ISLOCA has no impact on the PSA or EOGs.

5A.4.2 Safety Injection System

5A.4.2.1 Description of Interface

The SIS injection line is directly connected to the reactor vessel and is a primary interface through which an ISLOCA can begin. Pressurization is postulated from the DVI nozzle and out of containment through the containment isolation valves to the low-pressure sections of the system. Once pressurized, other low-pressure systems are susceptible.

Figure 5A-2 shows the various pathways an ISLOCA in this class can take and the interfacing systems it would affect assuming all downstream interfacing valves are open. The SIS interfaces with the CSS, SS, and atmosphere are described in Subsections 5A.4.3, 5A.4.8, and 5A.4.9.5, respectively. The interface with the SCS is described in Subsection 5A.4.1; the SCS has a direct interface with the RCS and the enhancements that have been made satisfy the indirect interface through the SIS.

5A.4.2.2 Design Evaluation

The design pressure for most of the SIS is 144.1 kg/cm^{2}G (2,050 psig), which satisfies the ISLOCA acceptance criteria (i.e., at least 40 percent of normal RCS pressure or 63.3 kg/cm^{2}G [900 psig]). The design pressures of all other sections outside of containment not designed to 144.1 kg/cm^{2}G (2,050 psig) are designed to 63.3 kg/cm^{2}G (900 psig) including the sections from the containment to the SIS pump suctions. Figure 5A-11 shows the design response for this class of ISLOCAs.

These design responses protect the SIS lines and all interfacing systems from an ISLOCA challenge without adversely affecting performance or operations. These design responses satisfy the ISLOCA acceptance criteria for the SIS lines since all sections of the system and interfaces are designed to at least 40 percent of the RCS operating pressure criteria in accordance with the NRC position presented in the NRC Letter.

The design response to an ISLOCA has no impact on the PSA or EOGs.

5A.4.3 Containment Spray System

5A.4.3.1 Description of Interface

The CSS is not directly connected to the RCS during the modes of operation in which an ISLOCA challenge can occur.

Technical Specifications require the CSS to be aligned for the containment spray function during Modes 1, 2, 3, and 4. This requires the CSS pumps to be aligned for suction from the IRWST and discharge to the containment spray headers. With this alignment, the CSS does not interface directly with the RCS, but there is an indirect interface through the SCS.

The shutdown cooling pumps (SCPs) can be aligned to perform containment spray for limited periods of time. With this alignment, the CSS still does not interface directly with the RCS since the SCPs are aligned for suction from the IRWST and discharge to the containment spray headers.

When shutdown cooling operations are performed, the containment spray pumps (CSPs) may be substituted for SCPs. This requires the CSPs to be realigned for suction from the RCS hot legs (through the SCS) and discharge to the DVI nozzles (again, through the SCS). While the CSPs are directly connected to the RCS during this configuration, ISLOCAs are not postulated since the RCS pressure is less than the design pressure of the CSS during this operation mode.

For purposes of this ISLOCA evaluation, the CSS is not directly connected to the RCS. Figures 5A-1 through 5A-9 show the various pathways an ISLOCA can progress to reach the CSS and the interfacing systems it would affect, assuming all downstream interfacing valves are open.

5A.4.3.2 Design Evaluation

These design responses protect the CSS and all interfacing systems from an ISLOCA challenge without adversely affecting performance or operations. These design responses satisfy the ISLOCA acceptance criteria for the CSS since all sections of the system and interfaces are designed to at least 40 percent of RCS operating pressure criteria in accordance with the NRC position presented in the NRC Letter.

Installation of a spool piece in the refueling pool fill line provides physical separation of the low-pressure system from any pressurization sources in the CSS and thereby satisfies the ISLOCA acceptance criteria.

Figure 5A-12 shows the design response for this class of ISLOCAs.

5A.4.4 Letdown Line

5A.4.4.1 Description of Interface

The letdown line is directly connected to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the letdown nozzle, through the regenerative heat exchanger (HX), letdown orifices, letdown HX, containment isolation valves, and letdown control valves to the low-pressure sections of the system out of containment. Once pressurized, other low-pressure interfacing systems are susceptible. Figure 5A-3 shows the various pressurization pathways that this class of ISLOCAs can take and the systems it would affect assuming all downstream interfacing valves are open.

5A.4.4.2 Design Evaluation

The letdown line design was evaluated to satisfy the ISLOCA acceptance criteria for the APR1400 design. Figure 5A-13 shows the design response for this class of ISLOCAs. An additional high-pressure alarm has been provided to the letdown line and is located downstream of the letdown control valves. The alarm provides information to the operator that operating pressure is exceeding the design pressure. Thus, the control room operator can isolate the letdown line to prevent any further pressure communication downstream of the containment isolation valve.

This design response satisfies the ISLOCA acceptance criteria because:

a. System integrity is preserved since the portion of the letdown line pressurized by the ISLOCA is designed to full RCS design pressure.

b. The portion of the letdown line downstream of the letdown control valves and all other downstream interfacing systems are protected from an ISLOCA by the operator's manual action and therefore do not have an integrity challenge.

c. Primary coolant is not lost, and offsite doses do not change since system integrity is preserved.

The following design response to this event was also evaluated: increasing the design pressure of the entire CVCS to 40 percent of full RCS operating pressure (i.e., 63.3 kg/cm^{2}G (900 psig)) and necessarily all systems that interface with the CVCS downstream of the letdown control valves (see Table 5A-2). Although this design response satisfies the ISLOCA acceptance criteria, it is not considered practical because of the increased complexity to the CVCS, the addition of new systems, the number of systems affected, and the reliance on new unproven systems, equipment, and fabrication techniques. In addition, these features are not perceived to be commensurate with the benefit, especially when compared with the other design responses that were also considered.

Increasing the design pressure of the low-pressure portion of the CVCS may initially appear to be desirable simply because no operator action is required to mitigate the event. However, there are many other low-pressure systems that interface with the CVCS (Table 5A-2) that could also be exposed to RCS pressure given the ground rules for postulating ISLOCA precursors. Thus, these additional systems would also need to be designed to a higher pressure to mitigate the effects of this event.

Table 5A-3 shows the low-pressure CVCS equipment**.** The magnitude of the effect on the equipment has been estimated based on NUREG/CR-5603, as follows:

a. Ion exchangers – These components, having a wall thickness of 8.0 mm (5/16 in), would be required to have their wall thicknesses increased by a factor of approximately 3 to 5.

b. Purification filters – These components, having a wall thickness of 4.8 mm (3/16 in) to 9.5 mm (3/8 in), would be required to have their wall thicknesses increased by a factor of 2.

c. Volume control tank – The wall thicknesses of the tanks would be increased by a factor of 10.

d. Field-fabricated tanks – The impact to very large field-fabricated tanks, such as the reactor makeup water tank (RMWT), with a volume of 1,495 m^3 (395,000 gal), is estimated to be even more significant. These tanks are fabricated in the field because of their size. Fabrication of high-pressure tanks of this size would constitute new technology since new fabrication techniques would have to be developed to accommodate the increased plate thickness, and a redesign of structural supports and basemats would be required to support the increased load.

The complexity of such equipment and systems is directly proportional to the effects of increasing design pressure, namely:

a. These kinds of design changes to commercially available equipment would require special designs that are not readily available.

b. Heavier components would require redesigning supporting structures such as pipe supports and building floors.

c. Heavier components would require a re-evaluation of the seismic analyses to account for the additional loads.

In addition, there are several low-pressure tanks in the CVCS (Table 5A-3) that are vented to the atmosphere and are used during various plant modes to process water for RCS operations. These vents prevent the pumps from drawing an excessive vacuum on the tank and subsequently damaging the tank or cavitating the pumps during draining and prevent the tanks from pressurizing during fill operations. If the design pressure of the subsystems in which these tanks are located were increased to 63.3 kg/cm^2G (900 psig), the tanks would have to be designed to the same pressure. As a result, the vents would have to be eliminated and a new supporting system designed to facilitate filling and draining operations. The new systems would in turn need to have a high-pressure design (at least 63.3 kg/cm^{2}G (900 psig)) and have a flow rate capacity sufficient to support the simultaneous operation of all of the tanks. Additionally, a failure to the new system would restrict the ability to control plant operations requiring RCS inventory and boron control.

These examples illustrate the impracticality of increasing the design pressure of the CVCS as a design response for the letdown line because:

a. New systems would have to be added

b. The complexity of the overall design would increase

c. Many additional systems would be affected (see Table 5A-2)

To satisfy the ISLOCA acceptance criteria, the enhanced design of the letdown line also satisfies the requirements of SECY-90-016 by providing:

a. High pressure alarm in the control room to warn the operators of the event

b. Containment isolation valve position indication and control in the MCR

c. Periodic leak testing of the containment isolation valves

The design response shown in Figure 5A-13 satisfies the ISLOCA acceptance criteria, defined by the NRC in the NRC Letter, and also accomplishes the following:

a. The operator limits the scope of the event by terminating the event at a location that prevents the rest of the CVCS and its interfacing systems from being pressurized.

b. The design response avoids the need to design and procure special equipment to withstand the high pressures characteristic of this event, some of which may not be considered available technology.

c. A pressure relief valve in the letdown line limits the pressurization to the design capability of this system.

Increasing the complexity of all the low-pressure systems interfacing with the CVCS downstream of the letdown control valve to mitigate an event such as an ISLOCA is impractical.

Based on these observations, the design response shown in Figure 5A-13 is considered the preferred approach for this class of ISLOCAs.

It was determined that additional Technical Specifications are not required and that the selected design response does not affect the PSA or EOGs.

5A.4.5 Charging Line

5A.4.5.1 Description of Interface

The charging line is directly connected to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the charging nozzle, through the shellside of the regenerative HX, charging control valve, and charging pump to the low-pressure sections of the system. Once pressurized, other low-pressure interfacing systems are susceptible. This process also assumes that the charging pump and the auxiliary charging pump (ACP) are not operating. Figure 5A-4 shows the various pathways the ISLOCAs in this class can take and the systems they would affect assuming all downstream interfacing valves are open. There are a total of three check valves in the charging line from the RCS nozzle to the charging pump discharge.

5A.4.5.2 Design Evaluation

The charging line design was evaluated to satisfy the ISLOCA acceptance criteria for the APR1400 design. Figure 5A-14 shows the installation of a pressure sensor to the charging line in the common suction line.

The pressure indicator provides information to the operator that the operating pressure is exceeding the design pressure.

This design response satisfies the ISLOCA acceptance criteria because:

a. System integrity is preserved since the portion of the charging line (charging pump discharge line) pressurized by the ISLOCA is designed to full RCS design pressure.

b. The portion of the charging line upstream of the charging pumps (including charging pump miniflow heat exchanger (MFHX) and all other upstream interfacing systems are protected from an ISLOCA by the operator's action and therefore do not have an integrity challenge.

c. Primary coolant is not lost, and offsite doses do not change since system integrity is preserved.

The justification for installing the pressure indicator follows the same rationale as presented in Subsection 5A.4.4.2 for the letdown line. Therefore, this section summarizes the justification and expands on items specific to the charging line.

The examples presented in Subsection 5A.4.4.2 illustrate the impracticality of increasing the design pressure of the CVCS as a design response.

The design of the charging line satisfies the requirements in the NRC Letter by providing:

a. A high-pressure alarm in the control room to warn the operators of the event

c. Containment isolation valve position indication and control in the control room

b. Periodic leak testing of the containment isolation valve

The design response shown in Figure 5A-14 satisfies the ISLOCA acceptance criteria in the NRC Letter and also accomplishes the following:

a. It limits the scope of the event by terminating the event at a location that prevents the rest of the CVCS and its interfacing systems from being pressurized.

b. It avoids the need to design and procure special equipment to withstand the high pressures characteristic of this event, some of which may not be considered available technology.

There is a difference between the letdown line and the charging line. In the letdown line, the postulated pressurization pathway is with the normal flow direction. This means that the letdown line has been designed to reduce and control the fluid pressure under normal and ISLOCA operating conditions. Additionally, a backup relief valve is provided. This relief valve has been sized to provide reasonable assurance that the design pressure in the low-pressure section is not exceeded.

The direction of the pressurization of the charging line, on the other hand, is opposite to the flow. The charging line has three check valves in series to prevent overpressurization of the CVCS. It is unlikely that three check valves would fail simultaneously with both the charging pump and ACP stopped. However, the pressure indicator upstream of the charging pumps is installed.

In addition, relief valves are provided in the volume control tank (VCT) discharge line, charging pump miniflow line, charging pump discharge line, and ACP discharge line.

Increasing the complexity of all of the low-pressure systems interfacing with the CVCS upstream of the charging pumps to mitigate an event such as an ISLOCA is therefore impractical. Based on these observations, the design response shown in Figure 5A-14 is considered the preferred approach for this class of ISLOCAs.

An evaluation was also conducted to assess the impact of this design change to the Technical Specifications, PSA, and EOGs. It was determined that additional Technical Specifications are not required and that the selected design response does not affect the PSA or EOGs.

5A.4.6 Reactor Coolant Pump Seal Injection Line

5A.4.6.1 Description of Interface

The seal injection line is directly connected to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated (1) from the charging nozzle through the shellside of the regenerative HX to the seal injection line and (2) from the RCP seals through the tubeside of the high-pressure seal cooler to the discharge side of the charging pump. Once pressurized, other low-pressure interfacing systems are susceptible. This process also assumes that the charging pump and the ACP are not operating. Figure 5A-5 shows the various pathways ISLOCAs in this class can take and the systems it would affect assuming all downstream interfacing valves are open. This section addresses the seal injection line and the interfaces to the CVCS. The seal injection interface with the atmosphere is described in Subsection 5A.4.9.5.

5A.4.6.2 Design Evaluation

The evaluation of the RCP seal injection line design resulted in a determination that no design improvements for an ISLOCA are required. The sections that are susceptible to an ISLOCA are the vent and drain lines from the filters. Figure 5A-15 shows the design response for this class of ISLOCAs.

