ENCLOSURE 1
FINAL MARK-UPS FOR CHAPTER 5 AND CONFORMING CHANGES FOR FSAR REVISION

Description of Change	Pages Affected
Changes associated with Design Change 113-7006837-000 addition of Radiation Shield Door	Page 5.1-8 > Figure 5.1-2 (SUNSI)
Editorial changes to clarify issues identified as part of the Chapter 5 Freeze process.	Page 5.1-12 > Figure 5.1-4 Page 5.2-8 Page 5.2-43 > Table 5.2-3 Pages 5.3-1, 5.3-9 Page 5.3-15 > Table 5.3-1 Pages 5.4-1, 5.4-22, 5.4-24, 5.4-26, 5.4-28, 5.4-34, 5.4-43 Page 5.4-62 > Table 5.4-7
Conforming Chapter 11 changes that were identified during the March 5, 2014 Chapter 11 Public Meeting.	Pages 5.2-31, 5.2-32, 5.2-33
Change associated with Corrective Action to resolve incorrect references to Topical and Technical Reports.	Pages 5.3-8, 5.3-14
Change associated with Corrective Action to conform the System, Structure, and Component information provided in Tier 1 vs. Tier 2. Added new FSAR Table 5.3-8 – Acceptable RPV Key Dimension Variations to provide supporting information for existing FSAR Tier 1, Table 2.2.1-6.	Page 5.3-2 Page 5.3-22 > Table 5.3-8
Change to make the Extended Loss of AC Power (ELAP) acronym consistent.	Page 5.4-2
Editorial changes to Section 5.4.7 to clarify the language concerning cool-down references.	Pages 5.4-26, 5.4-34, 5.4-35, 5.4-36, 5.4-37 Page 5.4-67 > Figure 5.4-3 Page 5.4-68 > Figure 5.4-4 Page 5.4-77 > Figure 5.4-13 Page 5.4-78 > Figure 5.4-14 Page 5.4-79 > Figure 5.4-15 Page 5.4-80 > Figure 5.4-16 Page 5.4-81 > Figure 5.4-17 Page 5.4-82 > Figure 5.4-18
RAI 579 Supplement 1 Final Response Conforming changes to other Chapters.	Page 5.1-9 > Figure 5.1-3 Pages 5.4-27, 5.4-30, 5.4-31, 5.4-32, Page 5.4-83 > Figure 5.4-19 Tier 1 > Pages 2.2-65, 2.2-86, 2.4-54, 2.4-55, 2.4-183, 2.4-184, 2.4-191 Tier 2 > Pages 1.8-22, 7.1-153, 7.1-154, 7.1-171, 7.1-183, 7.1-184, 7.3-31, 7.3-32, 7.3-63, 7.3-121, 7.6-3, 7.6-4, 7.6-23, 7.6-24, 7.6-27, 14.2-50, 14.2-51 Page 3.3.2-17, 3.4.7-3, 3.4.8-2, 3.5.8-1, B3.3.2-4, B3.5.8-2, B3.9.5-3, B3.9.5-4, B3.9.7-1, Page 19.1-138, 189, 194, 196, 751, 841

Figure 5.1-2—RCS Layout

Figure 5.1-3—RCS Elevation

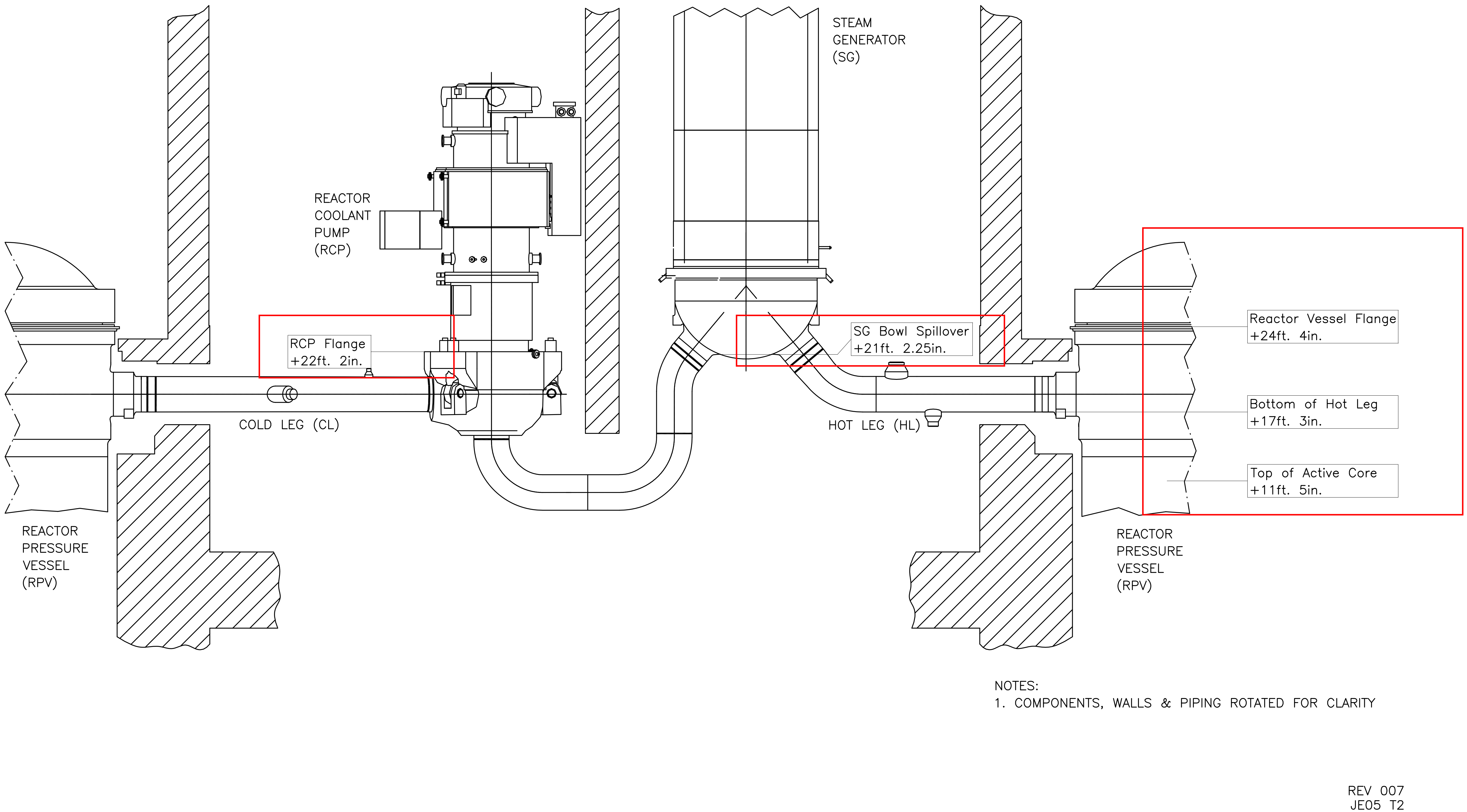
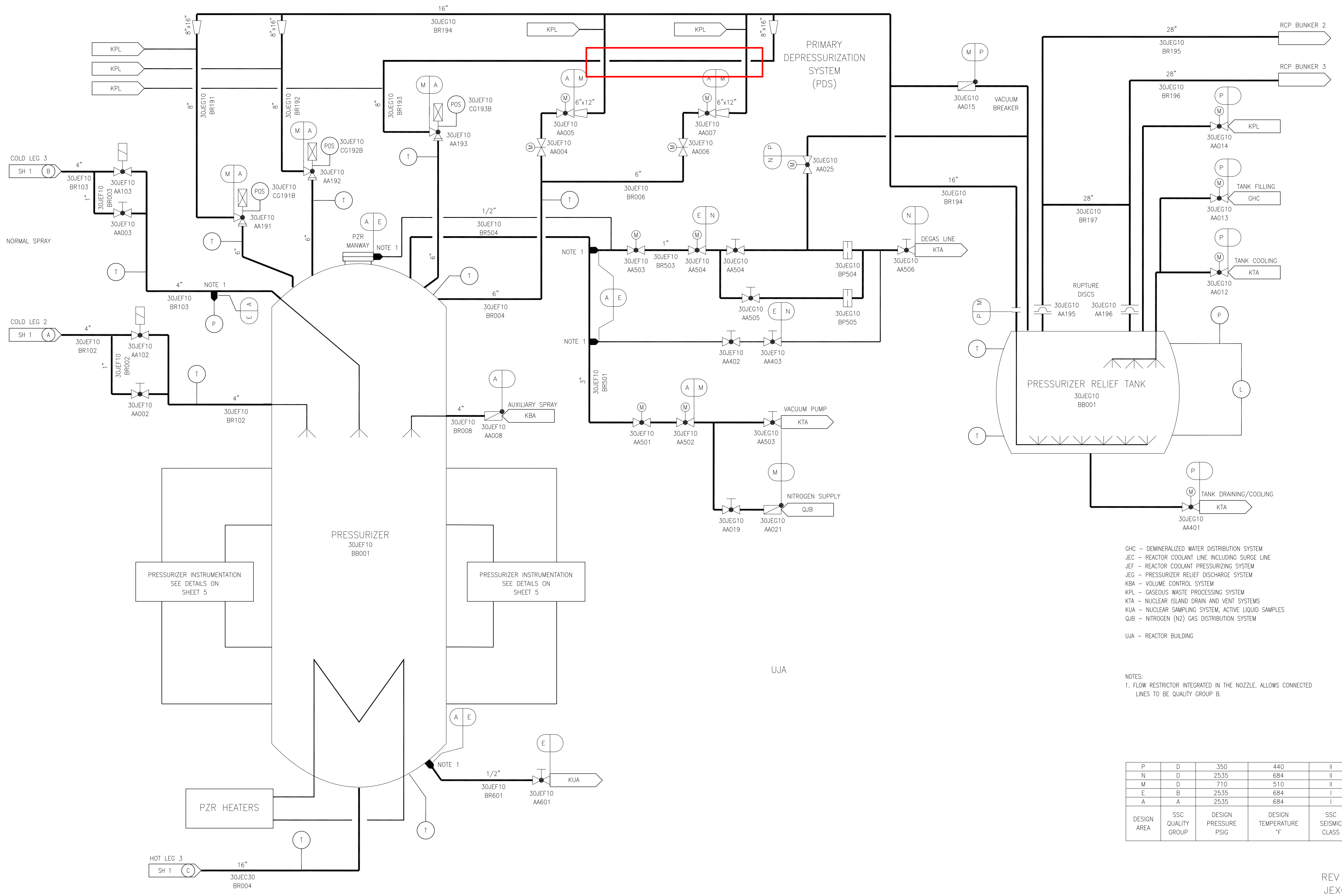


Figure 5.1-4—RCS Piping and Instrumentation Diagram
Sheet 3 of 7



GHC – DEMINERALIZED WATER DISTRIBUTION SYSTEM
JEC – REACTOR COOLANT LINE INCLUDING SURGE LINE
JEF – REACTOR COOLANT PRESSURIZING SYSTEM
JEG – PRESSURIZER RELIEF DISCHARGE SYSTEM
KBA – VOLUME CONTROL SYSTEM
KPL – GASEOUS WASTE PROCESSING SYSTEM
KTA – NUCLEAR ISLAND DRAIN AND VENT SYSTEMS
KUA – NUCLEAR SAMPLING SYSTEM, ACTIVE LIQUID SAMPLES
QJB – NITROGEN (N2) GAS DISTRIBUTION SYSTEM

UJA – REACTOR BUILDING

NOTES:
1. FLOW RESTRICTOR INTEGRATED IN THE NOZZLE. ALLOWS CONNECTED LINES TO BE QUALITY GROUP B.

DESIGN AREA	SSC QUALITY GROUP	DESIGN PRESSURE PSIG	DESIGN TEMPERATURE °F	SSC SEISMIC CLASS
P	D	350	440	II
N	D	2535	684	II
M	D	710	510	II
E	B	2535	684	I
A	A	2535	684	I

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probability of rapidly propagating fracture and gross rupture of the RCPB. (GDC 14, GDC 31)

RCPB materials are handled, protected, stored, and cleaned according to recognized and accepted methods that are designed to prevent damage or deterioration. Process specifications stipulate the procedures covering these controls in compliance with 10 CFR 50, Appendix B, Criterion XIII.