The design pressure of the seal injection line is 212.7 kg/cm^{2}G (3,025 psig), which satisfies the ISLOCA acceptance criteria. However, the design pressure for the filter vent and drain lines is 1.4 kg/cm^{2}G (20 psig) in order to be compatible with the equipment drain header and the equipment drain tank (EDT). It is considered impractical and unnecessary to increase the design pressure of the EDT, as described in Subsection 5A.4.4.

The design response protects the interfacing systems from an ISLOCA without adversely affecting performance or operations. This design response satisfies the ISLOCA acceptance criteria for the seal injection line. It is considered acceptable since the flow rate would be low, would not cause a rapid fill of the EDT, and would allow sufficient time for operator action to terminate the event. The EDT is provided with pressure, level, and temperature instrumentation that indicates and alarms in the MCR. Thus, the design of the seal injection line satisfies the requirements in the NRC Letter by providing:

a. A high EDT pressure alarm in the control room to warn the operators of the event

b. Containment isolation valve position indication and control in the control room

c. Periodic leak testing of the containment isolation valves

In addition, the failure of the vent and drain lines is immediately detected by loss or a reduction of the RCP seal injection flow because of flow diversion through the vent or drain line. The leak tightness of the filter vent and drain valves is confirmed during normal operation because seal injection pressure is normally 185.0 kg/cm^2 G (2,630 psig).

5A.4.7 Reactor Coolant Pump Controlled Bleedoff Line

5A.4.7.1 Description of Interface

The controlled bleedoff line is directly connected to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the RCP seals through a flow-limiting orifice and a flow meter out of containment to the charging pump MFHX or VCT. Once pressurized, other low-pressure interfacing systems are susceptible. Figure 5A-6 shows the various pathways ISLOCAs in this class can take and the systems it would affect assuming all downstream interfacing valves are open. This section addresses the controlled bleedoff line and the interfaces to the CVCS. The controlled bleedoff with the atmosphere is described in Subsection 5A.4.9.5.

5A.4.7.2 Design Evaluation Change

The evaluation of the RCP controlled bleedoff line resulted in a determination that no design improvements are required for an ISLOCA. Figure 5A-16 shows the design response for this class of ISLOCAs. The design provides a pressure rating of 174.7 kg/cm^2G (2,485 psig) to the outermost section of the system that could be pressurized.

The design pressure for the controlled bleedoff system is divided into high- and low-pressure sections. The high pressure, 174.7 kg/cm^2G (2,485 psig), exceeds the minimum design pressure requirement for an ISLOCA, as provided in SECY-90-016. The low-pressure section outside containment is designed to 5.3 kg/cm^2G (75 psig) and 14.1 kg/cm^2G (200 psig) to be compatible with the design pressure of the VCT and charging pump MFHX, respectively.

The low-pressure section of the RCP controlled bleedoff line meets the ISLOCA requirements since there is no plausible scenario in which this line could be pressurized above its design pressure. The upstream pressure is limited by a fixed-resistance flow control orifice, and the only means of isolating the flow in the low-pressure section requires isolation of the VCT and charging pump MFHX inlet, which are already protected from overpressurization by relief valves. The relief valves discharge to the EDT. The flow through the controlled bleedoff line during normal operation is 48.5 L/min (12.8 gpm), which is within the makeup capability and can be collected in the reactor drain tank (RDT) for more than 30 minutes before requiring operator action. Furthermore, the VCT and EDT are provided with an indication and alarm for liquid level, temperature, and pressure. The indication and alarm serve to alert the operator in conjunction with the indication of the RCP controlled bleedoff line pressure of any pressurization occurrence.

5A.4.8 Sampling System

5A.4.8.1 Hot Leg Sampling

5A.4.8.1.1 Description of Interface

The SS has a direct connection to the RCS hot leg, which results in a primary interface through which an ISLOCA can begin. Pressurization is postulated from the sampling nozzle on the hot leg through the containment isolation valves and out of containment to the low-pressure sections of the system. Figure 5A-7 shows the various pathways an ISLOCA in this class can take and the interfacing systems it would affect. All other interfaces are resolved by resolving the direct interface to the RCS.

5A.4.8.1.2 Design Evaluation

The design of the hot leg sampling line provides a fixed-resistance orifice and a small line size that limits the flow and pressure during all modes of plant operation. This line discharges to the VCT, which is already protected from overpressurization by a relief valve. The relief valve discharges to the EDT.

The only means of isolating the flow in the low-pressure section requires isolating the discharge to the VCT and the recycle drain header and opening the containment isolation valves. For this event, a relief valve has been added to provide protection for the system. The relief valve is sized to pass a flow rate that is equivalent to the hot leg sample line flow rate prior to sample collection. With this flow rate, the flow control orifice upstream of the pressure relief valve creates a pressure drop that limits the pressure in the system to an acceptable value. This flow rate is limited to within makeup capability. Furthermore, the EDT is provided with indication and alarm for liquid level, temperature, and pressure to alert the operator, in conjunction with the indication of the containment isolation valve (CIV) position, with sufficient time for operator action to terminate the event. Figure 5A-17 shows the design response for this class of ISLOCAs.

There is no impact to the SS as a result of this design response. Leakage from the relief valve is detected in the EDT and appropriate operator action is taken to terminate the event. In addition, none of the changes affect the PSA or EOGs for this event.

5A.4.8.2 Pressurizer Surge Line Sampling

5A.4.8.2.1 Description of Interface

The SS has a direct connection to the pressurizer surge line, which results in a primary interface through which an ISLOCA can begin. Pressurization is postulated from the sampling nozzle on the surge line through the containment isolation valves and out of containment to the low-pressure sections of the system. Figure 5A-8 shows the various pathways an ISLOCA in this class can take and the interfacing systems it would affect. This subsection addresses the pressurizer surge line interface. By resolving the direct interface to the RCS, all other interfaces are resolved.

5A.4.8.2.2 Design Evaluation

The design of the SS for the pressurizer surge line provides a fixed-resistance orifice and a small line size that limits the flow and pressure during all modes of plant operation. This line discharges to the VCT, which is already protected from overpressurization by a relief valve. The relief valve discharges to the EDT.

The only means of isolating the flow in the low-pressure section requires isolating the discharge to the VCT and the recycle drain header and opening the containment isolation valves. For this event, a relief valve has been added to provide protection for the system. The relief valve is sized to pass a flow rate that is equivalent to the pressurizer surge line flow rate prior to sample collection. With this flow rate, the flow control orifice upstream of the pressure relief valve creates a pressure drop that limits the pressure in the system to an acceptable value. This flow rate is limited to within makeup capability. Furthermore, the EDT is provided with indication and alarm for liquid level, temperature, and pressure to alert the operator, in conjunction with the indication of the CIV position, with sufficient time for operator action to terminate the event. Figure 5A-18 shows the design response for this class of ISLOCAs.

There is no impact to the SS as a result of this design response. Leakage from the relief valve is detected in the EDT, and appropriate operator action is taken to terminate the event. In addition, none of the changes affect the PSA or EOGs for this event.

5A.4.8.3 Pressurizer Steam Space Sampling

5A.4.8.3.1 Description of Interface

The SS has a direct connection to the pressurizer steam space, which results in a primary interface through which an ISLOCA can begin. Pressurization is postulated from the sampling nozzle on the pressurizer steam space through the containment isolation valves and out of containment to the low-pressure sections of the system. Figure 5A-9 shows the various pathways an ISLOCA in this class can take and the interfacing systems it would affect. This section addresses the pressurizer steam space interface. By resolving the direct interface to the RCS, all other interfaces are resolved.

5A.4.8.3.2 Design Evaluation

The design of the sampling system for the pressurizer steam space provides a fixed-resistance orifice and a small line size that limits the flow and pressure during all modes of plant operation. This line discharges to the VCT, which is already protected from overpressurization by a relief valve. The relief valve discharges to the EDT.

The only means of isolating the flow in the low-pressure section requires isolating the discharge to the VCT and the recycle drain header and opening of the containment isolation valves. For this event, a relief valve has been added to provide protection for the system. The relief valve is sized to pass a flow rate that is equivalent to the pressurizer steam space sample line flow rate prior to sample collection. With this flow rate, the flow control orifice upstream of the pressure relief valve creates a pressure drop that limits the pressure in the system to an acceptable value. This flow rate is limited to within makeup capability. Furthermore, the EDT is provided with indication and alarm for liquid level, temperature, and pressure to alert the operator, in conjunction with the indication of the CIV position, with sufficient time for operator action to terminate the event. Figure 5A-19 shows the design response for this class of ISLOCAs. There is no impact to the SS as a result of this design response. Leakage from the relief valve is detected in the EDT, and appropriate operator action is taken to terminate the event. In addition, none of the changes affect the PSA or EOGs for this event.

5A.4.9 Component Design Responses

5A.4.9.1 Flanges

5A.4.9.1.1 Description of Interface

For the systems that may be subjected to an ISLOCA event, all gasketed flange connections represent a possible leakage pathway leading to the atmosphere. This pathway is applicable only in the systems for which designing to 40 percent of RCS operating pressure is chosen; other systems that are isolated from the RCS because of alternative design responses (such as those described in Subsections 5A.4.4 and 5A.4.5) are not pressurized to full RCS pressure. Flanged connections in the former case therefore have a minimum pressure rating of 40 percent of RCS operating pressure or 63.3 kg/cm^2G (900 psig). This section addresses the gasketed flanged connections in the systems that may be subjected to full RCS pressure.

5A.4.9.1.2 Design Evaluation

The flanges that may be subjected to full RCS pressure are located in systems designed to 63.3 kg/cm^2G (900 psig) (40 percent of RCS operating pressure). These flanges are designed to a minimum of Class 600 of ANSI B16.5 (Reference 5).

All bolting for flanges meet or exceed SA193 GR B7, SA-193 GR B8, or SA-540 B23 allowable stress limits. These material specifications provide for higher preloading stress than used in the NUREG/CR-5603 analysis and therefore increase the maximum allowable pressure without leakage.

All systems with design conditions requiring flanges with a rating higher than Class 600 satisfy the ISLOCA acceptance criteria because the pressure at which these flanges are tested[(1)] is greater than RCS operating pressure.

[(1)] Per ANSI B16.5, flanged joints and flanged fittings may be subjected to system hydrostatic tests at a pressure not to exceed 1.5 times the 38 °C (100 °F) rating rounded off to the next higher 1.76 kg/cm^2D (25 psid) increment.

ANSI B16.5 requires that Class 600 flanges meet, at a minimum, the design conditions of 66.1 kg/cm^2G (940 psig) at 204.4 °C (400 °F). NUREG/CR-5603 presents an analysis to determine the leakage rate from the gasketed flange connections from pressurization for existing power plant designs. The calculated leak rate at the gross leak pressure[(2)] is 1 to 2 mg/sec or approximately 3.6 to 7.2 cc/hr. This leakage rate is considered acceptable for beyond design basis events. In addition, NUREG/CR-5603 states that "… the Gross Leak Pressures for all sizes of Class 600 rated flanges are very high and greater than the range of interest. Thus, Class 600 flanges are not expected to leak when subject to pressures as high as the Reactor Coolant System operating pressure."

NUREG/CR-5603 indicates that the leak rate from a flanged joint is dependent, in part, on the preloading on the gasket. The flange bolts used in the NUREG/CR-5603 study were SA-193 GR B8, which, for a Class 600 flange, caused the bolts to be stressed beyond their yield point. The bolts in the APR1400 design are SA-193 GR B7, SA-193 GR B8, or SA-540 B23, which will allow for higher preloading stresses and therefore increase the maximum allowable pressure without leakage.

The hydrostatic test pressure for all classes of flanges above Class 600 is at least 189.8 kg/cm^2G (2,700 psig), which exceeds the normal RCS operating pressure and therefore meets the ISLOCA acceptance criteria.

5A.4.9.2 Valves

5A.4.9.2.1 Description of Interface

For systems that may be subjected to full RCS pressure, all valves represent a possible leakage pathway leading to an ISLOCA event. This pathway is applicable only in the systems for which designing to 40 percent of RCS operating pressure is chosen; other systems that are isolated from the RCS because of alternative design responses (such as those described in Subsections 5A.4.4 and 5A.4.5) are not pressurized to full RCS pressure. Valves in the former case therefore have a minimum pressure rating of 40 percent of RCS

(2) Gross leak pressure is defined as the system pressure that produce stresses on the bolts equal to the preload stresses. For the Class 600 flanges, this pressure is between 223.8 kg/cm^2G (3,183 psig) and 498.1 kg/cm^2G (7,085 psig), depending on the size of the flange.

operating pressure or 63.3 kg/cm^{2}G (900 psig). This section addresses valves in the systems that may be subjected to full RCS pressure.

5A.4.9.2.2 Design Evaluation

Valves that may be subjected to full RCS pressure are located in systems designed to 63.3 kg/cm^{2}G (900 psig) (40 percent of normal RCS operating pressure). According to the ANSI Standard, ANSI B16.34 (Reference 6), all valves in 63.3 kg/cm^{2}G (900 psig) systems must be at least Class 600 valves. The requirements in this standard indicate that a Class 600 valve will maintain system integrity in the event the system is pressurized to 158.2 kg/cm^{2}G (2,250 psig), normal RCS operating pressure.

All systems with design conditions requiring valves of a rating higher than Class 600 satisfy the ISLOCA acceptance criteria because the pressure at which these valves are tested is greater than RCS operating pressure.

ANSI B16.34 requires the use of at least Class 600 stainless steel valves to meet the design conditions of 63.3 kg/cm^{2}G (900 psig) at 204.4 °C (400 °F). The actual allowable working pressure for the Class 600 valve is 72.4 kg/cm^{2}G (1,030 psig), which exceeds the 63.3 kg/cm^{2}G (900 psig) (40 percent of RCS operation pressure) design criteria for systems exposed to RCS pressure. The working pressure is considered adequate to provide reasonable assurance that the structural integrity of the valves is maintained when pressurized to 158.2 kg/cm^{2}G (2,250 psig), normal RCS operating pressure.

The evaluation presented in NUREG/CR-5603 supports this observation. NUREG/CR-5603 states that "… failure of the adjacent piping will occur prior to the failure of the valve body …" and "… the types of valve stem packings currently used in most nuclear plants tends to compress under high-pressure conditions providing greater resistance to leakage." Thus, according to NUREG/CR-5603, the only concern with respect to failure of valves subject to normal RCS operating pressures is the integrity of the valve bonnet seal. The integrity of the valve bonnet seal is proven during the valve body hydrostatic testing.

The hydrostatic test pressure for all classes of valves higher than Class 600 have a design operating condition of at least 151.9 kg/cm^{2}G (2,160 psig) at 38 °C (100 °F) of test temperature. For a Class 900 valve, the hydrostatic test is performed at 228.5 kg/cm^{2}G

(3,250 psig), which is well above RCS operating pressure., and shows that any valve with a rating higher than Class 600 meets the ISLOCA acceptance criteria.

5A.4.9.3 Pump Seals

5A.4.9.3.1 Description of Interface

For the systems that may be subjected to full RCS pressure, all pump seals represent a possible leakage pathway leading to an ISLOCA event. This pathway is applicable only in systems for which designing to 40 percent of normal RCS operating pressure is chosen; other systems that are isolated from the RCS because of alternative design responses (such as those described in Subsections 5A.4.4 and 5A.4.5) are not pressurized to full RCS pressure. This section addresses the pump seals in the systems that may be subject to full RCS pressure.

5A.4.9.3.2 Design Evaluation

The pumps of interest for an ISLOCA event are the SCP, safety injection pumps (SIP), the CSP, and charging pumps. These pumps are centrifugal and are located in systems that are designed to at least 63.3 kg/cm^{2}G (900 psig) or 40 percent of RCS operating pressure. The design pressure of the CSPs and SCPs is 63.3 kg/cm^{2}G (900 psig), the design pressure of the SIPs is 144.1 kg/cm^{2}G (2,050 psig), and the design pressure of the charging pumps is 225.0 kg/cm^{2}G (3,200 psig). Because the hydrostatic test pressure of SIPs is above the normal RCS normal operating pressure, the pump seals are not expected to leak and the ISLOCA acceptance criteria are met. The hydrostatic test pressure of CSPs and SCPs is approximately 79 kg/cm^{2}G (1,125 psig). NUREG/CR-5603 indicates that the nominal leak rate through the pump seals can be predicted to be 100~200 mg/sec (360~720 cc/hr) with an uncertainty variability of approximately 0.20. This leakage rate is within the makeup capability of the charging pumps and therefore meets the ISLOCA acceptance criteria.

NUREG/CR-5603 indicates that failure of pumps is related to leakage through the spring-loaded mechanical shaft seals. In this study, the mechanical seals for the subject pumps were designed to withstand a pressure of 84.37 kg/cm^{2}G (1,200 psig) to 87.9 kg/cm^{2}G (1,250 psig) without leaking. For greater pressures, the leakage was limited to 100~200 mg/sec (360~720 cc/hr) with an uncertainty variability of 20 percent.

The seal design of the APR1400 pumps for the systems that may be exposed to normal RCS operating pressure is expected to yield similar results. With a nominal leak rate of 100~200 mg/sec (360~720 cc/hr), the ISLOCA acceptance criteria are met due to the fact that the charging capability can make up the lost inventory. In the APR1400 design, the SCP and CSP design pressure is increased to 63.3 kg/cm^{2}G (900 psig), which allows for a higher pressure before seal leakage.

5A.4.9.4 Heat Exchangers

5A.4.9.4.1 Description of Interface

A number of systems that interface with the RCS contain HXs as an integral part of the system design. The postulated rupture of a HX tube provides a pressurization pathway for an ISLOCA event. This section addresses HXs located in systems that may be exposed to full RCS pressure.

5A.4.9.4.2 Design Evaluation

The HXs located in systems that interface with the RCS are designed to at least 40 percent (63.3 kg/cm^{2}G (900 psig)) of full RCS pressure. The HXs that are in operation during Mode 1 are designed to pressures greater than RCS operating pressure except for the charging pump MFHX.

The preliminary design pressures for the HXs that directly connect to the RCS satisfy the ISLOCA acceptance criteria. These HXs and their design pressures are listed in Table 5A-4.

The rationale for excluding a HX tube rupture as an ISLOCA pathway is based on the following factors:

a. Heat exchanger design – the design rating of the HXs is at least 63.3 kg/cm^{2}G (900 psig) except for the charging pump MFHX

b. Tube rupture size – the diameter of the HX tubing is small (no larger than 15.9 mm [5/8 in]), thereby limiting the total loss of inventory in the event of a tube rupture

c. Offsite dose limits – the offsite doses due to a HX tube rupture do not challenge the acceptance criteria for an ISLOCA

d. Event detection – the detection of the event is provided in the MCR to give the operator with ample time to secure the affected HX

e. HX isolation – the isolation capability of the affected HX is provided in the MCR

Heat Exchanger Design

The design of the HXs takes into account several parameters that provide reasonable assurance of maximum protection against tube failure. The selection of the parameters is based on a consideration of normal and abnormal operating conditions, including ISLOCAs. The following summarizes the key considerations for HX sizing as applicable to the ISLOCA concerns:

a. The chemistry of the fluid is closely controlled on both sides of the HXs so that there are no significant changes in impurity levels. In addition, the HXs are designed to preclude the potential for impurity concentration within the HX, such as would occur with fluid boiling at the tube-to-coolant interface. These characteristics are consistent with long-term trouble-free operation.

b. Flow rates and pressure losses are established to retain single-phase flow throughout the HXs to minimize the potential for erosion and maximize component life.

c. The pressure-retaining parts of the HXs are based on simple geometry, which enhances the ability to properly size the component and provide reasonable assurance of a high level of confidence in the design. All HX parts are either flat plates or tubes without any configurations that would produce excessive local stress concentrations. In addition, the ISLOCA acceptance criterion (40 percent of

RCS normal operating pressure) that is applied to system piping is also applicable to the tubes inside the HX except charging pump MFHX.

d. The wall thickness of the HX tubes is specified in accordance with Tubular Exchanger Manufacturers Association (TEMA) Standards. These standards are more conservative than ASME Code requirements. Therefore, the tubes are specified with a greater design wall thickness than required. As an example, the shutdown cooling HX tubing can withstand approximately 33 percent greater design pressure than is required by the ASME Code. This margin in HX tubing is typical of the HXs shown in Table 5A-4.

The pressure-retaining parts of the HXs that are required to be in service during normal operation (i.e., Mode 1) are designed to the same pressure as the attached piping. This pressure is in excess of normal RCS operation pressure except for the charging pump MFHX. Furthermore, the HXs are tested to 125 percent of their design pressure, which provides additional reasonable assurance that the HX tubes can maintain system integrity. This position is consistent with the results of NUREG/CR-5603, excluding high-pressure HXs, as a potential for ISLOCA. This position is also consistent with the NRC position for valves in NUREG/CR-5102 that the valve design is acceptable if tested to a pressure in excess of normal operating pressure.

The letdown HX is an example of a high-pressure design HX. The design pressure of the letdown HX for the APR1400 design is 174.7 kg/cm^{2}G (2,485 psig). Therefore, in accordance with NUREG/CR-5603, the design is acceptable for ISLOCA. All design conditions presented above (i.e., chemistry control, single-phase flow, and TEMA sizing requirements) are incorporated in the letdown HX.

The charging pump MFHX design pressure is 14.1 kg/cm^{2}G (200 psig), which is based on the approaches similar to those in Subsection 5A.4.5. In addition, a backup relief valve is provided in the charging pump miniflow line.

HXs that can be aligned to the RCS but are isolated from the RCS during Modes 1, 2 and 3 satisfy the ISLOCA acceptance criteria based on their design pressure of 40 percent of RCS operating pressure. The SCS HX is an example of a low-pressure design HX normally isolated from the RCS. The design pressure of the SCS HX is 63.3 kg/cm^{2}G (900 psig). This is consistent with the NRC position on piping as presented in NUREG/CR-5102. All design conditions presented above (i.e., chemistry control, single-phase flow, and TEMA sizing requirements) are incorporated in the SCS HXs.

The normal operating pressure for the SCS HX is between approximately 7.0 kg/cm^2G (100 psig) and 31.6 kg/cm^2G (450 psig) and is only in use during shutdown conditions. These conditions do not challenge the design conditions (63.3 kg/cm^2G (900 psig at 204.4 °C [400°F)] of the HX. The HXs are designed for more robust conditions than required for their normal use. Therefore, they have the margin to provide reasonable assurance of integrity.

The design requirements for the CSS are to provide containment pressure control following a design basis event. Because the CSS is also designed to back up the SCS, the HX design must include the same conditions as the SCS HX. Therefore, the normal operating pressure for the CSS HX must be considered to be between approximately 7.0 kg/cm^2G (100 psig) and 31.6 kg/cm^2G (450 psig).

These conditions, as with the SCS, do not challenge the design conditions (63.3 kg/cm^2G (900 psig) at 204.4 °C (400°F)) of the HX. The HXs are designed for more robust conditions than required for their normal use. Therefore, they inherently have margin to provide reasonable assurance of integrity.

Based on these observations, postulating a HX tube rupture as part of the ISLOCA evaluation is not considered necessary.

Tube Rupture Size

The ISLOCA pathway through an HX is limited by the diameter of a tube based on a double-ended guillotine break. Table 5A-4 provides the best estimate for the tube diameter of the HXs that is potentially subjected to operating RCS pressure. Since these diameters are all much less than the letdown line diameter, the consequences of an HX tube rupture is bounded by the consequences of a letdown line break. This implies that RCS inventory is easily controlled by the makeup system’s capability.

As an example, the letdown HX tube is 15.9 mm (5/8 in) in diameter and the letdown line is 50.8 mm (2 in) in diameter, more than a factor of 3 larger. Mass flow rate out of a break is proportional to the square of the diameter, which implies that, for the same pressure drop across the break, the mass flow rate out of a letdown line break is more than 10 times larger than the flow rate out of a HX tube rupture. The safety analysis in Subsection 15.6.2 considers letdown line breaks that clearly bound the consequences of HX tube ruptures.

Offsite Dose Limits

The ISLOCA acceptance criteria for offsite dose limits are bounded by the leakage rate and total discharge of the postulated letdown line break presented in Subsection 15.6.2. Assuming that all leakage from a tube rupture is discharged directly to the auxiliary building instead of the secondary system, a HX rupture still satisfies the ISLOCA acceptance criteria since the tube diameters are all less than the letdown line diameter.

Event Detection

The various means to detect a tube rupture in HXs include pressure, temperature, flow, liquid level, valve position indication, and radiation indications and alarms. All of these methods are provided in the MCR. Examples of the method of detection are presented in the following subsections on high- and low-pressure HXs.

Heat Exchanger Isolation

All HXs except for the charging pump MFHX can be isolated from the MCR with equipment designed to operate under the worst-case flow and pressure.

Detection and isolation of the affected HX are provided, going beyond the credited acceptance criteria for ISLOCA and assuming a failure of a letdown HX tube. The detection and isolation are provided to protect the component cooling water system (CCWS) and all system interfaces off the CCWS.

The CCWS is a closed system whose only interface with the RCS is through other systems' HXs. The major components of the system are the pumps, HXs, relief valves, and surge tanks. Of these components, the surge tank and relief valves at each of the HXs listed in Table 5A-4 provide the necessary protection for an ISLOCA. The surge tank provides a reservoir of water for the expansion and contraction of the component cooling water

(CCW) due to thermal effects and leakage and provides a reservoir for dampening pressure transients. These capabilities allow the system to accommodate mass additions without pressurization. The relief valves, located in the outlet line from each HX, provide additional thermal and mass addition relief protection. Further, since all HXs have a relief valve, the combined effect provides redundant protection.

Example of a High-Pressure Heat Exchanger

The letdown HX is subjected to full RCS pressure and designed to a pressure in excess of RCS operating pressure. A postulated letdown HX tube rupture during high RCS pressure opens a pathway to the CCW. The result of this ISLOCA event is mitigated by the CCWS relief valve for the letdown HX. The mass flow rate through the postulated break is discharged through the CCW/letdown HX's relief valve to the sump in containment. Since the event is totally within containment and the CCW system is protected from overpressurization, the ISLOCA acceptance criteria are satisfied. Isolation of a letdown HX tube rupture is provided by the RCS PIVs in the letdown line.

Example of a Low-Pressure Heat Exchanger

The shutdown cooling HX is aligned to the RCS but is only subjected to RCS pressure during low-pressure operation. Therefore, it is designed to 63.3 kg/cm^2G (900 psig) in accordance with the NRC position presented in the NRC Letter. However, if the system is misaligned during high RCS pressure and a tube is postulated to rupture, a pathway to the CCWS would be provided. For this ISLOCA event, the CCWS integrity is maintained, as with the letdown HX ISLOCA event since the mass flow rate added to the CCWS would be discharged through the CCW/SCS HX's relief valve to the respective sump.

Detection for a SCS HX tube rupture is provided by the instrumentation of the SCS, CCW, and local area monitors throughout the plant. Subsection 5A.5 provides a general overview of the instrumentation provided in the primary systems necessary to detect an ISLOCA. The following details the indication and alarms provided in the MCR for a SCS HX tube rupture.

Prior to a SCS HX tube rupture, the operator is alerted to a pressurization event from either the return line or supply line (Figure 5A-1) through indication and alarms in the MCR. The return line requires three normally locked closed valves (that have the power removed from their motor operator) to open. These valves all have position indication in the MCR and alarm if they are not fully closed concurrent with high RCS pressure.

Furthermore, the RCS low-temperature overpressure protection (LTOP) relief provides another source of overpressure protection since these valves open at 38.32 kg/cm^{2}A (545 psia) after the first two SCS isolation valves open and discharge to the IRWST within containment. Although the design criteria for the LTOP valves are to protect the RCS during a low-temperature transient, they are operable under this scenario and assist in mitigating the severity of a postulated tube rupture.