5.2.3.1 Material Specifications

Table 5.2-2 lists the materials for Class 1 primary components incorporated into the design of the RCPB (excluding the reactor pressure vessel), including grade or type and final metallurgical condition. Table 5.2-2 includes the materials specified for the steam generators, PZR, RCPs, RCPB piping, and control rod drive mechanism. ASME Boiler and Pressure Vessel Code, Section II material specifications are used for materials in the RCPB, including weld materials.

The weld filler materials used for joining the base materials of the RCPB including the weld filler materials used for joining nickel-chromium-iron (NiFeCr) alloys in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Section II Part C material specifications and classifications listed in Table 5.2-2. Carbon and low alloy steel weld filler materials are limited to non-reactor coolant exposed applications.

Low alloy steel pressure boundary forgings have limited sulfur content not exceeding 0.008 wt%, (wt = weight). Clad low alloy steel pressure boundary materials have ASTM grain size 5 or finer.

Austenitic stainless steel base metal conforms to RG 1.44. Austenitic stainless steel base metal and weld metal have limited carbon content not exceeding 0.03 wt%. Austenitic stainless steel base metal and weld filler metal in contact with RCS primary coolant has limited cobalt content not exceeding 0.05 wt%. Austenitic stainless steel base metal in contact with RCS primary coolant has limited sulfur content not exceeding 0.02 wt%. When supplementary chemical analysis is performed which would be more complete than the analysis used to check the content of specific elements, the results will show that the sample contains no more than residual antimony. In addition, the carbon portion of the reactor coolant pump journal bearings will have no antimony.

Austenitic stainless steel welds in RCS piping, including surge line piping, have delta ferrite content limited to a ferrite number (FN) between 5 and 10, measured as determined by ASME Section III, NB-2433. Austenitic stainless steel weld materials for stainless steel weld~~s~~ joints in the balance of the RCPB system have delta ferrite content limited to an FN between 5 and 20, as determined by ASME Section III, NB-2433.

- Steam generator blowdown radiation monitors.
- Main steam line ^{16}N radiation monitors.

These monitors (described in Section 11.5 and Table 11.5-1) indicate in the MCR. These measurements are supplemented by process sampling and laboratory analysis.

5.2.5.3.3 Component Cooling Water System

Leakage from the RCS to the component cooling water system (CCWS) is identified by these methods:

- Radiation monitors, which detect contamination of the system, indicate and alarm in the MCR.
- Monitoring of the CCWS surge tank level and discharge flow from selected components. Surge tank level indication is provided in the MCR.
- Leakage through the LHSI heat exchanger tubes into the CCWS is identified by temperature sensors in the heat exchanger inlet and outlet piping, which indicate and alarm in the MCR.
- Leakage from the RCP thermal barriers to the CCWS is detected by pressure, temperature, and flow sensors downstream of the barriers, which indicate and alarm in the MCR. In the unlikely event of a thermal barrier tube rupture, CCWS flow to the thermal barrier automatically isolates.
- Leakage from the letdown line heat exchangers to the CCWS is detected by radiation monitors and flow sensors which indicate and alarm in the MCR. In the unlikely event of a tube rupture, CCWS flow to the letdown line heat exchanger automatically isolates.

These methods are supplemented by radiation monitors, process sampling, and laboratory analysis, which indicate increased CCWS system activity from small leaks. Section 9.2.2 and Section 11.5 further address the control of RCS leakage into the CCWS.

5.2.5.4 Inspection and Testing Requirements

The leakage detection systems are designed to permit operability testing and calibration during plant operation. Refer to Chapter 16 (SR 3.4.14) for surveillance requirements. Periodic testing of the floor drainage system verifies that it is free of blockage.

5.2.5.5 Instrumentation Requirements

The leakage detection systems provide data to the instrumentation and control systems for indication, alarm, and archival. Operators in the MCR are provided with the leakage rate (gpm) from each detection system and a common leakage equivalent (gpm) from both identified and unidentified sources. Alarms indicate that leakage has exceeded predetermined limits. The instrumentation system is described in Section 7.1. A COL applicant that references the U.S. EPR design certification will develop procedures in accordance with RG 1.45, Revision 1.

5.2.5.5.1 RCDT Indications

The RCDT collects continuous flow during operation from PZR degassing and the RCP seals' leakoff. This flow is quantified from tank level and pump run time indications and a baseline normal in-leakage rate is established. Changes in this rate indicate leakage from additional components whose discharge is routed to the RCDT. Such leakage can be identified through indications from these components and, once quantified, can be monitored as identified leakage.

The additional monitored leakage connections that discharge to the RCDT include the PSRV valve body drains, the reactor vessel O-ring seal leakoff, RCP static seal (main flange) leakoff, and safety valve discharge lines from the combined RCP #1 seal return line, the four RCP thermal barrier return lines, the CVCS letdown line, and the CVCS charging line. Additional equipment and component drain connections to the RCDT are used only during shutdown or during startup operations and are isolated from the RCDT by a closed manual valve, or are disconnected and flanged, during power operation and are not expected to affect RCPB leakage monitoring efforts.

5.2.5.5.2 Reactor Building Sump Level

During normal operation the Reactor Building sump collects water from the reactor building floor drains and the Reactor Building annular space floor drain sump. Sump level and automatic pump operation for both sumps are indicated in the MCR to allow prompt identification of any unidentified leakage in the Reactor Building.

5.2.5.5.3 Containment Atmosphere Particulate Radiation Monitoring

Containment atmosphere particulate radioactivity monitoring is one of the systems used in the US EPR design for RCS leakage detection. The particulate monitor is a low range monitor capable of detecting 3E-10 to 1E-6μCi/cc (Refer to Section 11.5.4.8 and Table 11.5-1~~, R 10~~). The monitor sensitivity requirement is to be able to detect a leakage increase of one gpm within one hour (see U.S. EPR FSAR, Tier 2, Chapter 16, TS 16.3.4.12 and corresponding Bases, RG 1.45 and RIS-2009-02), based on a realistic RCS source term, as described in Section 11.5.4.8. The particulate radiation monitoring system continuously monitors airborne radioactivity in the containment equipment area. Radiation levels are indicated in the MCR. Alarms alert the operators of elevated levels of radioactivity to allow for prompt identification of RCS leakage into the equipment area. The monitor is located in the service area of the containment, which is accessible during normal operation. It draws air from the containment building internal filtration subsystem (KLA-5) of the containment building ventilation system described in Section 9.4.7.2, which filters airborne radioactivity within the equipment area. The monitoring system will be designed to function properly in the containment environment.

5.2.5.5.4 Main Steam Line Radiation Monitors for Steam Generator Tube Leakage

The primary instruments for quantification of primary-to-secondary leakage during normal operation are the safety-related main steam line radiation monitors. These monitors measure the concentration of radioactive materials in the four main steam lines (N-16 and noble gases) and provide early indication of steam generator tube leakage. There are four redundant measuring arrangements for each of the four main steam lines, with a total of 16 monitors mounted adjacent to the steam lines within the main-steam and feedwater valve compartments.

The main steam line radiation monitors will be of high sensitivity, with each detector placed within specially designed lead shielding that would limit the angle of view to the steam line being monitored. Such an arrangement minimizes the contribution of scatter radiation as well as direct radiation emanating from the adjacent steam lines. The monitors are capable of satisfying the technical basis of the primary to secondary maximum leakage rate of 150 gallons per day using realistic RCS radioactivity concentrations. The required monitor sensitivity range was determined to be 1.0E-08 to 1.0E-02 μCi/cc of N-16. The steam generator leakage determination is based on correlations which predict the leakage rate as a function of primary-coolant N-16 concentration, power level and monitor reading, taking into consideration the variation of steam flow with power and the N-16 in-transit decay from the leakage point to the radiation monitor location.

In the presence of N-16, noble gas activity in the main steam lines has minimal contribution to the monitor response. The same monitors (refer to Section 11.5.4.1) are used for identification of the affected steam generator in a steam generator tube rupture (SGTR) event, based on the ensuing noble-gas activity within the steam line.

5.2.6 References

1. ASME Boiler and Pressure Vessel Code, Section III," Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, 2004.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants, The American Society of Mechanical Engineers, 2004.

3. EPRI Report 1014986, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Volume 1, Revision 6, Electric Power Research Institute, December 2007.

Table 5.2-3—Reactor Coolant Water Chemistry - Control Parameters

Control Parameter	~~Normal Operating Conditions~~Control Ranges and Impurity Limitations During Power Operations
Lithium (pH control)	0.39 to 4.0 mg/kg
Hydrogen	25 to 50 cc(STP)/kg (2.2 to 4.5 mg/kg)
Dissolved Oxygen	< ~~0.005~~0.100 mg/kg
Chloride	< ~~0.010~~0.150 mg/kg
Fluoride	< ~~0.010~~0.150 mg/kg
Sulfate	< ~~0.010~~0.150 mg/kg
Total Boron and Boron 10	As required for reactivity control

Next File

5.3 Reactor Vessel

The reactor pressure vessel (RPV) and closure head form what is the enclosure that contains the reactor core. The RPV holds the internals that support the fuel assemblies and that direct the reactor coolant flow through the reactor core. Eight nozzles provide inlet and outlet connections to the four reactor coolant system (RCS) loops. The general design of the RPV is described in Section 5.3.3.1.

5.3.1 Reactor Vessel Materials

The RPV is part of the reactor coolant pressure boundary (RCPB) and is designed and constructed to meet the requirements for ASME Boiler and Pressure Vessel Code Section III (Reference 1), Class 1 components, in accordance with 10 CFR 50.55(a). The RPV materials are selected, designed and constructed to minimize the probability of significant degradation or rapidly propagating fractures in the RPV (GDC 1, GDC 14 and GDC 30).

As addressed in Section 5.3.3.1, the RPV provides support for internal reactor components and is designed to accommodate the effects of environmental conditions associated with normal operations, maintenance, testing, postulated accidents and anticipated operational occurrences (AOO) as defined by GDC 4. Section 3.9 identifies the design transients for which the RPV is designed.

The RPV meets the fracture toughness requirements of 10 CFR Part 50, Appendix G and those associated with ASME Section III, Class 1 components (10 CFR 50.60). The ferritic materials provide sufficient margin to account for uncertainties associated with flaws and the effects of service and operating conditions, while allowing the vessel to behave in a non-brittle manner and minimizing the probability of rapidly propagating fracture (GDC 31).

An RPV material surveillance program monitors the RPV beltline materials for changes in fracture toughness resulting from exposure to neutron irradiation and the thermal environment (GDC 32). The program complies with 10 CFR Part 50, Appendix H, as described in Section 5.3.1.6 (10 CFR 50.60).

Material cleaning control for the RPV conforms to RG 1.37 and meets the quality assurance requirements of 10 CFR Part 50, Appendix B, Criterion XIII.

5.3.1.1 Material Specifications

The RPV is made of low-alloy steel due to its mechanical and physical properties, toughness, availability in the required sizes and thicknesses, satisfactory prior service in neutron fields, fabricability, and weldability. The low-alloy steel is also compatible with the stainless steel and Ni-Cr-Fe alloy cladding used for corrosion resistance. The austenitic stainless steels and non-ferrous materials used for RPV appurtenances are used for their corrosion resistance, acceptable mechanical properties, and fabricability. The RPV surfaces normally in contact with the reactor coolant are either austenitic stainless steel or Ni-Cr-Fe alloy. A listing of material specifications for the RPV and its appurtenances is provided in Table 5.3-1—Reactor Pressure Vessel Material Specifications and Table 5.3-2—Reactor Pressure Vessel Weld Material Specifications. The RPV materials meet the requirements of the ASME Section III and comply with fracture toughness requirements of 10 CFR Part 50, Appendix G as addressed in Section 5.3.1.5.

The weld filler materials used for the reactor vessel conform to ASME Section II Part C material specifications SFA 5.4, 5.9, 5.11, 5.14, 5.22, 5.23, 5.28, 5.29, and 5.30.

The shell forgings of the RPV beltline are restricted to the maximum composition limits shown in Table 5.3-3—Maximum Limits for RPV and Appurtenances Material Composition. The phosphorous, nickel, and copper content is limited to reduce sensitivity to radiation embrittlement of the vessel. The weld filler metals used in the beltline region of the RPV are restricted to the limits shown in Table 5.3-3.

Stainless steel normally in contact with the reactor coolant has a maximum cobalt content of 0.05 wt percent. Stainless steel base and weld filler materials have a limited carbon content not exceeding 0.03 wt percent and are supplied in accordance with RG 1.44. Stainless steel base and weld filler metal in contact with the reactor coolant has a limited sulfur content, as shown in Table 5.3-3.

The Ni-Cr-Fe Alloy 600 base metal and Alloy 82/182 weld filler metal are not used in Ni-Cr-Fe applications. Alloy 690 base metal and Alloy 52/52M/152 weld filler metal are used in Ni-Cr-Fe applications. The Ni-Cr-Fe base metal in contact with the reactor coolant has a limited sulfur content not exceeding 0.02 percent.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The RPV is a vertically mounted cylindrical vessel consisting of forged shells, heads, and nozzles joined by circumferential welds. The design of the RPV is addressed in Section 5.3.3.

The RPV is fabricated in accordance with ASME Section III, NB-4000 and RPV materials comply with the requirements of ASME Section III, NB-2000.

The internal surfaces of the RPV low alloy steel that could come into contact with the reactor coolant are clad using weld metal overlay primarily with stainless steel, with the exception of the areas where Alloy 690 radial keys are to be welded, as shown in Table 5.3-8 and on Figure 5.3-4. For these areas, Ni-Cr-Fe weld filler materials are used to clad the low alloy steel to minimize dissimilar materials in the overall RPV pressure boundary, clad, attachment configuration. The cladding at all interfaces the requirements of 10 CFR Part 50, Appendix G. The analysis follows the guidance provided in RG 1.99 and the methodology of ASME Section XI (Reference 11), Appendix G. From the analyses, curves are developed that specify pressure-temperature limits to envelop plant operation for 60 years (conservatively considering 60 EFPY).

ANP-10283P-A, Revision 2, "US EPR Pressure-Temperature Limits Methodology for RCS Heat-Up and Cool-Down" (Reference 12), contains the detailed methodology for developing the P-T limit curves. The P-T limits are revised as necessary, for various reasons including refined end of life (EOL) fluence estimations obtained from the material surveillance program, in accordance with the recommendations of RG 1.190. Testing of each surveillance capsule will be performed in accordance with 10 CFR 50, Appendix H. The material data will be evaluated using the guidance of RG 1.99. The P/T limits will be recalculated or the applicable EFPY will be adjusted, as necessary, to confirm that the 1/4T and 3/4T adjusted RT_{NDT} of the RPV based P/T limits is not exceeded. The initial RT_{NDT}, final predicted RT_{NDT} or adjusted reference temperature (ART), and the copper and nickel contents for materials in the RPV beltline are provided in Table 5.3-3 and Table 5.3-4. The fluence attenuation to the 1/4T and 3/4T locations and the ART values are calculated per RG 1.99, Revision 2.

Generic heatup and cooldown curves for the U.S. EPR RPV design are provided in Figure 5.3-1—Reactor Coolant System Heatup Pressure-Temperature Curve and Figure 5.3-2—Reactor Coolant System Cooldown Pressure-Temperature Curve, and are based on limiting vessel material properties. A COL applicant that references the U.S. EPR design certification will provide a plant-specific pressure and temperature limits report (PTLR), consistent with an approved methodology.

5.3.2.2 Operating Procedures

Plant operating procedures provide reasonable assurance that the P-T limits identified in Section 5.3.2.1 will not be exceeded during conditions of normal operation, AOOs and system hydrostatic tests. The transient conditions considered in the design of the RPV, as presented in Section 3.9.1.1, are representative of the operating conditions considered to occur during plant operation. The selected transients form a conservative basis for evaluation of the RCS and do not result in pressure-temperature changes that exceed the heatup and cooldown rate limits used in the development of the Pressure-Temperature Limit curves of Section 5.3.2.1.

5.3.2.3 Pressurized Thermal Shock

The RPV design provides protection against unstable crack growth under faulted conditions. A safety injection actuation following an emergency or faulted event produces relatively high thermal stresses in regions of the RPV contacting the cooler water from the safety injection system. Consideration is given to these areas, including the beltline region and the RPV nozzles, which provide reasonable assurance of RPV integrity under these postulated transients.

An analysis was performed to determine the RPV pressurized thermal shock reference temperatures (RT_{PTS}) applicable to 60 EFPY. The RT_{PTS} values were conservatively calculated for various RPV materials over 60 EFPY with the most limiting core design. These values, calculated in accordance with 10 CFR 50.61 and presented in Table 5.3-4, do not exceed the screening criteria. A COL applicant that references the U.S. EPR design certification will provide plant-specific RT_{PTS} values in accordance with 10 CFR 50.61 for vessel beltline materials.

5.3.2.4 Upper-Shelf Energy

The minimum Charpy upper-shelf energy values for RPV beltline materials, which meet the requirement of paragraph IV.A.1.a of Appendix G, are specified in Section 5.3.1.5.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Design

The RPV and closure head form the enclosure which contains the reactor core. The vessel holds the internals that support the fuel assemblies and that direct the reactor coolant flow through the reactor core. Eight nozzles provide inlet and outlet connections to the four reactor coolant system (RCS) loops. The RPV design data is given in Table 5.3-7—Reactor Pressure Vessel Design Data.

The closure head is attached to the RPV with a stud-nut-washer set. The joint between the RPV and the closure head is sealed by two seals located in concentric, circular recesses on the head flange. The closure head can be removed for refueling and vessel maintenance.

The control rod drive mechanisms (CRDM) are installed on top of the closure head. They are affixed to adapters welded to the RPV head. Instrumentation adapters are mounted to the vessel head via welded adapter penetrations to monitor the core temperature and neutron flux.

Section 5.3.1 identifies the regulations with which the RPV design complies, including GDC 1, GDC 14, GDC 30, GDC 31, GDC 32, 10 CFR 50.55a, 10 CFR 50.60 and 10 CFR Part 50, Appendix G. Component classifications are identified in Section 3.2.

The RPV consists of the following forged components, as shown in Figure 5.3-4—Reactor Pressure Vessel:

- Closure Head Assembly:

3. SRM-SECY-04-0032," Programmatic Information Needed For Approval of Combined License Without Inspections, Test, Analyses and Acceptance Criteria," Secretary of the Commission, U.S. Nuclear Regulatory Commission, 2004.

4. ASTM E185-02, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," American Society for Testing and Materials, 2002.

5. ASTM A370-07a, "Standard Test Methods and Definitions for Mechanical Testing of Steel Products," American Society for Testing and Materials, June 2007.

6. ASTM E23-07ae1, "Standard Test Methods for Notched Bar Impact Testing of Metallic Materials," American Society for Testing and Materials, 2007.

7. ASTM E08-04, "Standard Test Methods for Tension Testing of Metallic Materials," American Society for Testing and Materials.

8. ASTM E1921-05, "Standard Test Method for Determination of Reference Temperature, To', for Ferritic Steels in the Transition Range," American Society for Testing and Materials, 2005.

9. BAW-2241P-A-002, "Fluence and Uncertainty Methodologies," AREVA NP Inc., April 2006.

10. NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation of Failure in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1990.

11. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," The American Society of Mechanical Engineers, 2004.

12. ANP-10283P-A, Revision 2, "U.S. EPR Pressure-Temperature Limits Methodology for RCS Heat-Up and Cool-Down," AREVA NP Inc., August 2012.

Table 5.3-1—Reactor Pressure Vessel Material Specifications

Component	Material Specification
Lower head	ASME SA-508 Grade 3 Class 1
Transition ring	ASME SA-508 Grade 3 Class 1
Upper/lower core shells	ASME SA-508 Grade 3 Class 1
Nozzle/Flange integrated shell	ASME SA-508 Grade 3 Class 1
Radial Keys	ASME SB-564 Alloy 690 (UNS N06690)
Safe ends for inlet and outlet nozzles	ASME SA-182 Grade F316 (See Notes 1 & 2) or ASME SA-336 Grade F316 (See Notes 1 & 2)
Upper head	ASME SA-508 Grade 3 Class 1
Closure head flange	ASME SA-508 Grade 3 Class 1
Head studs/nuts/washers	ASME SA-540 Grade B24V (4340V Mod) Class 3
CRDM adapter tubes	ASME SB-167 Alloy 690 (UNS N06690)
CRDM adapter flanges	ASME SA-182 Grade F304 (See Notes 1 & 2) or ASME SA-336 Grade F304 (See Notes 1 & 2)
Instrument adapter tubes	ASME SB-167 Alloy 690 (UNS N06690)
Instrument adapter flanges	ASME SA-182 Grade F304 (See Notes 1 & 2) or ASME SA-336 Grade F304 (See Notes 1 & 2)

Notes:

1. Solution annealed and rapidly cooled.
2. Carbon content not exceeding 0.03 wt%.

Table 5.3-8—Acceptable RPV Key Dimension Variations

Description	Nominal Value (inches)	Acceptable Variation (inches)
Vessel Inside Diameter (in.) (to cladding)	191.73	+1.0/-1.0
Vessel Beltline Shell thickness (in.) without cladding	9.84	+0.88/-0.12
Vessel Lower Head thickness (in.) without cladding	5.71	+1.0/-0.12
Vessel Inlet/Outlet Nozzle Inside Diameter (in.) at Safe End (in.)	30.71	+0.37/-0.12
Elevation from Mating Surface to Centerline of Inlet/Outlet Nozzle (in.)	70.87	+0.25/-0.25
Elevation from Mating Surface to Inside of Bottom Head (to cladding)	408.66	+1.0/-0.5

5.4 Component and Subsystem Design

This section presents the design bases, descriptions, and evaluations of the primary RCS components. The components of the RCS are designed to operate in the environment present when the reactor is at power. Components important to safety are designed to perform their safety functions in an environment degraded by a design basis accident. Section 3.11 provides a detailed list of components in the RCS and their environmental qualifications. Radiological considerations including radiation effects on operations and maintenance are provided in Chapter 12.

5.4.1 Reactor Coolant Pumps

The reactor coolant pumps (RCP) provide forced flow circulation of the reactor coolant to transfer heat from the reactor core to the steam generators (SG). The RCPs form part of the reactor coolant pressure boundary (RCPB) during all modes of operation, thereby retaining the circulated reactor coolant and entrained radioactive substances. Figure 5.4-1—Reactor Coolant Pump and Subassemblies, shows a typical RCP assembly.

5.4.1.1 Design Bases

The RCPs provide adequate forced flow circulation during normal operation and anticipated operational occurrences (AOO), except during loss of power. The design flow rate of the RCP is based on the thermal hydraulic considerations established in Section 4.4.

The RCPs are part of the RCPB and are designed, fabricated, erected and tested so as to have an extremely low likelihood of abnormal leakage, of rapidly propagating failure, and of gross rupture (GDC 14). In the event of station blackout (SBO) or extended loss of AC power (ELAP), the integrity of the RCP pressure boundary is maintained to prevent unacceptable RCPB leakage.

The RCPs are designed, fabricated, erected, and, tested to quality standards commensurate with the safety-related functions to be performed. Section 3.2 identifies component classifications (GDC 1, 10 CFR 50.55a(a)(1)).

5.4.1.2 Design Description

A typical RCP assembly is shown in Figure 5.4-1. The RCP design data are given in Table 5.4-1—Reactor Coolant Pump Design Data. The RCP materials are given in Section 5.2.3. The RCP supports are described in Section 5.4.14.

The RCPs are constructed to ASME Section III of the Boiler and Pressure Vessel Code (Reference 1). The RCP pressure boundary components and supports are constructed to ASME Section III, Class 1 requirements. The RCP design stress limits, transient conditions, and combined loading conditions are described in Section 3.9.

There are four identical RCP assemblies used in the U.S. EPR plant design. The RCPs are centrifugal, single stage pumps with mechanical shaft seals driven by squirrel-cage type induction motors as shown in Figure 5.4-1. Each RCP assembly has one common vertical shaft line for the pump and motor with main and auxiliary bearings, one single double thrust bearing and a flywheel located at the top of the motor shaft.