From the supply line, four check valves in series would have to fail and two normally closed isolation valves would have to open. Indication and alarm are provided in this pathway through continuous monitoring of the SCS pressure upstream of the DVI check valve. The intent is to monitor the integrity of the DVI check valve during plant operation. For normal pressurization, the operating requirements provide the necessary instructions for relieving pressure to the RDT, within containment, upon high pressure of 70.3 kg/cm^{2}G (1,000 psig) or the alarm setpoint. The two isolation valves in this line have position indication in the MCR.

The CCWS is protected from overpressurization by a relief valve at each HX, as described above. Therefore, the instrumentation provided in the control room is intended for monitoring the impact on the entire system. The instrumentation includes a radiation monitor in each train of the CCWS, level indication in each surge tank, CCW pump differential and discharge pressure, and CCW system temperature.

Finally, area radiation alarm provides immediate identification of an event. Chapter 12 describes such radiation monitors and their use.

Isolation of the SCS HX is performed from the MCR by closing any of the valves in the SCS supply and return lines. Since all of the SCS outside containment are designed to 40 percent of the RCS operating pressure in accordance with the NRC position in NRC the Letter, the ISLOCA acceptance criteria are met. Based on the primary side systems' design pressure, pressure isolation provisions, indications, alarms, and relief protection provided in the plant, the design pressure of the shellside of the SCS HX and all other low design pressure HXs need not be set at 63.3 kg/cm^2 G (900 psig) solely for tube rupture isolation.

5A.4.9.4.3 RCP High-Pressure Seal Cooler

The RCP high-pressure seal cooler tubeside in the APR1400 design is the same as the design pressure of the RCS. A description of the overpressure protection provided in the component cooling water system is provided in Subsection 9.2.2.

5A.4.9.5 Vents, Drains, and Local Samples

5A.4.9.5.1 Description of Interface

Various systems that interface directly or indirectly with the RCS have vents and drains that provide a potential pathway for an ISLOCA event. These vents and drains are necessary for thermal and pressure relief, for venting and draining operations, and for the collection of samples from local sections in the systems. This section addresses vents, drains, and local sample connections in the systems that may be subjected to full RCS pressure.

5A.4.9.5.2 Design Evaluation

For the systems that may be subjected to normal RCS operating pressure, all vents and drains represent a possible leakage pathway leading to an ISLOCA event. This pathway is applicable only in the systems for which designing to 40 percent of normal RCS operating pressure is chosen; other systems that are isolated from the RCS because of alternative design responses (such as those described in Subsections 5A.4.4 and 5A.4.5) are not pressurized to full RCS pressure.

The APR1400 design provides an isolation capability against pressures equal to or greater than 158.2 kg/cm^{2}G (2,250 psig) or RCS normal operating pressure.

The justification of this design response is based on recognizing that:

a. All vent and drain lines are of a sufficiently small diameter that discharge flow rates are limited[(3)]

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(3) Assuming all isolation valves are open

b. The discharge flow rates and the resulting offsite doses are bounded by the results described in Chapter 15.

All system vents and drains are directed to a collecting tank either directly to the EDT or to the liquid waste management system (LWMS) through sump drains to allow sufficient time for operator response because of the limited flow rates characterizing these events and the size of the collecting tank. Furthermore, local radiation monitors alert the operator of an event.

Unlike discharges to tanks, discharges to the atmosphere are not characterized by collection or pressure containment capabilities. Therefore, it is considered sufficient, in view of the practical limitations of such interfaces, to design these interfaces with isolation capabilities to withstand normal RCS operating pressure without unacceptable leakage. The design of normally closed isolation valves conforms to the requirements described in Subsection 5A.4.9.2.

5A.5 Detection of ISLOCA

Section 5A.4 describes the various ISLOCA events that have been postulated to occur in the APR1400 design. The design responses implemented for all of these events meet the ISLOCA acceptance criteria described in Subsection 5A.3.2. In particular, these design responses provide reasonable assurance that the structural integrity of systems is maintained and that any leakages are limited to a small fraction of makeup capabilities. In general, the design responses described in Subsection 5A.4 are intended to prevent ISLOCAs from occurring.

Based on the evaluation presented in Subsection 5A.4, the ISLOCA detection capability emphasizes the detection of ISLOCA precursors (i.e., primarily events that pressurize a system) since ISLOCA events (i.e., events characterized by the actual loss of coolant outside containment) are limited to very small leakages. The detection of ISLOCA events relies on the ability to detect relatively small flow rates of reactor coolant, such as leakages within a system (e.g., through valves) or to atmosphere (e.g., through seals and valve packings).

As a result, the following instrumentation types were reviewed to provide reasonable assurance of their inclusion in the MCR:

a. Valve position indication

b. Pressure indication

c. Tank level indication

d. Radiation monitoring

5A.5.1 Valve Position Indication

In accordance with the ISLOCA definition provided in Subsection 5A.3.1, ISLOCA precursors can occur because of the inadvertent opening of valves, failure of containment isolation, or the postulation that valves are fully opened. In any case, information in the MCR concerning the position of valves appears to be critical to detecting such precursor events. This observation is consistent with SECY-90-016 and SECY-93-087 (Reference 8), which require that valve position indication be available in the MCR when CIV operators are de-energized. Table 5A-5 identifies all CIVs and their indication capabilities for the systems and subsystems for which ISLOCAs have been identified to potentially begin. Table 5A-5 shows that all valves that could initiate an ISLOCA precursor have appropriate valve position indication in the MCR with which operators can detect any misalignment.

5A.5.2 Pressure Indication

ISLOCA precursor events can be characterized by leakages (i.e., relatively small flow rates) through valves within systems. Such events can result in the pressurization of a lower-pressure system. Table 5A-6 identifies pressure indication in the MCR for the regions of systems and subsystems described in Section 5A.4 that are susceptible to such precursor events. These pressure indicators allow operators to detect pressurization events that are caused by leakages too small to be measured reliably using existing equipment.

5A.5.3 Tank Level Indication

ISLOCA precursor events can also be characterized by leakages to tanks (e.g., through vents and drains to the EDT). Such events result in overfilling of the tanks with associated overpressurization. Table 5A-7 identifies the level indication in the MCR for all of the tanks described in Section 5A.4 that would collect leakages during such precursor events. These level indicators allow operators to detect tank filling and with the design changes described in Section 5A.4, allow the operator time to determine the source of the leakage.

5A.5.4 Radiation Monitoring

ISLOCA events can be characterized by leakages from systems to the atmosphere, such as through valve packings and pump seals, or by misaligned vent or drain valves. Because the flow rates involved can be too small to detect reliably using existing instrumentation and because the sources of such leakages are too many and distributed widely throughout the plant, it is concluded that leakages of this type can be detected most effectively with radiation area monitors throughout the plant. Chapter 12 discusses such radiation monitors and their use.

5A.6 References

1. Information Notice 92-36, “Intersystem LOCA Outside Containment,” U.S. Nuclear Regulatory Commission, May 7, 1992.

2. NRC Letter, “Preliminary Evaluation of the Resolution of the Intersystem Loss-of-Coolant-Accident (ISLOCA) Issue for the Advanced Boiling Water Reactor (ABWR) – Design Pressure for Low Pressure Systems,” Docket Number 52-001.

3. NUREG/CR-5603, “Pressure-Dependent Fragilities for Piping Components: Pilot Study on Davis-Besse Nuclear Power Station,” U.S. Nuclear Regulatory Commission, October 1990.

4. SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirement,” U.S. Nuclear Regulatory Commission, January 12, 1990.

5. ANSI B16.5, “Pipe Flanges and Flanged Fittings,” American National Standards Institute, 2009.

6 ANSI B16.34, “Steel Valves,” American National Standards Institute, 2009.

7 NUREG/CR-5102, “Interfacing Systems LOCA: Pressurized Water Reactors,” U.S. Nuclear Regulatory Commission, February 1989.

8 SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs,” U.S. Nuclear Regulatory Commission, April 2, 1993.

Table 5A-1

APR1400 ISLOCA Design

Pressurization Pathway	Design	Figure
Shutdown Cooling System	The design pressure for all sections outside containment is at least 63.3 kg/cm^{2}G (900 psig).	5A-10
SIS Delivery Line	The design pressure for all sections outside containment is at least 63.3 kg/cm^{2}G (900 psig).	5A-11
Containment Spray System	All piping sections outside containment have been designed to 63.3 kg/cm^{2}G (900 psig). Connections to the refueling pool fill line in CSS IRWST return line have been relocated from the SCS return line and a spool piece has been included.	5A-12
Letdown Line	A high-pressure alarm has been installed downstream of the letdown control valves to initiate a manual isolation of the letdown line by the containment isolation valve when high pressure is sensed.	5A-13
Charging Line	A pressure sensor has been installed upstream of the charging pumps to initiate a manual isolation of the charging line by the containment isolation valve when high pressure is sensed.	5A-14
Hot Leg Sampling	A pressure relief valve has been added upstream of discharge isolation valves to VCT and recycle drain header.	5A-17
Pressurizer Surge Line Sampling	A pressure relief valve has been added upstream of discharge isolation valves to VCT and recycle drain header.	5A-18
Pressurizer Steam Space Sampling	A pressure relief valve has been added upstream of discharge isolation valves to VCT and recycle drain header.	5A-19

Table 5A-2

Systems Interfacing with the CVCS Downstream of the Letdown Control Valve

Hydrogen Supply System
Nitrogen Supply System
Gaseous Waste Management System
Solid Waste Management System
Liquid Waste Management System
Makeup/Boron Recovery System
Sampling System

Table 5A-3

Low-Pressure Equipment of the CVCS

Equipment	Diameter, Height, Fabrication Location	Volume	Design Pressure
Boric Acid Storage Tank	Field fabricated	946.0 m^3 (250,000 gal)	0.11 kg/cm^2G (1.5 psig)
Volume Control Tank	2,438.4 mm (8 ft) diameter 5,852 mm (19.2 ft) high	25.4 m^3 (6,700 gal)	Internal 5.3 kg/cm^2G (75 psig) External 1.05 kg/cm^2G (15 psig)
Equipment Drain Tank	3,048.0 mm (10 ft) diameter 5,486.0 mm (18 ft) long	36.0 m^3 (9,500 gal)	Internal 2.1 kg/cm^2G (30 psig) External 1.05 kg/cm^2G (15 psig)
Boric Acid Batching Tank	1,524.0 mm (5 ft) diameter 914.4 m (3 ft) high	2.38 m^3 (630 gal)	Atmospheric
Holdup Tank	Field fabricated	1,590.0 m^3 (420,000 gal)	0.11 kg/cm^2G (1.5 psig)
Reactor Makeup Water Tank	Field fabricated	1,495 m^3 (395,000 gal)	0.56 kg/cm^2G (8 psig)
Purification Ion Exchangers	1,066.8 mm (3.5 ft) diameter 3,013.0 mm (9.9 ft) high	1.6 m^3 (416.7 gal)	14.1 kg/cm^2G (200 psig)
Deborating Ion Exchanger	1,066.8 mm (3.5 ft) diameter 3,013.0 mm (9.9 ft) high	1.6 m^3 (416.7 gal)	14.1 kg/cm^2G (200 psig)
Pre-Holdup Ion Exchanger	1,066.8 mm (3.5 ft) diameter 3,013.0 mm (9.9 ft) high	1.6 m^3 (416.7 gal)	14.1 kg/cm^2G (200 psig)

Table 5A-4

Heat Exchangers that may be Exposed to Full RCS Operating Pressure

Heat Exchanger		Tubeside Design Pressure, kg/cm^{2}G (psig)	Typical Tube OD, mm (in)
Normally Aligned to RCS	High Pressure Seal Cooler	212.7 (3,025)	9.5 (3/8)
	Regenerative	212.7 (3,025) [(1)]	9.5 (3/8)
	Charging Pump Miniflow	14.1 (200)	15.9 (5/8)
	Letdown	174.7 (2,485)	15.9 (5/8)
Isolated from RCS by Valves	Shutdown Cooling	63.3 (900)	19.1 (3/4)
	SCP Miniflow	63.3 (900)	15.9 (5/8)
	Containment Spray	63.3 (900)	19.1 (3/4)
	CSP Miniflow	63.3 (900)	15.9 (5/8)

(1) Shellside design pressure; tubeside design pressure is 174.7 kg/cm^{2}G (2,485 psig)

Table 5A-5

Containment Isolation Valve Position Indication

System Interface	CIV	Type Indication [1]
SCS Supply Line	SI-653/654	ZS
	SI-655/656	ZS
SCS Return Line	SI-600/601	ZI and ZS
Safety Injection System	SI-616/626 636/646	ZI and ZS
	SI-322/332	ZS
	SI-602/603	ZI and ZS
	SI-321/331	ZI and ZS
Letdown Line	CV-515/516	ZS
	CV-522/523	ZS
Charging System	CV-240	ZS
	CV-524	ZS
Seal Injection Line	CV-241/242 243/244/255	ZS
RCP Controlled Bleed off	CV-505/506	ZS
Hot Leg Sampling	SS-001/002	ZS
Pressurizer Surge Line Sampling	SS-003/004	ZS
Pressurizer Steam Space Sampling	SS-005/006	ZS
Containment Spray System		All CSS interfaces with the RCS are through the SCS CIVs

(1) ZI designates position indication, and ZS designates a position switch indication in the MCR.