5.4.1.2.1 Pumps

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

The shaft seal system is made up of a series of three seals and a standstill seal. The shaft seal design provides redundancy so that a failure of a single seal stage will not result in an uncontrolled loss of reactor coolant. The standstill seal is a metal-to-metal contact seal that prevents leakage when the RCP has stopped and the three seal leak-off lines have been isolated. The standstill seal is normally used under these conditions:

- In the event of a concurrent loss of injection water (chemical and volume control system (CVCS)) and cooling water for the thermal barrier (component cooling water system (CCWS)).
- In the event of concurrent failure of all three shaft seals.
- In the event of a station blackout.
- In the event of an ELAP.

The standstill seal is activated by compressed nitrogen and is designed to stay closed if the gas pressure is lost, and to remain closed and maintain RCS pressure boundary integrity down to an RCS pressure of approximately 218 psia. If the nitrogen pressure is maintained on the standstill seal, it will maintain RCS pressure boundary integrity down to zero RCS pressure. Position indication is provided for the standstill seal. Standstill seal operability is maintained after a safe shutdown earthquake (SSE).

The temperature of the water within the RCP shaft seal assembly is normally maintained within acceptable limits by seal injection water supplied from the CVCS. Water from the CVCS is injected into the cavity upstream of the number 1 seal, at a pressure slightly higher than the reactor coolant, through a connection on the thermal barrier flange. A portion of this water descends through the thermal barrier heat exchanger (HX) and auxiliary bearing into the RCP casing, while the other portion rises through the shaft seal system.

If seal injection from the CVCS is lost, cooled water from the RCS will enter the shaft seal assembly. The hot RCS water is cooled by the thermal barrier HX before entering the RCP shaft seal assembly. The thermal barrier HX is sized to cool the RCS water below the maximum acceptable shaft seal injection water temperature. Cooling water

Such procedures include these as a minimum:

- Detailed accountability for tools and equipment used during any activity when the primary or secondary side is open.
- Appropriate controls and accountability for foreign objects such as eyeglasses and personnel dosimetry.
- Cleanliness requirements.
- Accountability for components and parts removed from the internals of major components (e.g., reassembly of cut and removed components).

Similar procedural requirements are established to prevent introduction of foreign objects during maintenance evolutions performed in other portions of the plant which could ultimately affect SG integrity. In particular, consideration is given to the potential for introduction of loose parts or foreign objects from secondary side systems.

5.4.2.5.2.10 Contractor Oversight

The licensee performs oversight of contracted work. When the licensee contracts potions of the SG program work scope, the responsibility for program implementation remains with the licensee. Guidance on contractor oversight can be found in NEI 97-06 (Reference 10).~~Additional guidance on contractor oversight can be found in the applicable EPRI steam generator guidelines, listed in Section 5.4.15, which govern the activity.~~

5.4.2.5.2.11 Self Assessment

Self assessments of the SG program are performed periodically by knowledgeable personnel. The assessment, or a combination of assessments, includes the major program elements as described Section 5.4.2.5.2.

5.4.2.5.3 Reporting

A report is submitted to the NRC within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the SG program. The report includes:

- The scope of the inspection performed on each SG.
- Active degradation mechanisms.
- Non-destructive examination techniques utilized for each degradation mechanism.
- Location, orientation (if linear) and measured sizes (if available) of service-induced flaw indications.

parameters for the RCS piping.

Each RCS loop contains three sections of piping:

- The hot leg, which connects the RPV outlet nozzle to the SG inlet nozzle.
- The crossover leg, which connects the SG outlet nozzle to the RCP suction nozzle.
- The cold leg, which connects the pump discharge nozzle to the inlet nozzle of the RPV.

The main coolant lines are forged austenitic stainless steel with inductive bends and separate forged elbows. Large nozzles are machined from the pipe forging and are integral with the main coolant lines. Austenitic stainless steel safe ends on RCS components allow for homogenous field welds to the RCS piping.

Two 4-inch spray lines connect the cold legs of RCS Loops 2 and 3 to spray lances in the steam space of the pressurizer. The pressurizer surge line connects the pressurizer bottom head to hot leg of RCS loop 3. The surge line slopes continuously downward from the pressurizer to the hot leg as shown in Figure 3.6.3-3. The surge line consists of forged austenitic stainless steel pipe with inductive bends.

The post-accident high point vent is described in Section 5.4.12. The line branches into two parallel vent lines connected to the normal RPV head vent piping upstream of the first RCPB isolation valve and discharges to an SG cubicle through a flow-restricting orifice.

Thermal sleeves on the CVCS charging nozzles and on the pressurizer nozzle connected to the surge line protect the nozzles and nozzle welds from the effects of thermal shock.

5.4.3.3 Design Evaluation

The reactor coolant lines are sized to minimize the pressure drop in the primary loops and to reduce the flow velocity. The surge line is sized to limit the frictional pressure loss in the line for the maximum in-surge to provide RCS overpressure protection during normal, upset, and emergency conditions.The effects of surge line thermal phenomena (thermal stratification, vortex penetration, and thermal shock) are minimized for the U.S. EPR because of the following:

- The surge line piping slopes continuously upward from the hot leg to the pressurizer as shown in Figure 3.6.3-3.
- During phases of RCS heatup, the pressurizer to hot leg temperature difference is maintained at a minimum value to minimize thermal shock in the surge line.

- RCS loop level.
- RCS loop flow.
- RCP differential pressure.

Temperature indications in both spray lines confirm the minimum flow through the bypass valves. Surge line temperature indication confirms sufficient continuous flow between the pressurizer and the RCS.

5.4.4 Not Used in U.S. EPR Design

5.4.5 Not Used in U.S. EPR Design

5.4.6 Not Used in U.S. EPR Design

5.4.7 Residual Heat Removal System

The safety injection system / residual heat removal system (SIS/RHRS) provides an emergency core cooling function for postulated events, and removes residual heat from the core and the RCS during normal shutdown and accident conditions. This section describes the residual heat removal (RHR) function of the SIS/RHRS. The emergency core cooling function of the SIS/RHRS is discussed in Section 6.3.

5.4.7.1 Design Basis

The heat removal function of the SIS/RHRS provides cooldown of the RCS during normal shutdown operations after secondary side heat removal by the steam generators has been completed. The heat removal function maintains the reactor coolant temperature within allowable limits for refueling and maintenance activities, including mid-loop operation. The RHR function of the SIS/RHRS also provides a flow path to the chemical and volume control system (CVCS) for low-pressure purification and mixing of the reactor coolant during shutdown operations. The SIS/RHRS also fills the reactor cavity and the SIS accumulators.

The RHR function of the SIS/RHRS provides normal cooldown of the RCS during normal shutdown operations after secondary side heat removal by the SGs has become ineffective. The U.S. EPR systems are designed to reduce RCS temperature from hot standby temperature (approximately 578°F) to approximately 131°F within 40 hours when the entire SIS/RHRS is operable, and to establish conditions that allow removal of the reactor pressure vessel (RPV) head and initiation of refueling operations within approximately 90 hours after initial reactor shutdown.

The U.S. EPR systems (with the SIS/RHRS in its heat removal function) is also capable of bringing the reactor to a cold shutdown condition using only safety-related equipment, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure (BTP 5-4).

The SIS/RHRS is designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety-related functions to be performed (GDC 1, 10 CFR 50.55a(a)(1)), and to remain functional after a safe shutdown earthquake, in accordance with RG 1.29 (GDC 2). Section 3.2 identifies component classifications.

The low head safety injection (LHSI) pump suction piping from the RCS hot legs to the LHSI pump is designed to be self venting to prevent the formation of loop seals (voids) within the piping. Similarly, suction piping from the IRWST for both the LHSI and medium head safety injection (MHSI) pumps is designed to be self-venting to preclude voids within the piping when the SIS/RHRS is in the emergency core cooling function. Therefore, the entire shutdown cooling loop remains flooded when connected to the RCS, protecting the LHSI pumps from suction cavitation and the piping from water hammer (due to voiding) during shutdown cooling operations (GDC 4).

The SIS/RHRS is not shared among nuclear power units (GDC 5).

Operation of the SIS/RHRS is performed from the main control room (MCR) for all operating conditions (GDC 19). Similar operation of the SIS/RHRS can also be performed from the remote shutdown station.

The SIS/RHRS is designed to transfer fission product decay heat and other residual heat from the reactor core at a rate such that acceptable fuel design limits and the design conditions of the RCPB are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities are provided to accomplish this function assuming a single failure with only onsite or only offsite electric power available (GDC 34).

The SIS/RHRS is designed to control and detect leakage outside containment following an accident. The section of the SIS/RHRS that is located outside the containment can be isolated from the containment in the event of a break in the SIS/RHRS piping (10 CFR 50.34(f)(2)(xxvi)). Leakage from the system is detected, monitored, and controlled by plant operating procedures and programs.

SIS/RHRS is required for long-term LOCA event mitigation because it represents the safety-related means to remove heat from the containment. Automatic trip of LHSI pumps operating in RHR Mode, on Low ΔP_{Sat} signal or Low RCS Loop Level protects the LHSI pumps operating in RHR Mode during these conditions (GDC 38).

The SIS/RHRS interfaces with the RCPB. This interface is considered part of the RCPB. The SIS/RHRS provides a containment isolation function when required.

5.4.7.2 System Description

Four physically separated and independently powered RHR trains comprise the SIS/RHRS. The instrumentation and controls used to manage the operation of the SIS/RHRS are separated and derive input from independent sources for process variables such as RCS pressure and temperature. The instrumentation and controls are independently powered from the same normal and emergency sources that power the associated motive equipment. Schematic piping and instrumentation diagrams for the four SIS/RHRS trains are shown in Figure 6.3-1 through Figure 6.3-3.

The RHR function of the SIS/RHRS performs a single mode of operation with a temperature controlled variable flow rate, and operating temperatures and pressures that vary throughout its normal operating range.

The shutdown cooling loop of each SIS/RHRS train consists of an LHSI pump, LHSI HX, LHSI HX bypass line with flow control valve, LHSI HX discharge line with a temperature control valve, and the RCS hot leg suction line. Various isolation valves are also provided to support maintenance or operations activities, including realignment for shutdown or accident mitigation.

A mini-flow and test line, which are isolated during RHR, branches off of the cold leg injection line upstream of the outboard SIS/RHRS to RCS isolation valve. A CVCS letdown line, to accommodate expansion and shrinkage of the corresponding SIS/RHRS train during shutdown and startup, connects to the SIS/RHRS cold leg injection and hot leg suction lines between the RCPB isolation valves of each train.

The four SIS/RHRS trains are functionally identical except for additional CVCS letdown connections from the LHSI HX discharge and bypass lines of Trains 3 and 4.

For shutdown cooling operations, each SIS/RHRS train is aligned to take suction from the corresponding RCS hot leg, pump the hot reactor coolant through the corresponding LHSI HX where it is cooled by the corresponding CCWS train, and return the reactor coolant to the RCS through the corresponding cold leg. Interlocks (permissive P14, refer to Section 7.2.1.3.9) prevent alignment of the SIS/RHRS to the RCS while RCS pressure and temperature is greater than approximately 464 psia and 350°F. This feature protects the SIS/RHRS components from overpressure due to exposure to the RCS pressure during reactor operation (intersystem LOCA). Additional features addressing intersystem LOCA are discussed in Section 5.4.7.2.2.

The initial stage of RCS cooldown is accomplished with SG cooling as addressed in Section 10.4.7 under normal operating conditions and Section 15.2.7 under abnormal operating occurrences (AOO). Two trains of the SIS/RHRS are normally placed in service for RHR at an RCS pressure and temperature of approximately 370 psia and 250°F. The remaining two trains are placed in service after the RCS temperature has been further reduced to approximately 212°F. If main steam bypass is unavailable, the

5.4.7.2.1 Design Features Addressing Shutdown and Mid-Loop Operations

The design features of the U.S. EPR that support improved safety during shutdown and mid-loop operations, addressing NRC Generic Letter 88-17 (Reference 16) and SECY 93-087 (Reference 17), are as follows:

- Inherent redundancy in the design of the four divisions of safety-related U.S. EPR SIS/RHRS, with each train having separate RCS connections.
- Automatic SIS Actuation (Protection System) ~~and automatic stop of the LHSI pumps in RHR mode (PAS)~~ in the event of a low loop level or low ~~delta~~Δ~~-~~P_{sat}~~. (difference between the RCS hot leg temperature and the RCS hot leg saturation temperature).~~ See Figure 7.3-2—SIS Actuation.
  - Automatic safety injection via MHSI with reduced discharge head during low loop level provides RCS makeup in the event of inadvertent draining of the RCS as described in Section 6.3.1. Operability of MHSI is controlled by Technical Specification 3.5.8.
  - Stage 1 Containment Isolation - isolates CVCS letdown.
- Manual opening and closure of the RHR suction isolation valves (in addition to interlocks) prevent unwanted RHR connection or isolation on irregular RCS pressure. See Figure 7.6-11—RHR Isolation Valves Interlock.
- Safety Related (SAS) automatic stop of the LHSI pumps in RHR mode in the event of a low loop level or low ΔP_{sat} prevents cavitation of operating LHSI pumps due to loss of suction or steam entering the system.
  - Automatic safety injection via MHSI in response to a low loop level or low ΔP_{sat} provides RCS makeup and allows time to re-establish RHR/LHSI heat exchanger flow to reject heat from the RCS and Containment to the plant cooling systems operating in RHR mode or in injection mode using feed and bleed cooling through the pressurizer safety valves as required to establish a bleed path.
- ~~Automatic safety injection via MHSI with reduced discharge head during low loop level ensures availability of the LHSI pumps for the RHR function. A note in Technical Specification 3.5.8 allows this automatic actuation feature to be removed from service temporarily for personnel protection during selected RCS maintenance activities.~~
  - ~~Routine RCS maintenance (e.g., refueling) will be performed during a full fuel offload.~~
  - ~~Infrequent RCS maintenance (e.g., mid-cycle steam generator repair) will be performed subject to the note in Technical Specification 3.5.8. During these infrequent RCS maintenance activities, automatic MHSI actuation may be disabled (as needed) to ensure personnel protection when fuel is in the reactor vessel. When this provision in Technical Specification 3.5.8 is used,~~

~~compensatory actions will be taken to provide reasonable assurance that the MHSI function can be promptly restored to manage the plant risk. The risk associated with disabling and restoring MHSI during these evolutions is discussed further in Chapter 19. Additionally, a COL applicant that references the U.S. EPR design certification will assess the risk (impact on the PRA and risk significant human actions) associated with RCS maintenance performed with fuel in the vessel.~~

- The RHR connection will be automatically isolated in the event of a break outside of the containment, based on the safeguard building sump level and pressure sensors. This non-safety function is performed by PAS.
- Spring-loaded safety relief valve, located at the RHR hot leg suction line, protects the SIS/RHRS against over-pressurization when in RHR mode.
- During reduced RCS inventory operations (e.g., mid-loop), a maximum RHR flow rate per train will be established for safe LHSI pump operation when operating in RHR mode. The decay heat at 36 hours (a representative time to reach Technical Specification Mode 5 conditions) can be matched with two trains of RHR running at a flow rate of approximately 1000 gpm per train. Thus a flow rate between 1000 and 1700 gpm per pump (1700 gpm per pump corresponds to the physical limit for RCS loop level for safe pump operation shown in Figure 5.4-19) is sufficient to remove decay heat, and prevent RHR pump cavitation, when the RHR system is operated in accordance with U.S. EPR FSAR Technical Specifications for Mode 5 relating to reduced RCS inventory conditions.~~During mid-loop operations, a maximum RHR flow rate will be established which minimizes the probability of suction pipe vortexing while providing adequate decay heat removal.~~
- ~~Redundant hot leg level sensors that initiate RCS make-up (safety and non-safety related) when the RCS hot leg has reached low level.~~
- RCS maintenance requiring disabling of MHSI will be performed during a full core offload. The U.S. EPR design does not include the use of nozzle dams with fuel in the vessel.
- ~~When nozzle dams are installed the following recommendations will be implemented:~~
  - ~~Removal of the pressurizer manway while the nozzle dams are installed and the reactor vessel head is in place. This action limits the pressurization of the RCS and inboard side of the nozzle dams which could follow an extended loss of decay heat removal.~~
  - ~~A hot leg manway will be the first manway to be opened.~~
  - ~~A hot leg nozzle dam will be the last dam to be installed.~~
  - ~~A hot leg manway and its associated hot leg pipe will be kept open to provide an adequate vent path whenever any cold leg openings are made.~~

– ~~The expeditious actions in GL 88 17 to be implemented any time that nozzle dams are installed.~~

- During mid-loop operation, the RCS loop level is normally controlled by the CVCS low pressure reducing valve to ensure there is sufficient RCS water inventory for operation of the LHSI pumps in RHR mode. The level control, limitation, and protection features are described below:

Loop Level Control Function	The RCS loop level control during mid-loop operation is regulated by the CVCS high pressure charging pumps and CVCS low pressure reducing station. See Section 7.7.2.2.3 for a description of loop level control. ~~After t~~The loop level control mode ~~has been~~is manually ~~validated~~ initiated~~, certain automatic protection functions are actuated~~. The nominal control band is shown on Figure 5.4-19.
Max1 RCS Loop Level Limitation Function	This setpoint initiates an open command for the CVCS low pressure letdown control valve in order to prevent inadvertent filling of the steam generator bowls. ~~(without nozzle dams)~~ The Max1 setpoint is shown on Figure 5.4-19.
Min1 RCS Loop Level Limitation Function	This setpoint initiates full closure of the CVCS low pressure letdown control valve ~~and the RHR~~ and the CVCS isolation valve~~s~~ in order to protect the LHSI pumps that are operating in RHR mode. This function covers the entire temperature range of the RHR system operation. The Min1 setpoint is shown on Figure 5.4-19.
Min1p RCS Loop Level Safety Function	This setpoint initiates the SIS in case of low RCS level in the primary loops in the event of a sudden drop in RCS level during mid-loop operation to provide RCS makeup. It initiates a stage 1 containment isolation signal. This setpoint also initiates automatic stop of the LHSI pumps in RHR mode to prevent cavitation of operating LHSI pumps due to loss of suction. The Min1p setpoint is shown on Figure 5.4-19. ~~in order to protect the RHR pumps and maintain adequate core cooling.~~

- The reactor pressure vessel (RPV) water level is continually monitored during outage with a level sensor. The level sensor taps are located on the top and bottom of each hot leg approximately ten feet from the steam generator center line and approximately six feet closer to the steam generator than the LHSI RHR suction line.~~discharge nozzle.~~

- Temperature sensors, located at the RCS hot legs, allow temperature measurement of each hot leg when in a reduced inventory condition.

5.4.7.3.1 Performance Evaluation with All Components Operable

In a normal cooldown, the reactor decreases in power by insertion of the rod cluster control assemblies (RCCA). The cooldown of the RCS must not exceed 90°F/hr, while the cooldown of the pressurizer must not exceed 212°F/hr. From power operation to the hot standby mode, all four reactor coolant pumps (RCPs) are in operation for mixing of the coolant in the RCS, the pressurizer level is automatically controlled by controlling the CVCS letdown flow, the primary pressure is automatically adjusted by the main spray flow and the pressurizer heaters, and the residual heat is being removed by the steam generators. The steam generators levels are controlled by the main feedwater system (refer to Section 10.4.7).

Automatic cooldown of the RCS by the secondary systems from the hot standby mode to SIS/RHRS connection point is accomplished in parallel with the automatic RCS depressurization via the pressurizer. In this phase, the reactor coolant make-up is performed using the CVCS, pressurizer level is automatically controlled by the CVCS letdown line, while the steam generators levels are controlled by the startup and shutdown system.

In the analysis performed, two RCPs are tripped when the RCS temperature decreases to 250°F, another RCP is tripped when the RCS temperature decreases to 158°F, and the last RCP is tripped when the RCS temperature decreases to 122°F.

Two trains of the SIS/RHRS are normally placed in service for residual heat removal when the RCS pressure and temperature decreases below approximately ~~390~~ 370 psia and 250°F. The remaining two trains are placed in service after the RCS temperature has been further reduced to approximately 212°F.

A representative ~~P~~performance curve showing the calculated cooldown rates for four trains operation is shown in Figure 5.4-3—Representative of RCS Cooldown for Four Train SIS/RHRS Shutdown Cooling Operation. From Figure 5.4-3, the time required to cool the plant down to approximately 250°F is around 7.3 hours after reactor trip~~,~~. RCS cooling is then shifted to the RHR system, which cools~~while the time required to cool~~ the RCS temperature down to approximately 131°F (using all four LHSI heat exchangers in the sequence explained) in~~s~~ about another 9.7 hours. The total time to cool the plant down to approximately 131°F (for refueling) is approximately 17 hours after reactor trip. This total time attained is shorter than the required time of 40 hours, specified in Section 5.4.7.1.

5.4.7.3.2 Performance Evaluation for Branch Technical Position 5-4 Cooldown

System(s) that can be used for heat removal, including depressurization, flow circulation, and reactivity control to take the reactor from normal operating conditions to cold shutdown should satisfy the general functional requirements of Branch Technical Position (BTP) 5-4.

The U.S. EPR meets the requirements of BTP 5-4 with the following exception:

> Cases where an EFW pump is unavailable due to single failure or maintenance, action outside of the control room may be required to re-align the manual supply header valves to provide access to the inventory from all four storage pools. Sufficient water inventory is available for six to eight hours of EFWS operation before this action is necessary.

5.4.7.3.2.1 Cooldown Analyses

Cooldown analyses were performed using the RELAP5 computer code for several single failure scenarios with and without offsite power available. The bounding case, with respect to EFWS inventory usage, is a natural circulation cooldown following the loss of offsite power with a failed closed EFW SG level control valve (LCV). This failure results in an unfed SG and associated stagnant RCS loop. The stagnant RCS loop conditions require a slow and controlled cooldown and depressurization to SIS/RHRS entry conditions.

5.4.7.3.2.2 Cooldown Scenario

For the BTP 5-4 cooldown, with only onsite power available, the cooldown to cold shutdown condition is achieved with use of only safety-related equipment. The LOOP results in immediate RCP coastdown and termination of the main feedwater supply. Cooldown prior to SIS/RHRS connection is achieved by natural circulation using the main steam relief trains (MSRT), while the steam generators levels are controlled by the emergency feedwater (EFW) system; the EFW system begins operation once the EFW pumps are automatically loaded onto the emergency diesel generators. During cooldown, boron concentration is adjusted as necessary through the use of the extra borating system (EBS). RCS make-up is accomplished by using the medium head safety injection (MHSI) pumps taking suction from the IRWST and by EBS injection flow from the EBS tanks.

The following provides the detailed scenario:

- Reactor and turbine are tripped at about 33 minutes. ~~2000 seconds~~.
- MSRT trains are available all the time.
- EFW Trains 1, 2 and 3 are available all the time. EFW LCV on Train 4 fails closed at the beginning of the transient and stays closed for the duration of the transient. SG 4 has no EFW flow during the transient; however, it can be steamed via its MSRT.
- Reactor Coolant Pumps (RCPs) are tripped at the beginning of the transient and stay tripped for the rest of the transient.

- Pressurizer (PZR) heaters, normal spray, and auxiliary spray systems are not available during the transient.
- PZR makeup/letdown system is not available during the transient. PZR makeup can be performed using the medium head safety injection (MHSI) system and two trains of extra borating system (EBS).
- Turbine bypass system (TBS) is not available and stays closed during the transient.
- Main feedwater (MFW) system trains are not available during the transient.
- MSIVs stay closed during the transient.
- Cooldown is initiated at about 4 hours ~~17,000 seconds or 15,000 seconds (4.17 hours)~~ after reactor trip (RT). This allows the RCS to remain at the hot standby condition for more than four hours. The cooldown rate is 25°F/hr.
- During cooldown, the RCS pressure is manually controlled by the operator through the safety-related PZR safety relief valve (PSRV). A minimum core exit subcooled margin of 50°F is maintained.
- SIS/RHRS entry conditions are achieved when the hot leg fluid temperatures at each non-stagnant loop and the saturation temperature at the stagnant loop is equal to or less than 350°F, which is the highest allowed for connection of the SIS/RHRS to perform its residual heat removal function.
- One train of the SIS/RHRS is aligned for its residual heat removal function at approximately 350°F, with another two SIS/RHRS trains placed in service at approximately 212°F. The RCS is then cooled to cold shutdown conditions.