Table 5A-6

Pressure Indication

System Interface	Instrument	Location
SCS Supply Line	P-300/301	Pump suction
SCS Return Line	P-302/305	Pump discharge
	P-329/349	DVI line
Safety Injection System	P-319/329/339/349	DVI line
Letdown Line	P-201	Upstream of the letdown control valves
	P-220	Pressure indicator downstream of the letdown control valves
	P-251	Equipment drain tank
Charging System	P-212A	Upstream of the regenerative HX
	P-211	Pressure indicator in the charging pump suction line.
Seal Injection Line	P-251	Indication in the EDT.
Controlled Bleedoff	P-215	Combined bleedoff discharge line.
Containment Spray System		All CSS interfaces with the RCS are through the SCS CIVs

Table 5A-7

Tank Level Indication

System Interface	Instrument
Equipment Drain Tank	L-251
Volume Control Tank	L-226/227

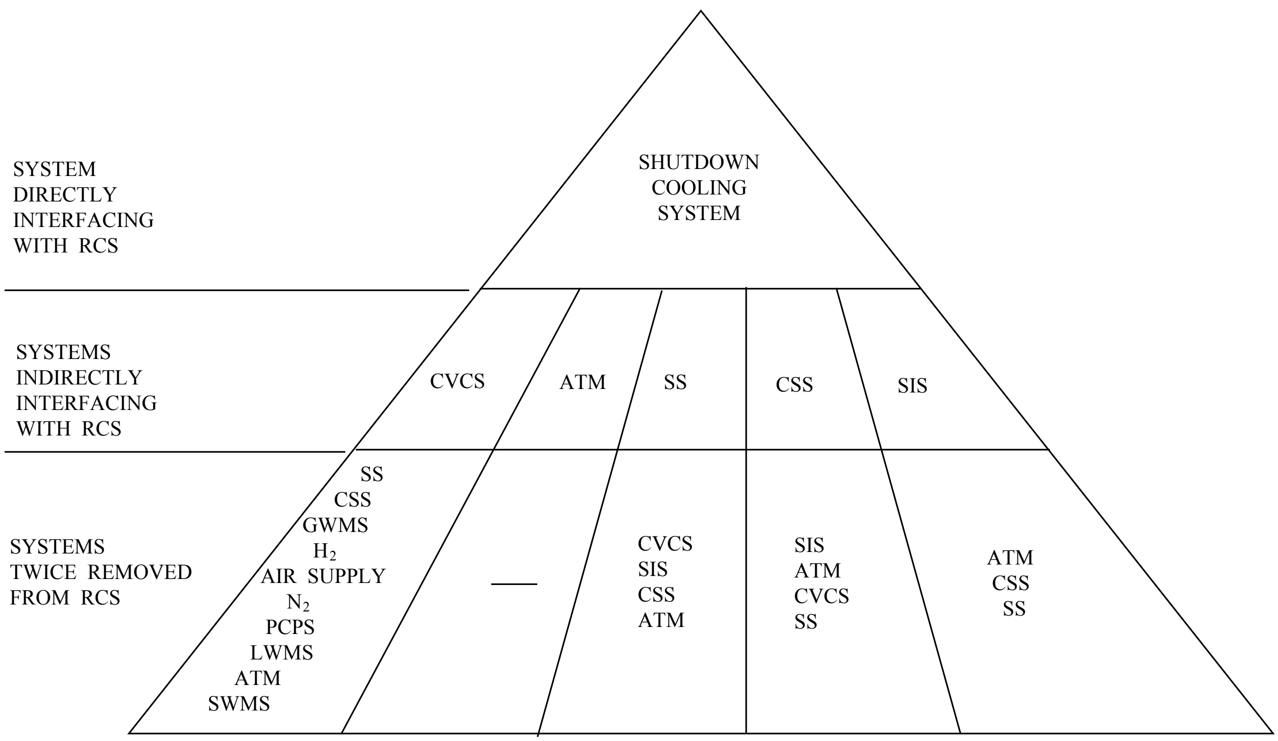

Figure 5A-1 Shutdown Cooling System

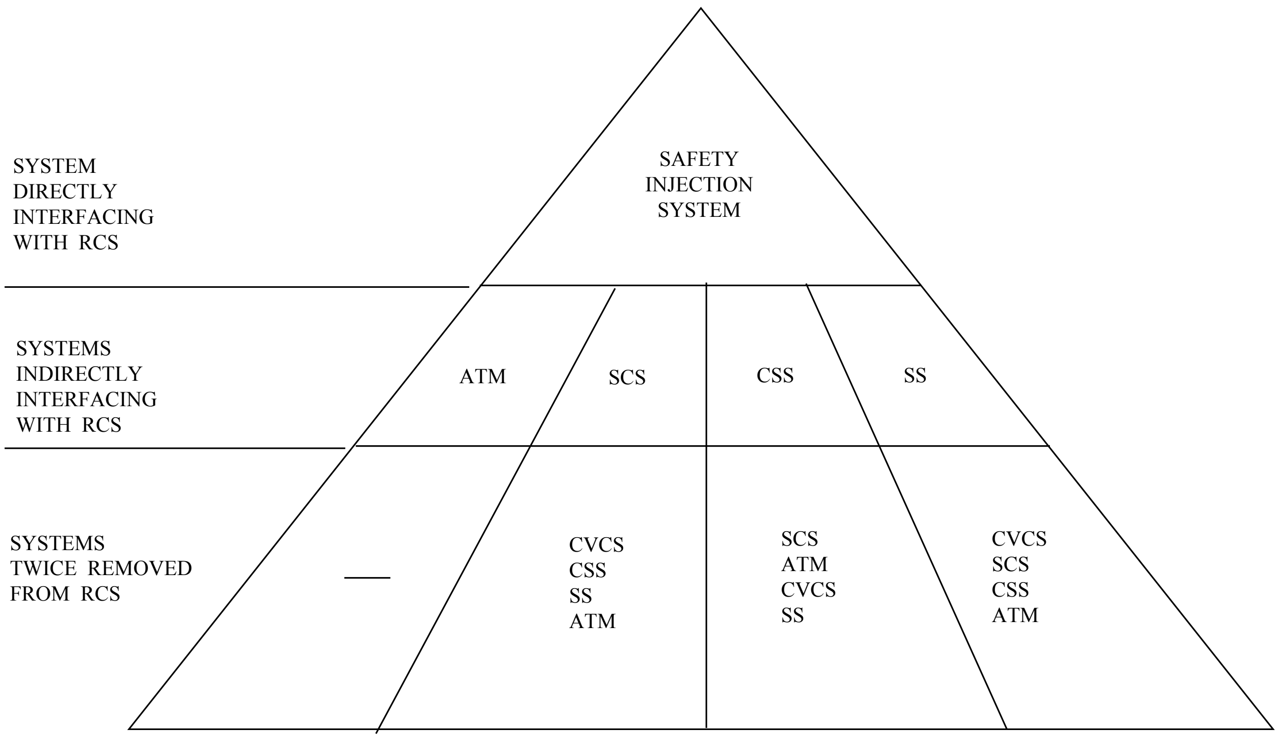

Figure 5A-2 Safety Injection System

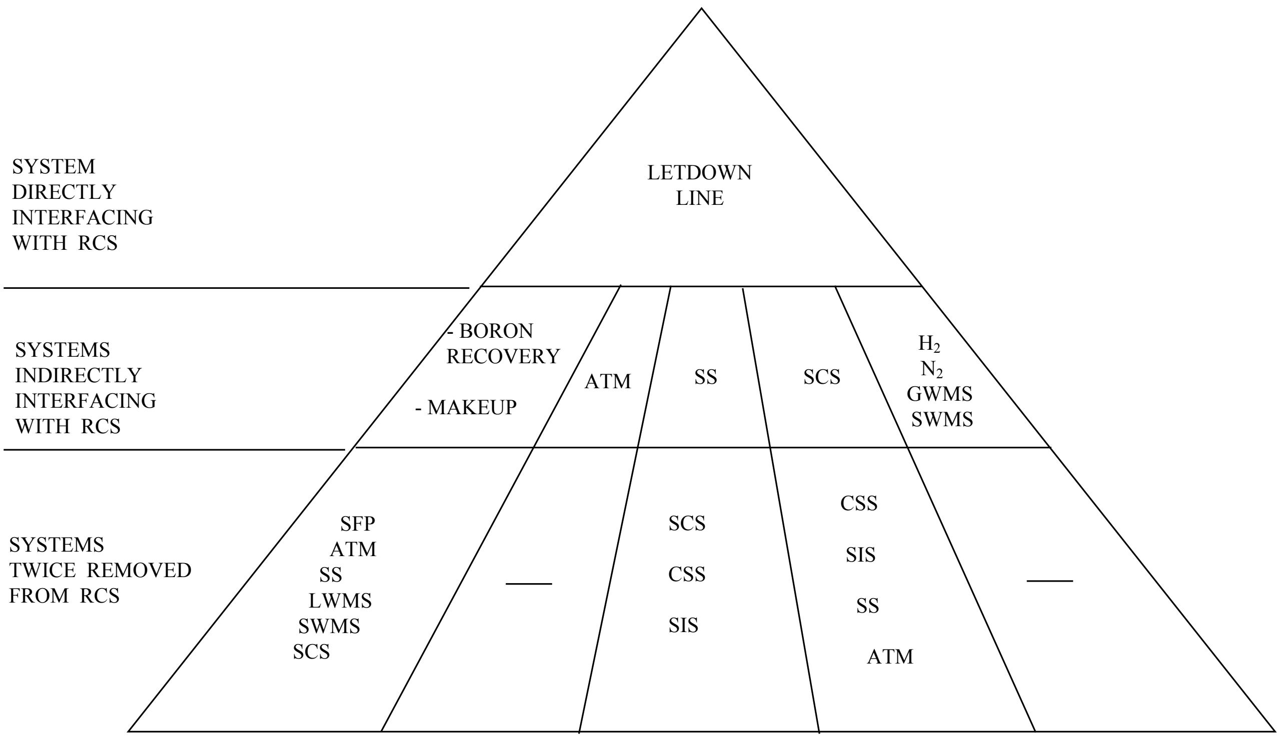

Figure 5A-3 Letdown Line

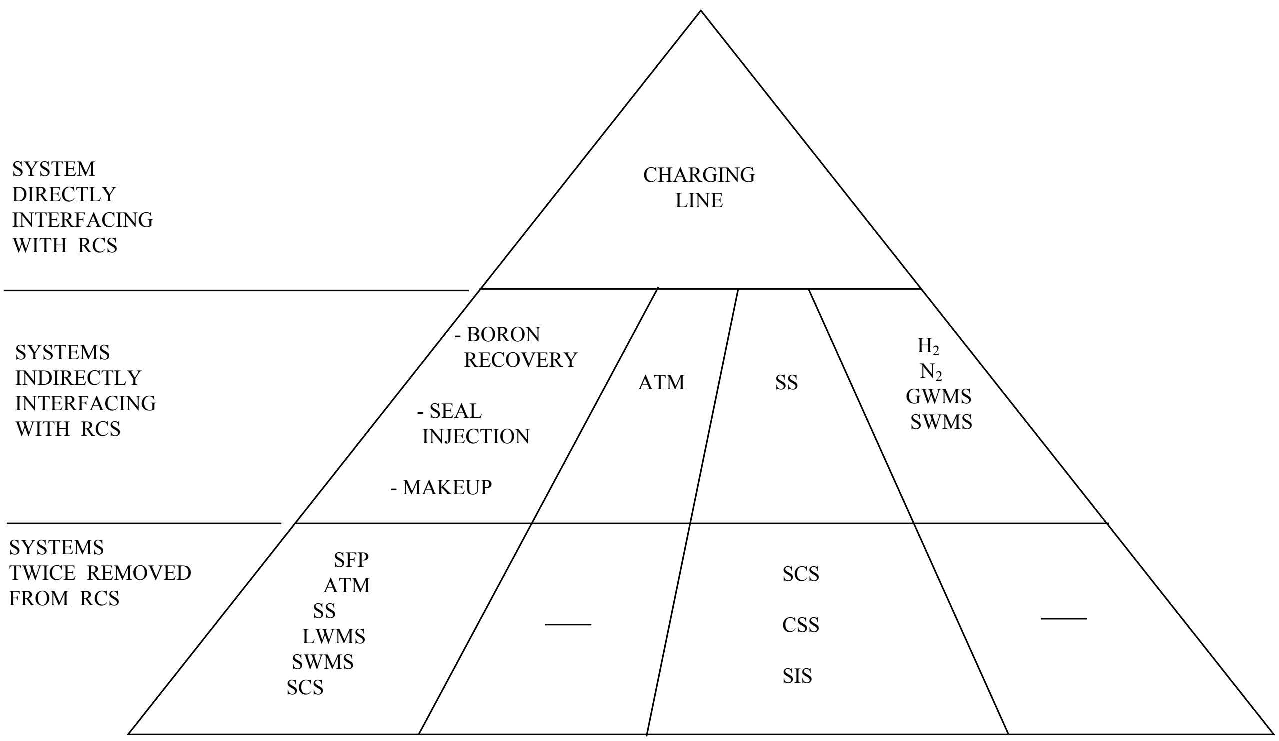

Figure 5A-4 Charging Line

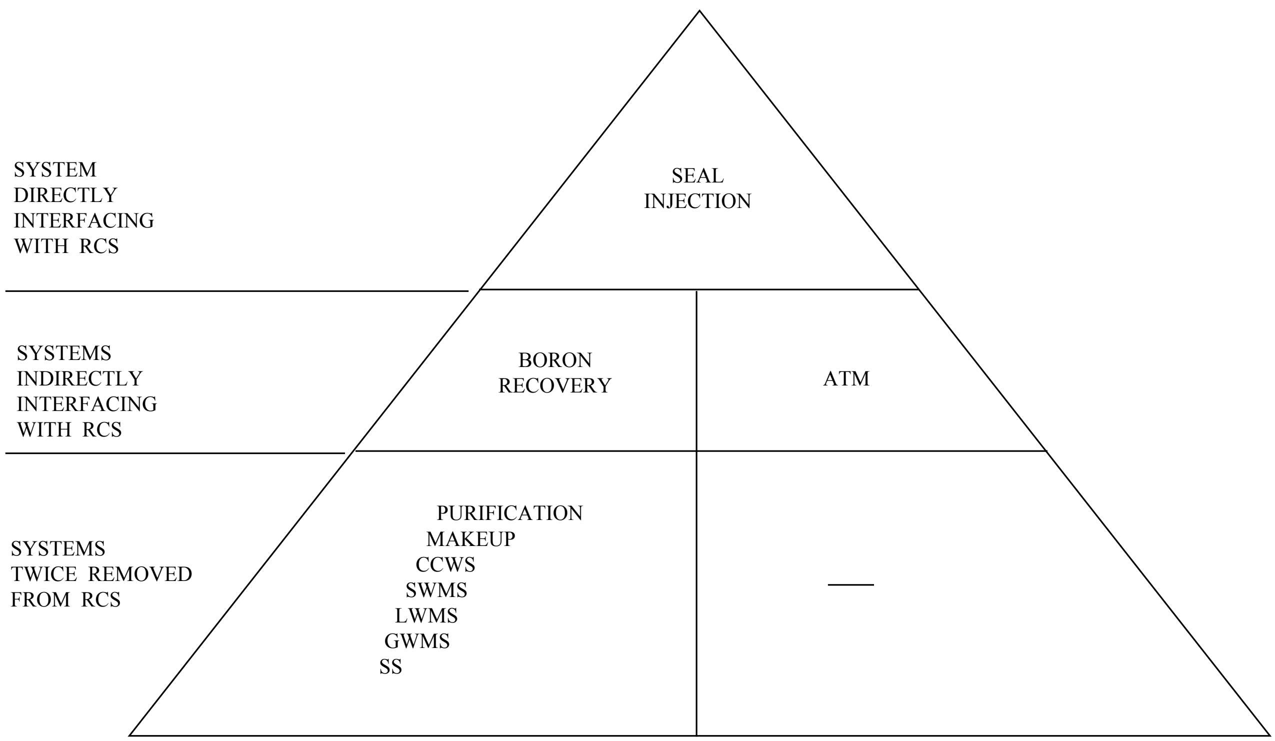

Figure 5A-5 RCP Seal Injection

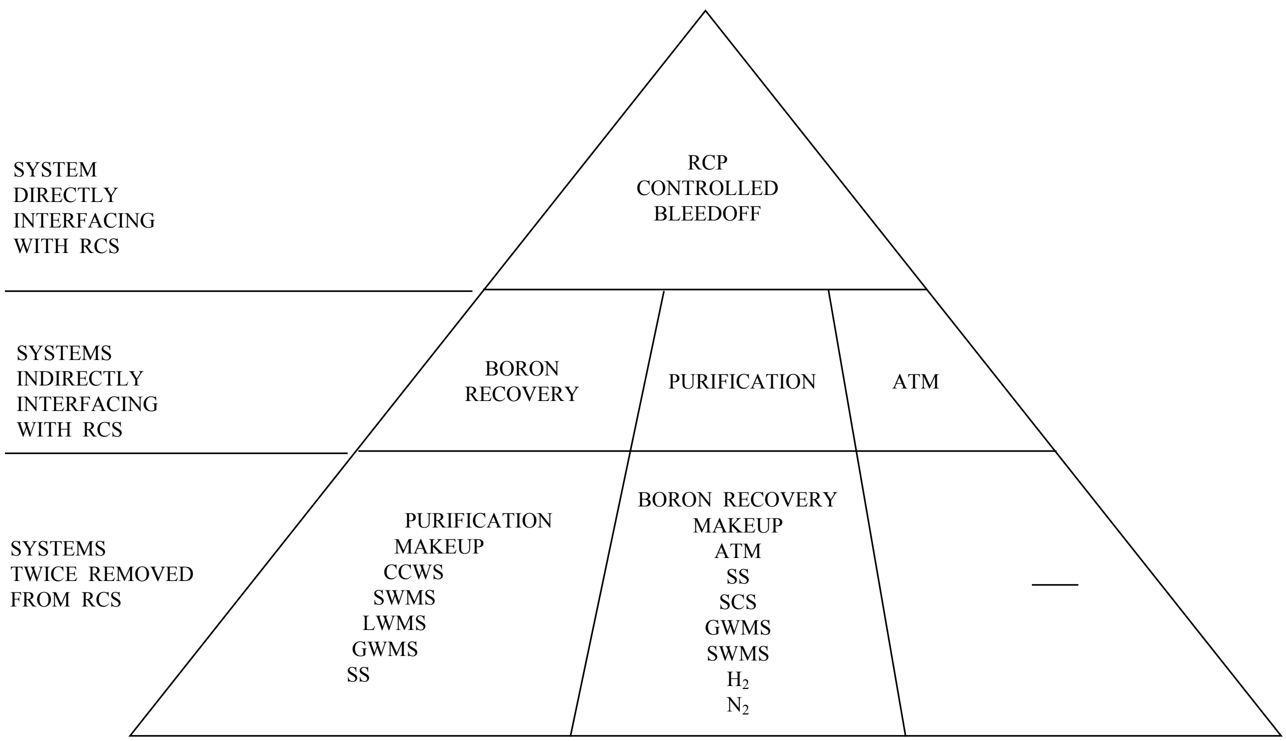

Figure 5A-6 RCP Controlled Bleedoff

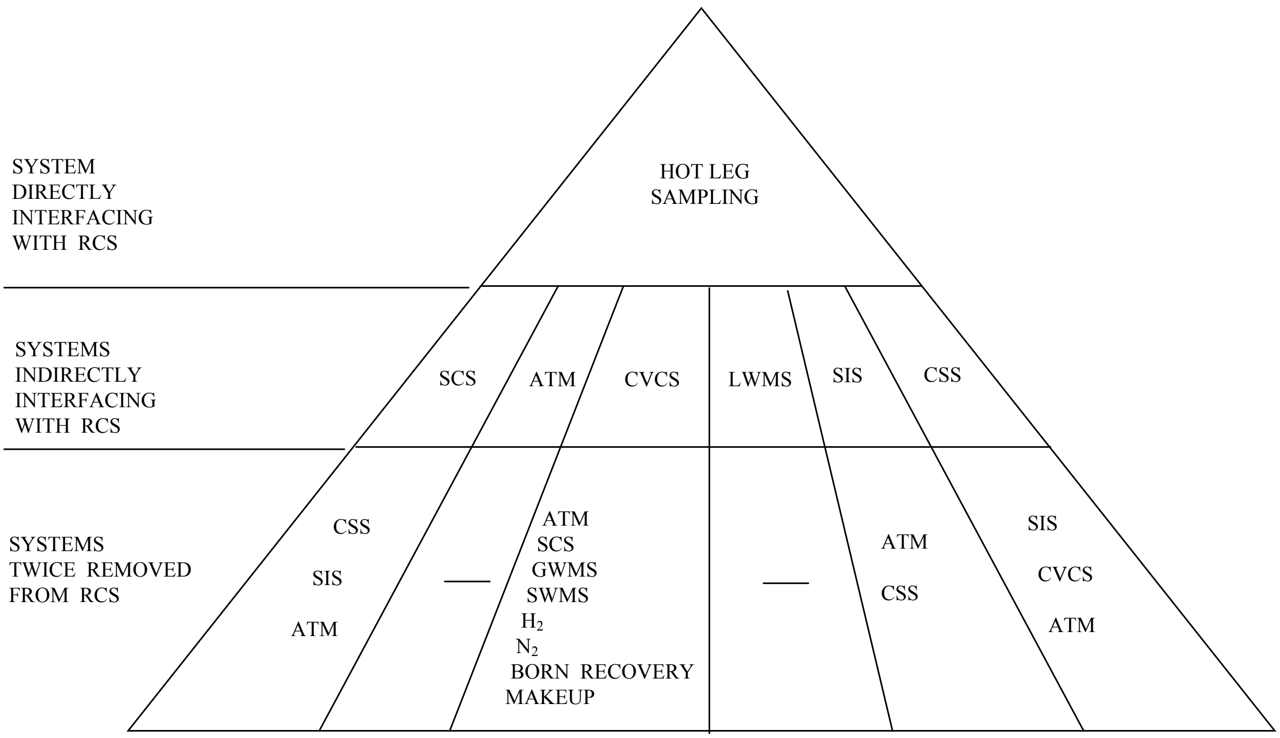

Figure 5A-7 Sampling – Hot Leg