5.4.7.3.2.3 Cooldown Analysis Results

The analysis starts with a RT from hot full power condition, at which time a LOOP is assumed to occur and de-energize the RCPs. The plant is assumed to maintain hot standby conditions ~~for the first 15,000 seconds~~ about the first 4 hours, at which time the operator initiates a RCS cooldown.

RCS pressure increases after RT and turbine trip as SG secondary pressure increases to the MSRT setpoint. Shortly afterwards, the pressure stabilizes at hot standby conditions. The pressure then drops after about 30 minutes when EFW is actuated on low SG level followed by an increase and then stabilizes. At approximately 1.5 hours ~~5,000 seconds~~, SGs 1, 2, and 3 are returned to normal level and EFW level control becomes active, except for SG 4 which has boiled dry.

At ~~15,000 seconds~~ approximately 4 hours, a 25°F/hr cooldown is initiated using the MSRTs to reduce secondary side pressure and the PSRV is manually cycled to decrease RCS pressure while maintaining the core exit subcooling margin between 50 and 75°F.

When the RCS pressure is decreased to the saturation pressure of the hot water in the stagnant RCS loop, some surging of pressurizer level occurs; however, the level stays within scale and there is no discharge of liquid water from the PSRV.

SIS/RHRS operation entry conditions are met when primary temperature is below 350°F and primary pressure is below 400 psia ~~at~~ within about ~~62,400 seconds~~ 17 hours after RT, or within about 13 hours ~~47,400 seconds~~ after the initiation of the cooldown. Curves showing pertinent information for the cooldown to RHRS entry conditions information are provided in Figures 5.4-13 through 5.4-18.

Figure 5.4-4—RCS Cooldown for BTP 5-4 (SIS/RHRS Shutdown Cooling Operation) conservatively shows the cooldown time for an SIS/RHRS connection at an earlier entry time of approximately ~~60,000 seconds~~ 17 hours. The cooldown time, with three SIS/RHRS trains in RHR mode, from approximately 350°F to 200°F (cold shutdown condition) is within approximately ~~11.6~~ 12 hours.

5.4.7.3.2.4 Conclusion

The cooldown analysis demonstrates that the plant can be cooled to cold shutdown conditions in a reasonable time following the BTP 5-4 functional requirements. The bounding EFW water inventory required to reach SIS/RHRS entry conditions is approximately 365,000 gallons.

Cold shutdown conditions are reached ~~at~~ within approximately 29 hours after reactor trip.

5.4.7.3.3 Performance Evaluation for Design Basis Accident Cooldown

For the design basis accident cooldown, with only onsite power available, two SIS/RHRS trains are available to remove residual heat, assuming one train is unavailable due to system maintenance and a second train is lost due to single failure.

Based on the cooldown analysis results provided in Section 5.4.7.3.2, the SIS/RHRS operation entry conditions are met when primary temperature is below 350°F and primary pressure is below 400 psia ~~at~~within about ~~62,400 seconds~~ 17 hours after RT, or within about 13 hours ~~47,400 seconds~~ after initiation of the cooldown.

Figure 5.4-5—RCS Cooldown for Design Basis Accident Condition (SIS/RHRS Shutdown Cooling Operation) conservatively shows the cooldown time for an SIS/RHRS connection at an earlier entry time of between approximately 3 to 5 hours. The cooldown time, with a higher component cooling water temperature limit, from approximately 350°F to 200°F (cold shutdown condition) is within approximately 11 hours.

thermal insulation for the pressurizer is removable in areas subject to ISI. The ISI of the RCPB is addressed in Section 5.2.4. The radiological considerations for pressurizer maintenance are addressed in Section 12.4.

The shape and slope of the welded parts, including safe-end-to-nozzle welds, allow performance of both radiographic and ultrasonic examination. Nozzle-to-head welds are sufficiently remote from other welds to allow the required ultrasonic examination.

5.4.11 Pressurizer Relief Tank

The pressurizer relief tank (PRT) collects and condenses the steam discharged from the pressurizer through the PSRVs and the primary depressurization system (PDS) valves. The pressurizer relief system piping routes the discharge from the valves to the PRT.

Section 5.4.13 describes the PSRVs. Section 19.2 addresses the PDS valves.

5.4.11.1 Design Bases

The pressurizer relief system and the PRT are non-safety-related. The PRT and the pressurizer relief system piping are designed to Seismic Category II requirements to prevent adverse impacts on safety-related systems in the event of an earthquake. The requirements for Seismic Category II structures, systems and components (SSC) are presented in Section 3.2 (GDC 2).

The PRT and associated rupture disks are sized and located to prevent unacceptable damage to safety-related systems or components resulting from missile generation or adverse environmental conditions (GDC 4).

The PRT isolates radioactive materials from the rest of the containment environment to limit contamination and to maintain personnel dose as low as reasonably achievable.

The PRT is designed to process 110 percent of the steam present in the pressurizer at full-power conditions without challenging the rupture disks or the design temperature of the PRT. This load on the PRT encompasses all step load decreases and a PZR discharge resulting from the turbine trip transient described in Section 15.2, which is the limiting pressurizer discharge for an anticipated operational occurrence. Section 5.2.2 describes the overpressure protection function of the PSRVs.

The PRT design incorporates two rupture disks that protect the PRT from overpressurization. The flow area of one rupture disk is larger than the PSRV discharge pipe and greater than what is required to handle the full flow rate of three PSRVs. The rupture disks prevent the PRT pressure from exceeding the design limits.

Table 5.4-7—Pressurizer Heater Capacities

Group	Power
Proportional heaters	720 kW
Group 1 heaters	576 kW
Group 2 heaters	1296 kW
Minimum total pressurizer heater capacity	2500 kW
Minimum heater capacity to compensate for pressurizer losses during an LOOP (Provided by any two emergency Heater Groups)	288 kW
Total minimum emergency heater available (Twice the minimum required emergency heater capacity)	576 kW

Figure 5.4-3—Representative of RCS Cooldown for Four Train SIS/RHRS Shutdown Cooling Operation

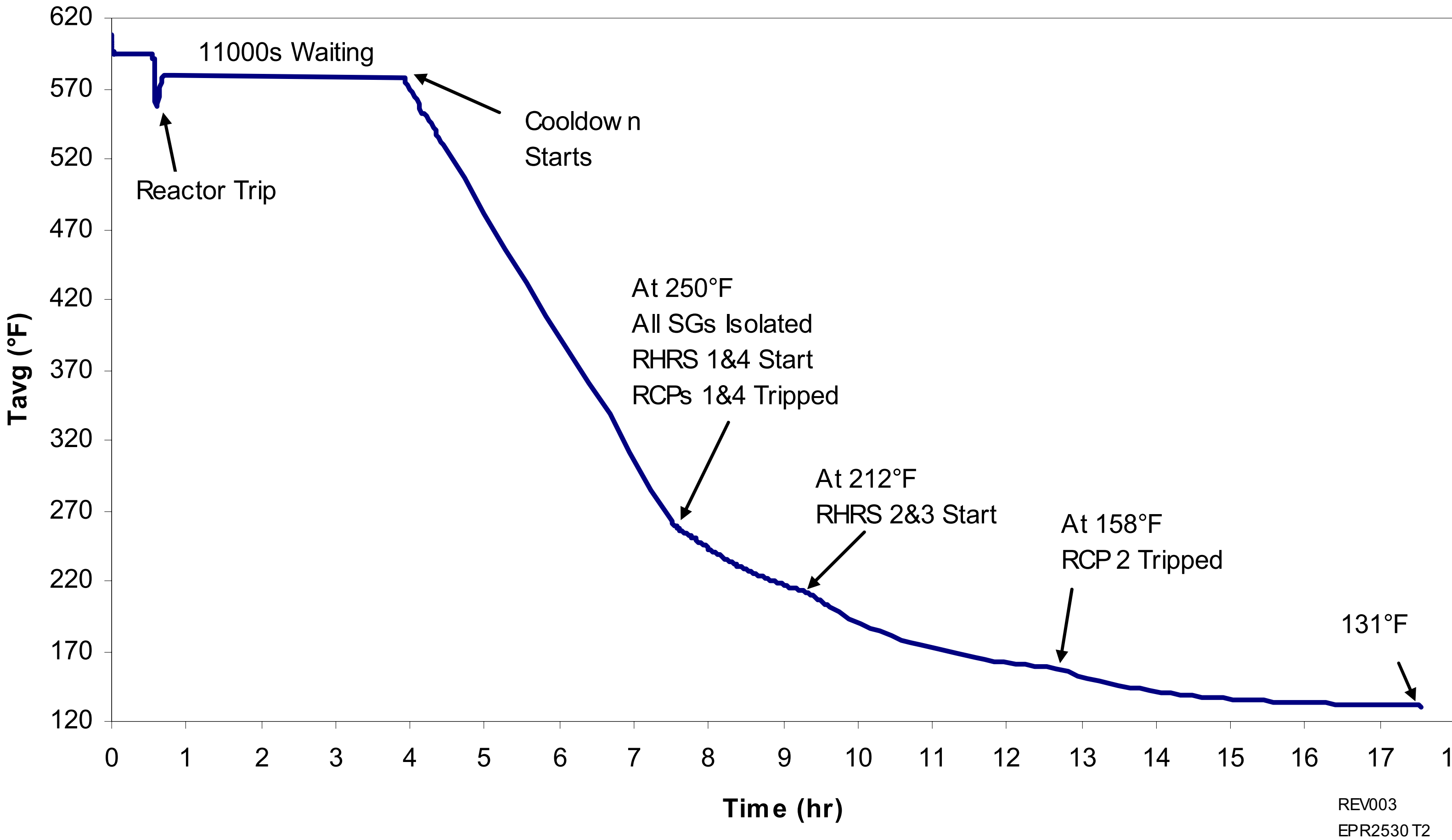
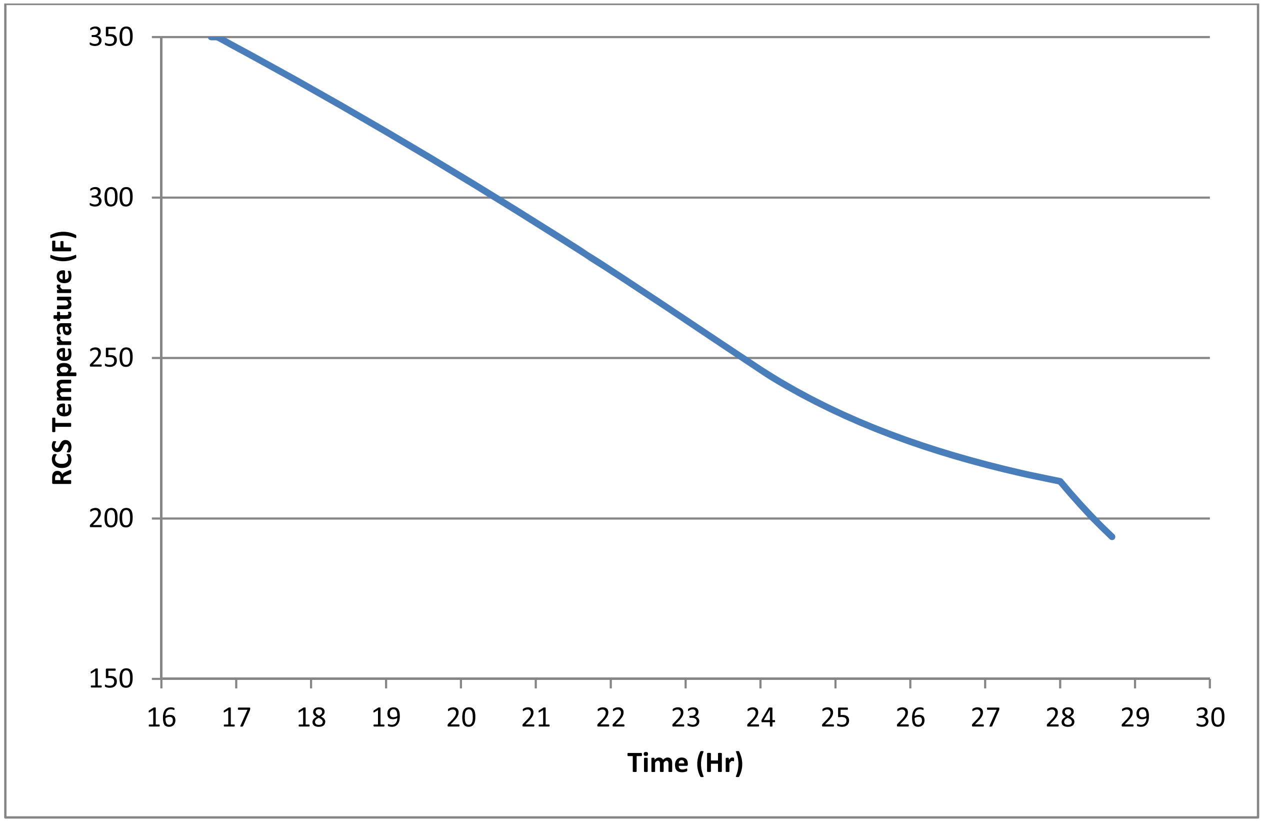
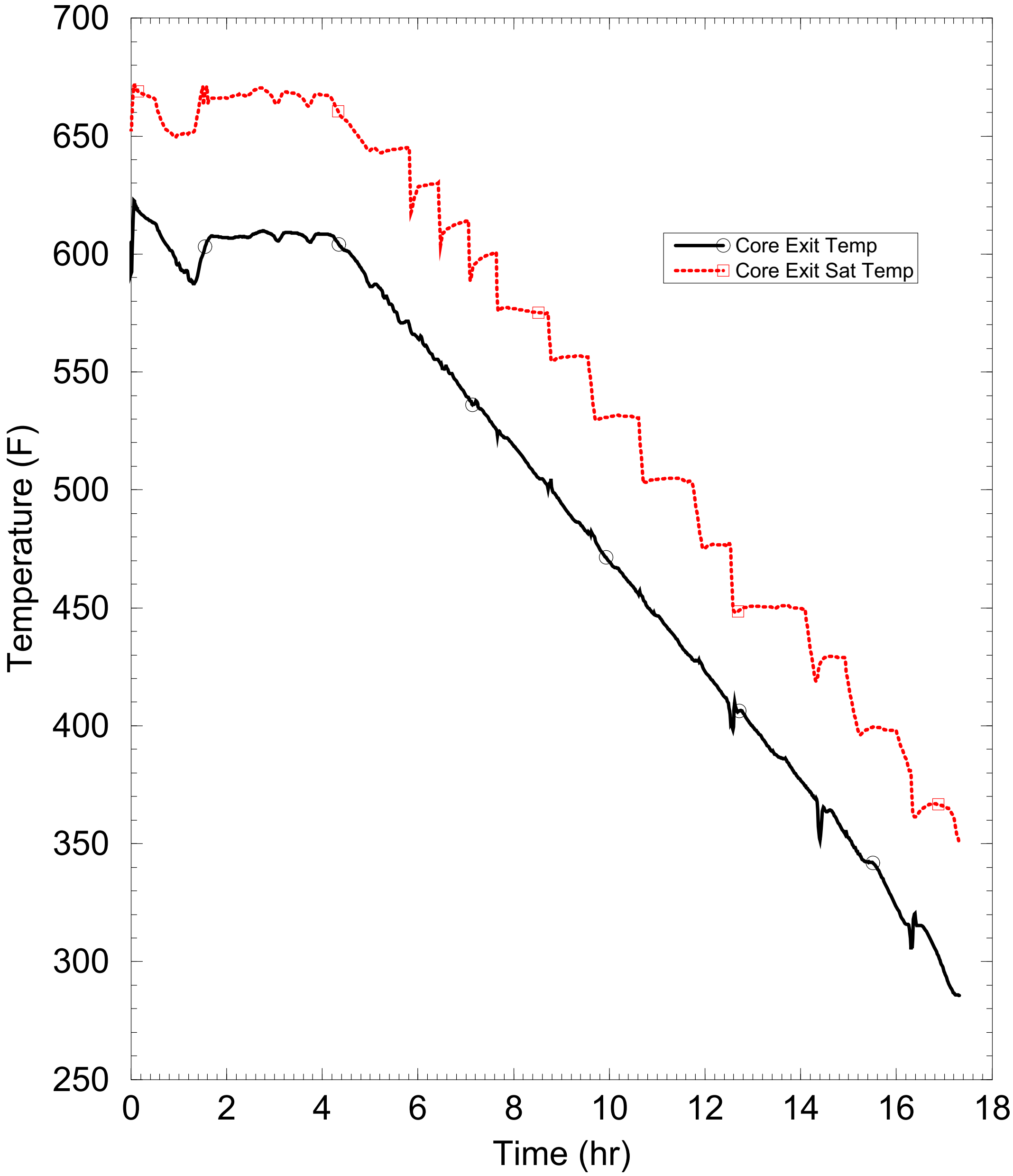