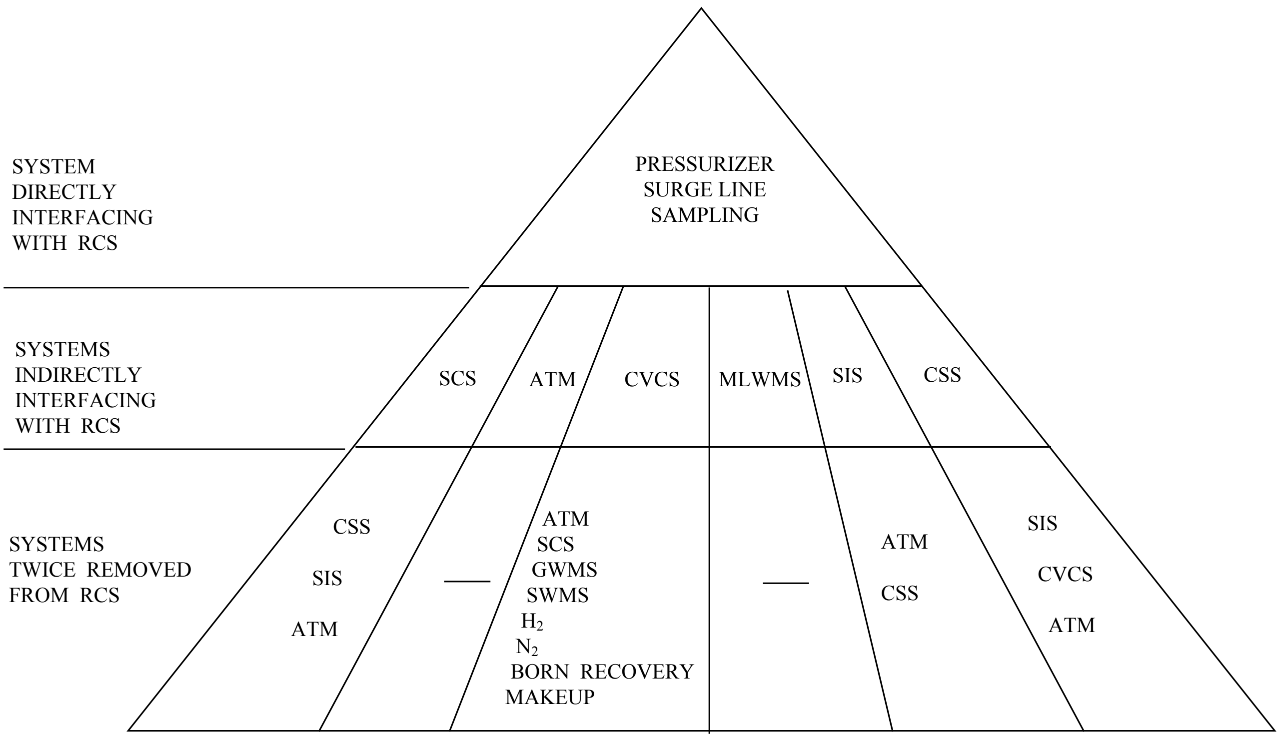

Figure 5A-8 Sampling – Pressurizer Surge Line

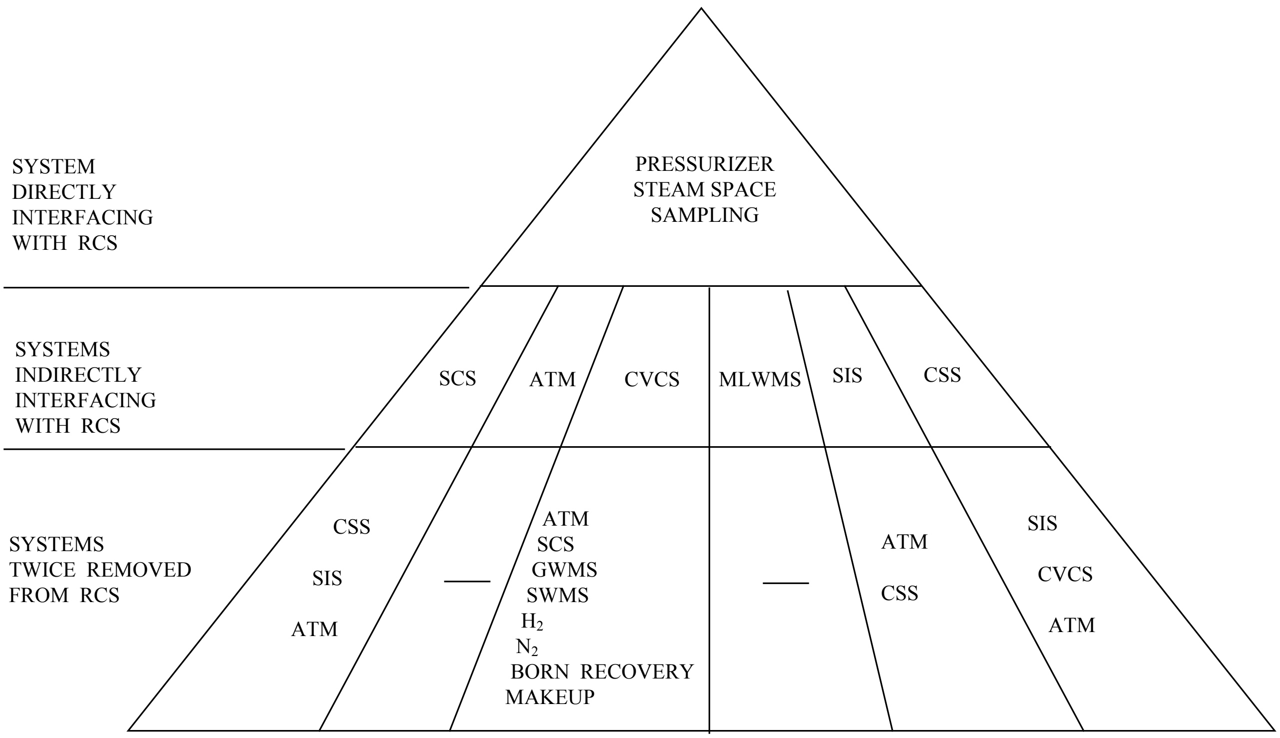

Figure 5A-9 Sampling – Pressurizer Steam Space

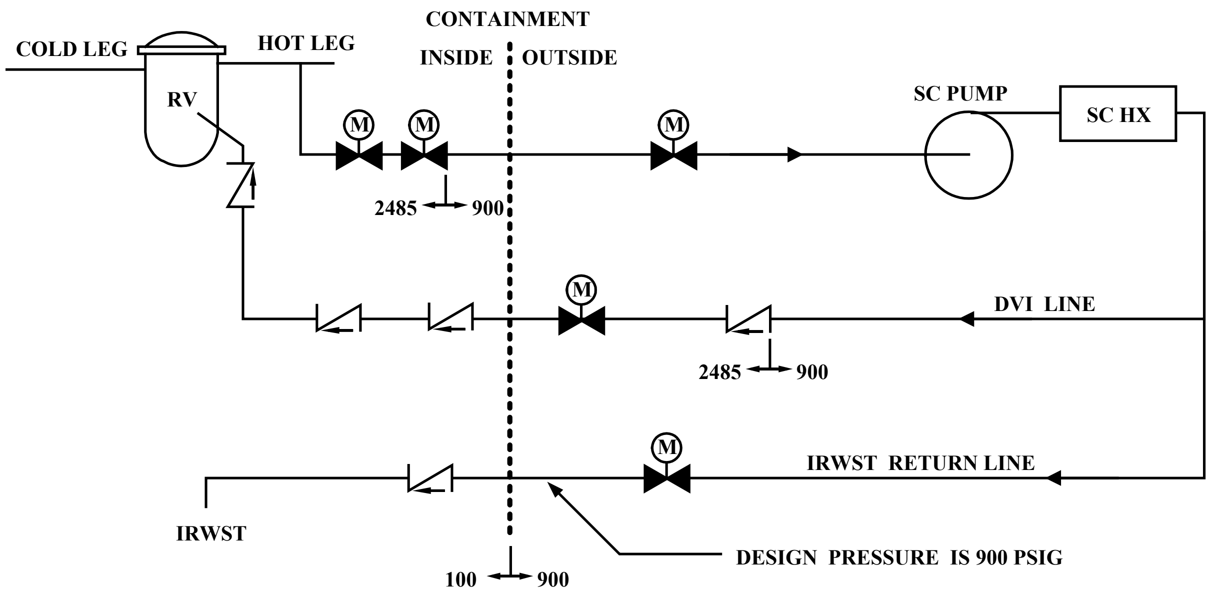

Figure 5A-10 Shutdown Cooling System

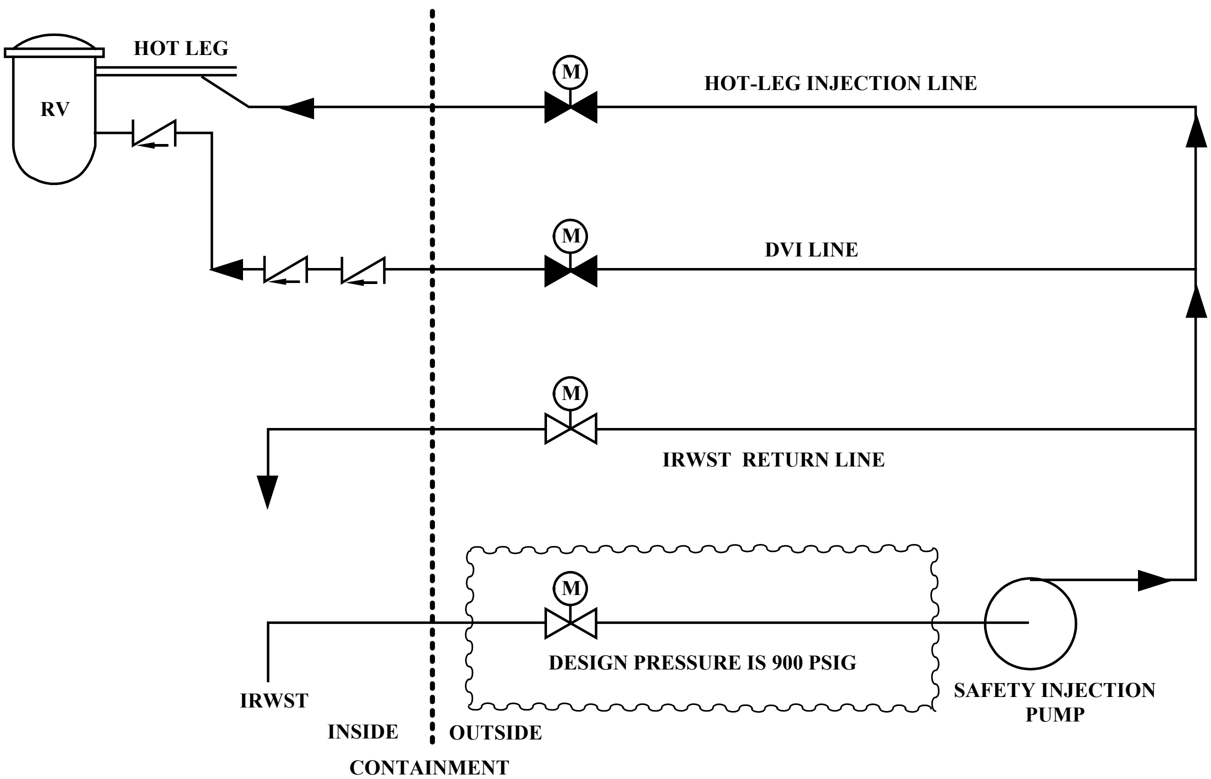

Figure 5A-11 Safety Injection System (Low-Pressure Line Interface)

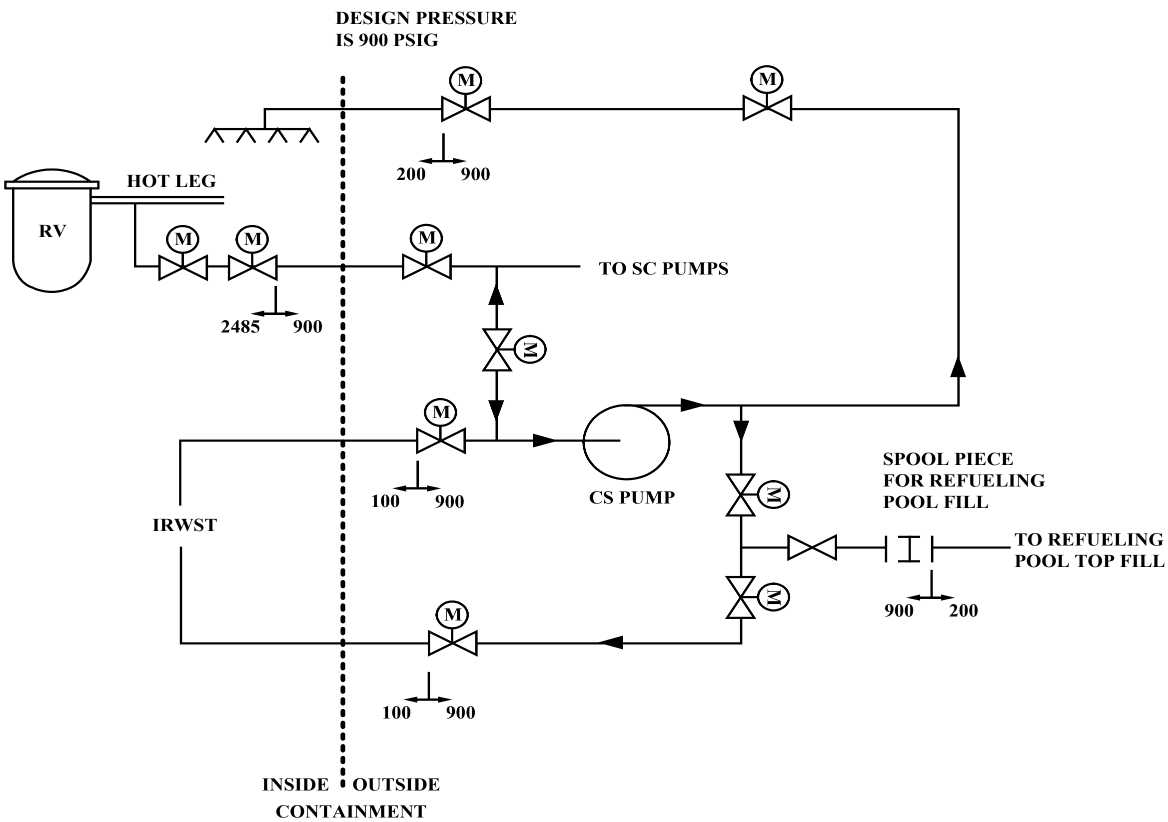

Figure 5A-12 Containment Spray System

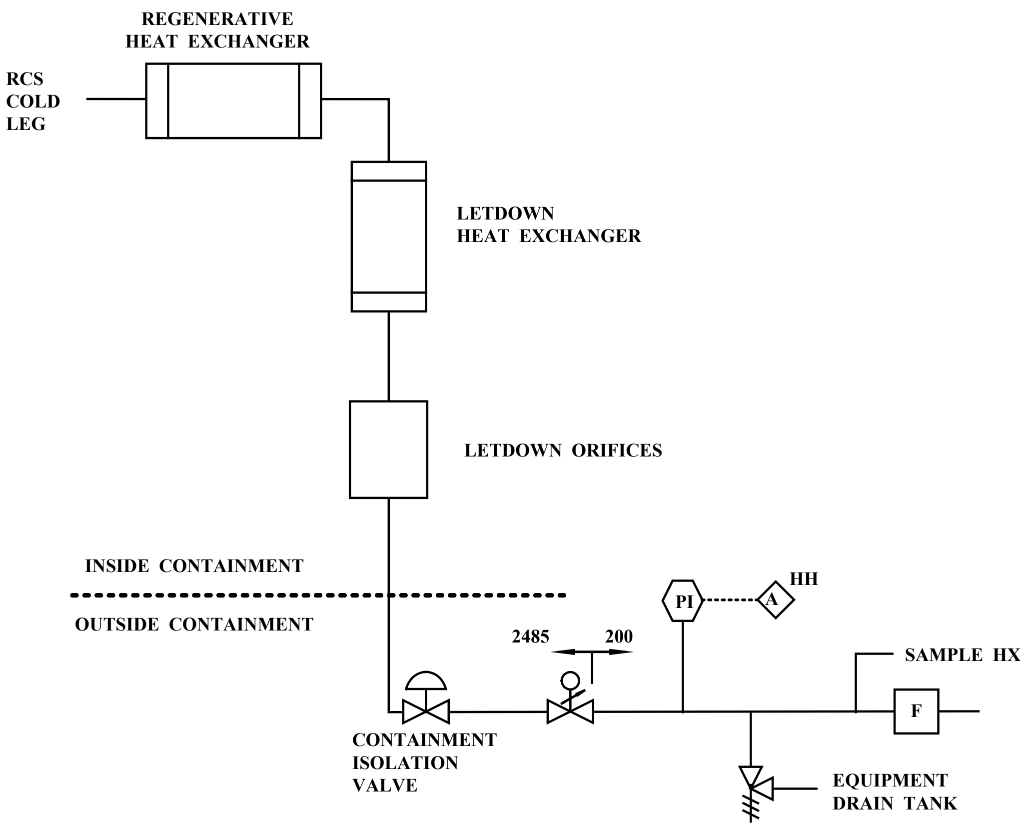

Figure 5A-13 Letdown Line (including Pressure Alarm)

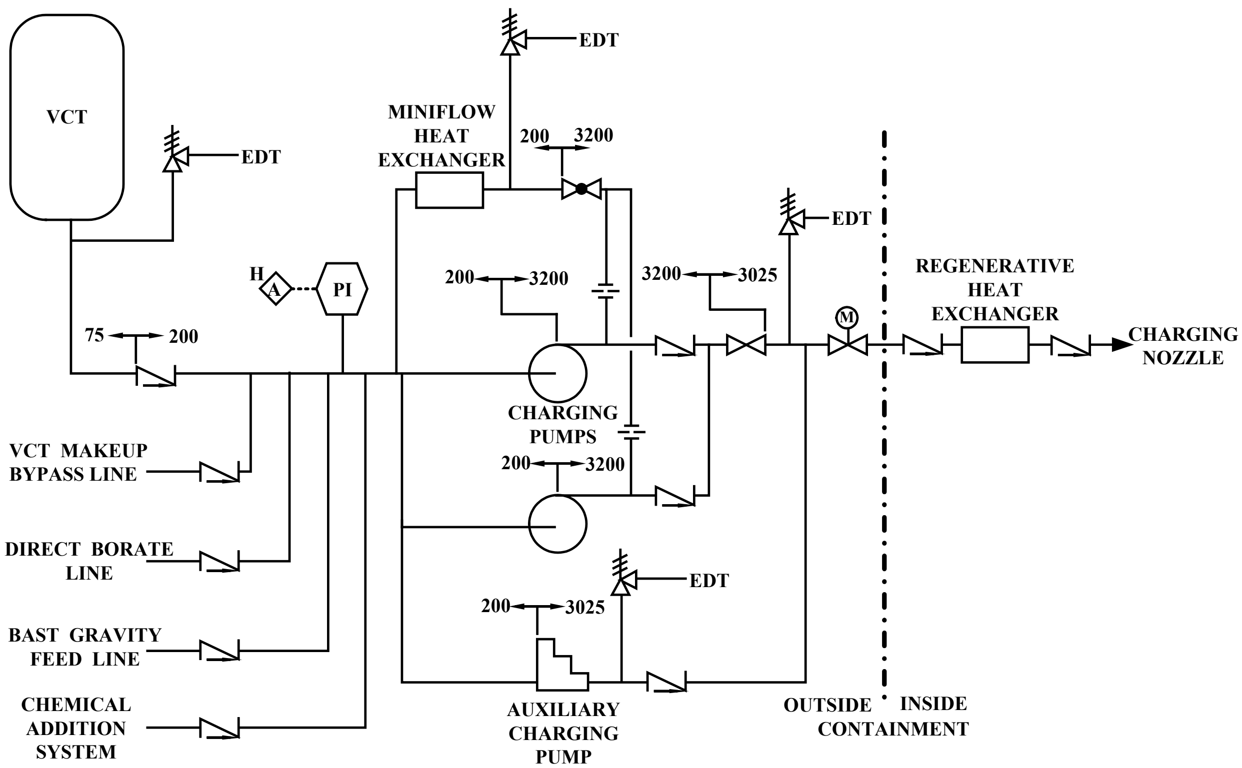

Figure 5A-14 Charging Line (including Pressure Alarm)