Figure 5.4-4—RCS Cooldown for BTP 5-4 (SIS/RHRS Shutdown Cooling Operation)



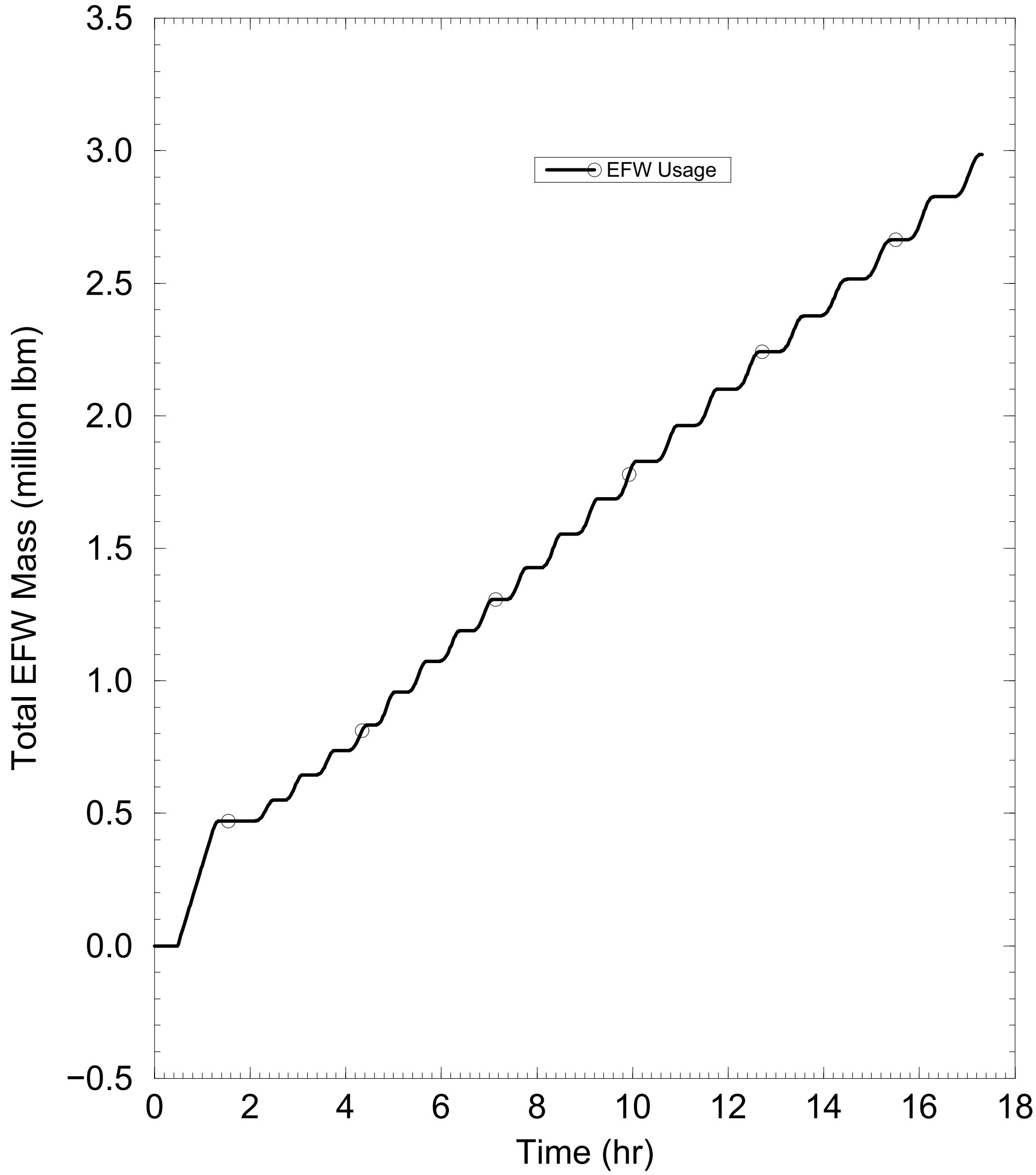
REV 007
EPR2540 T2

Figure 5.4-13—Core Exit Temperature



REV 007
EPR2550 T2

Figure 5.4-14—EFW Usage



REV 007
EPR2555 T2

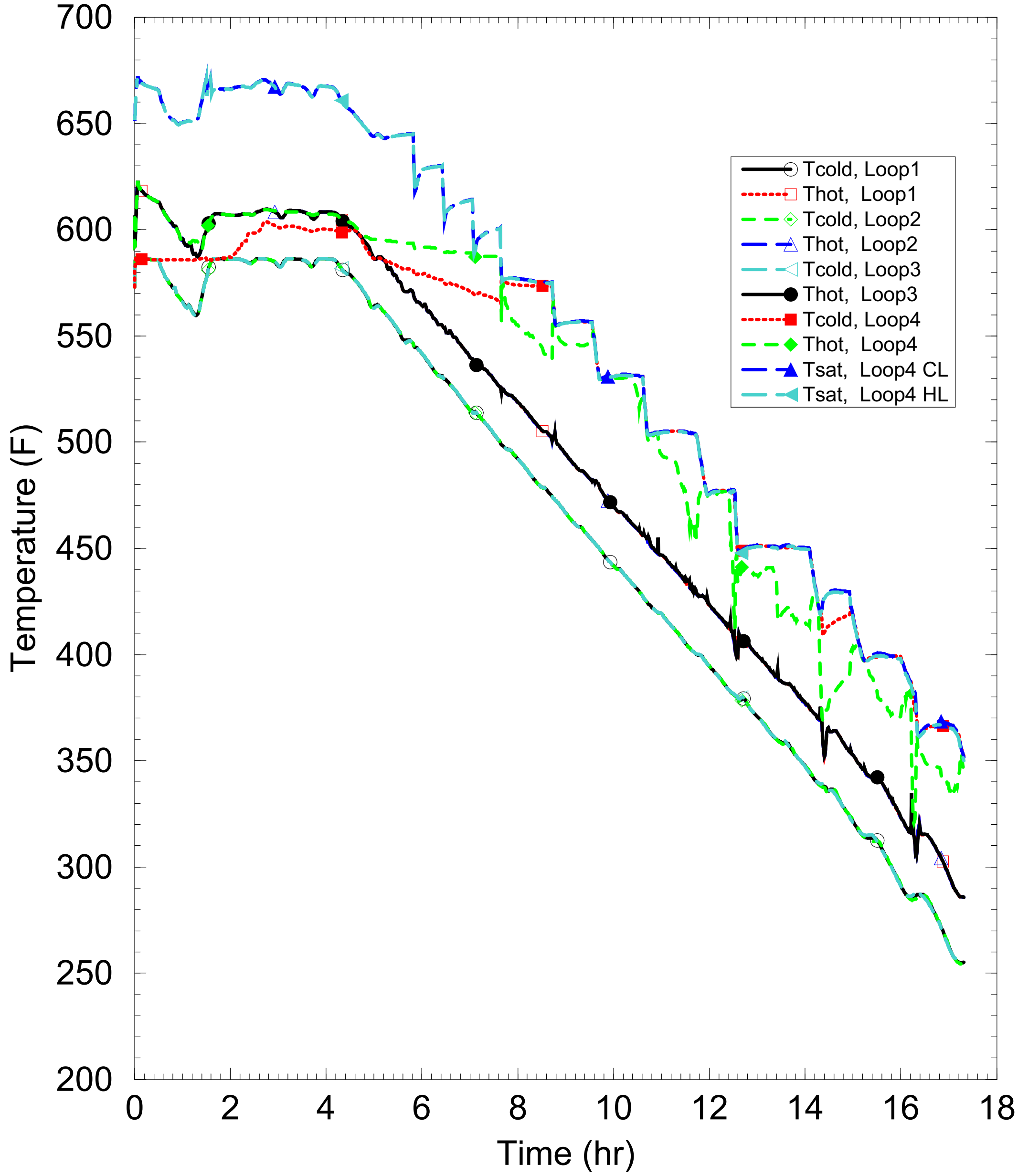