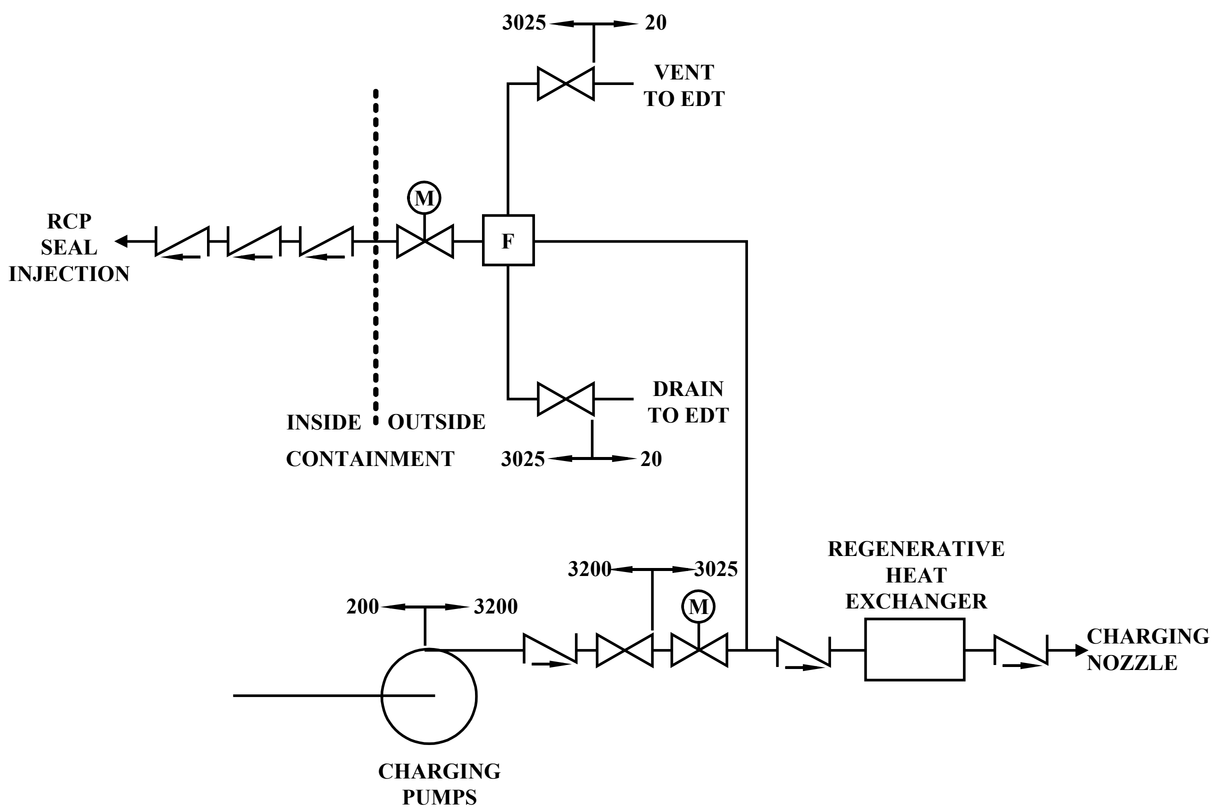

Figure 5A-15 Seal Injection Line

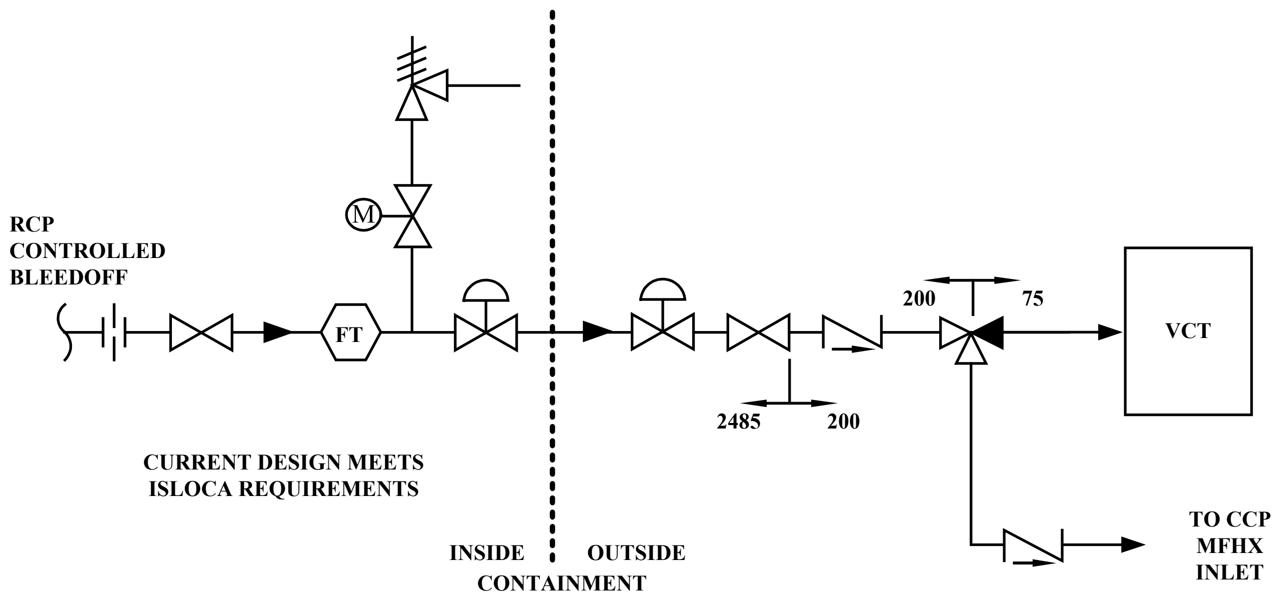

Figure 5A-16 Controlled Bleedoff Line

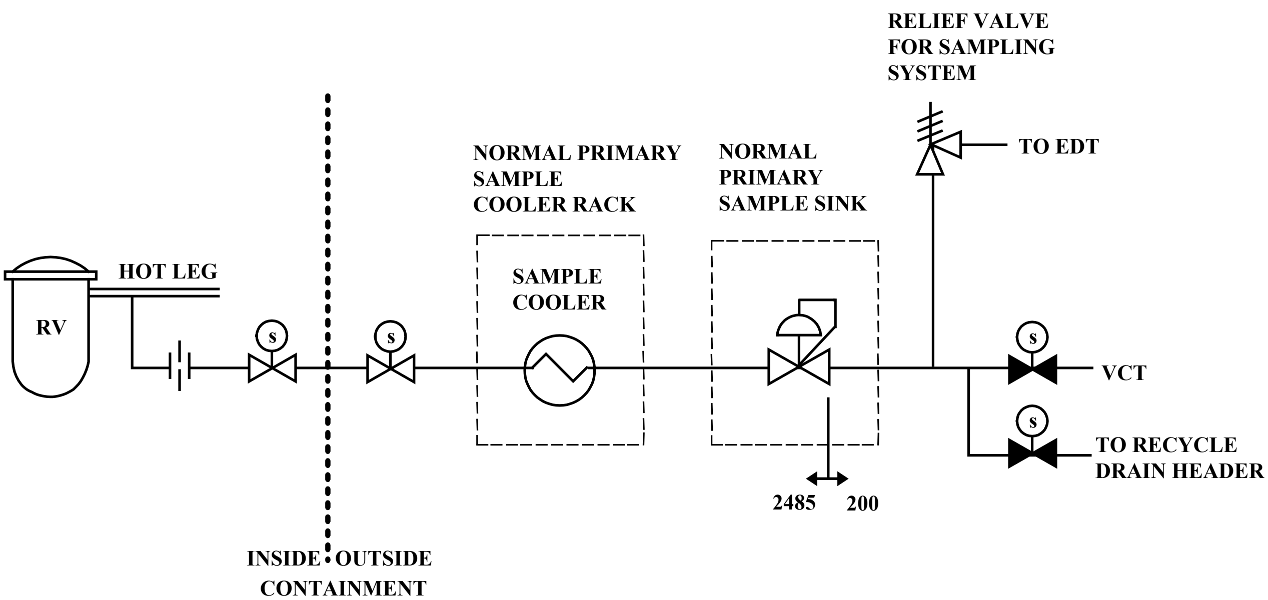

Figure 5A-17 Sampling System – Hot Leg

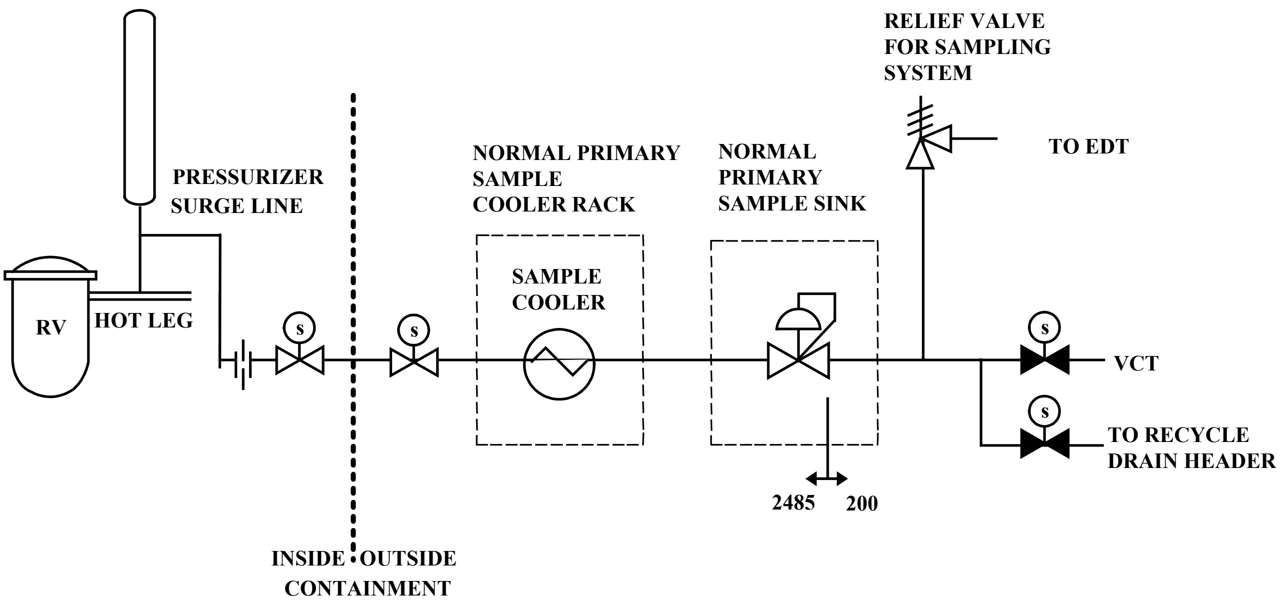

Figure 5A-18 Sampling System – Pressurizer Surge Line

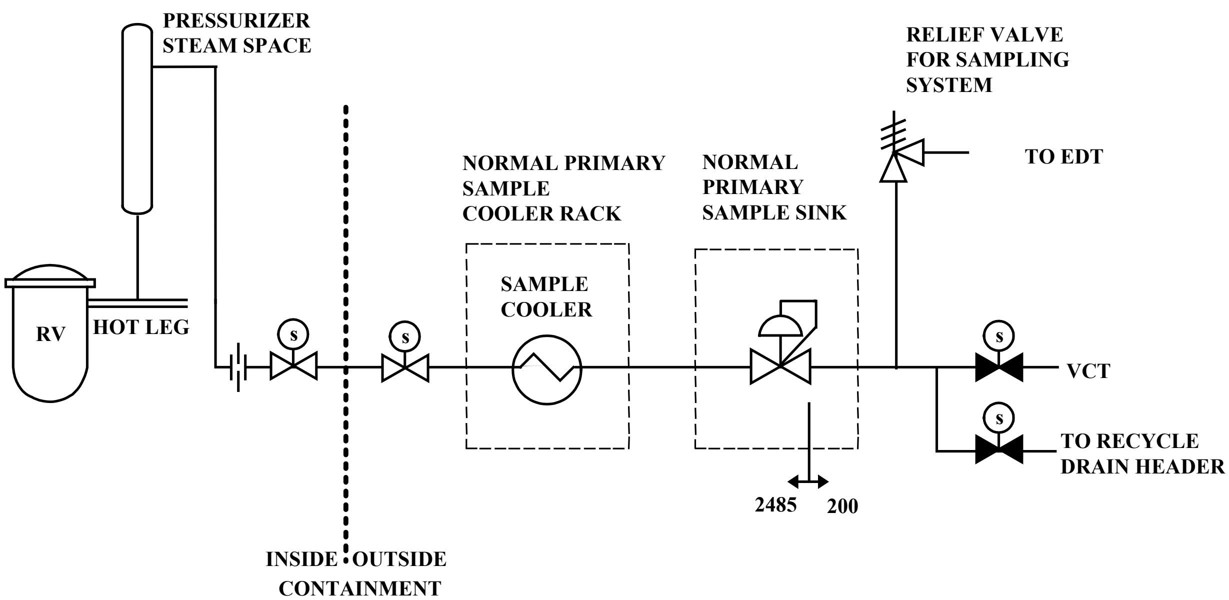

Figure 5A-19 Sampling System – Pressurizer Steam Space