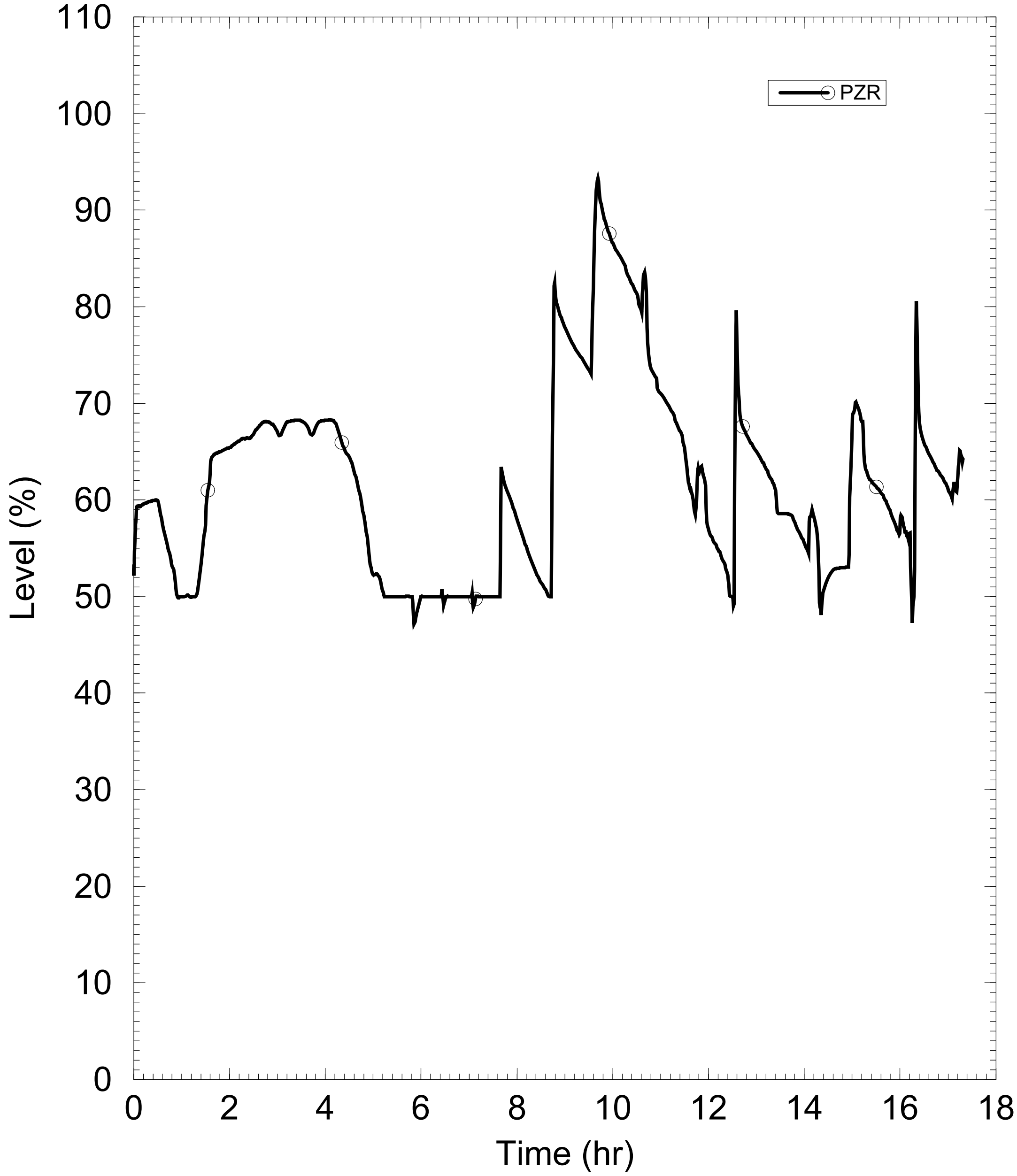


Figure 5.4-15—Primary Temperatures

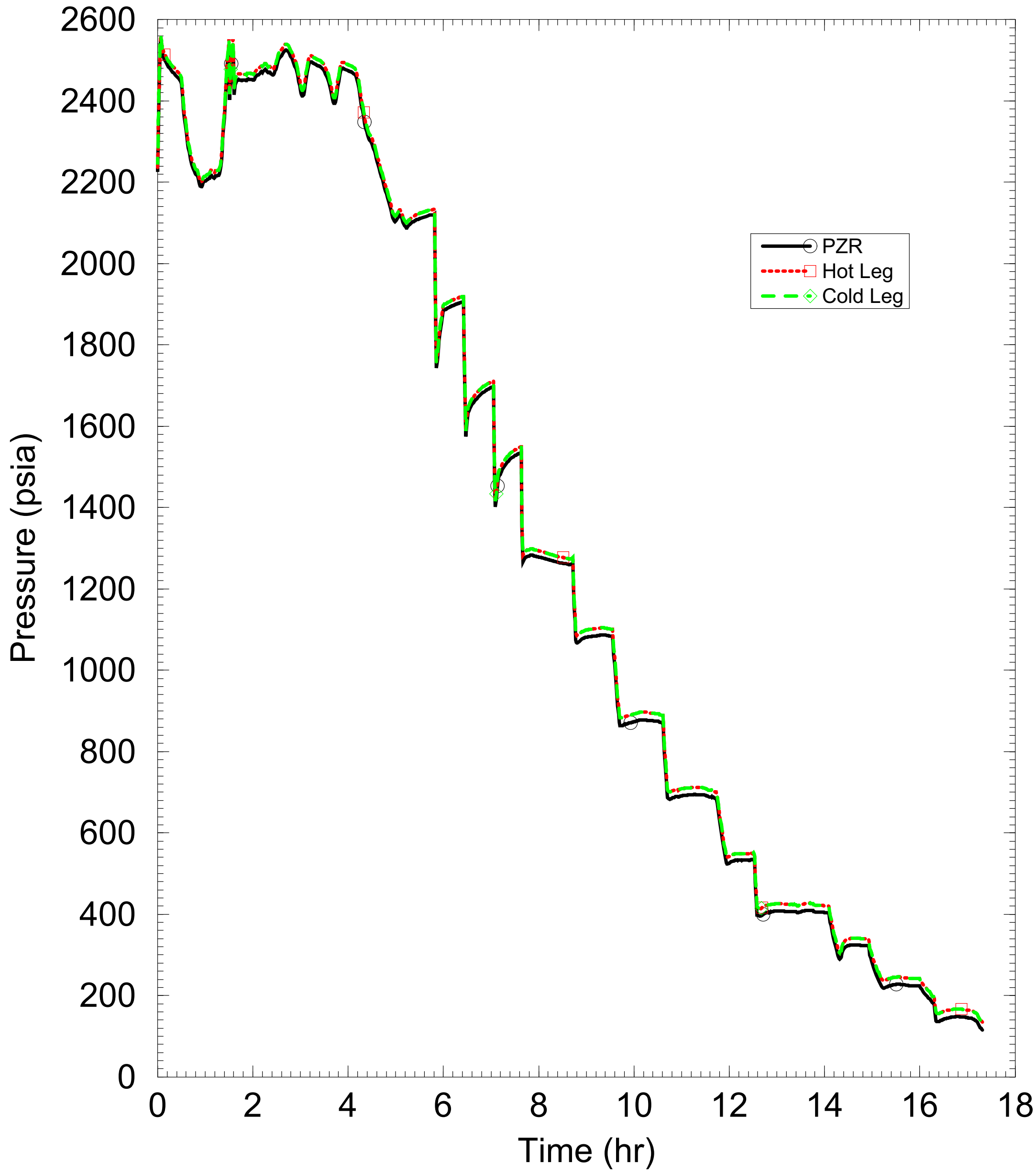
REV 007
EPR2560 T2

Figure 5.4-16—Pressurizer Level



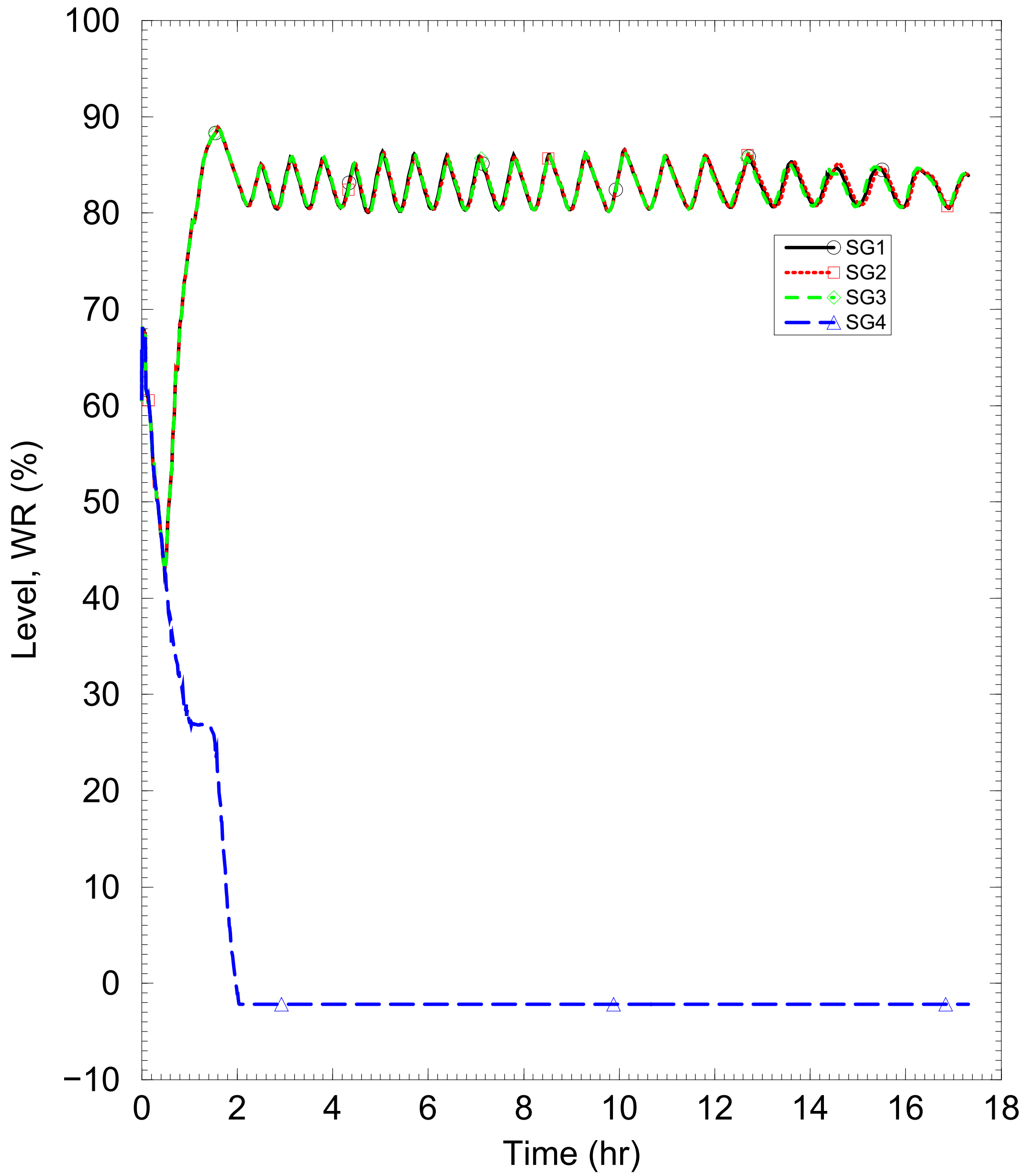
REV 007
EPR2565 T2

Figure 5.4-17—RCS Pressures



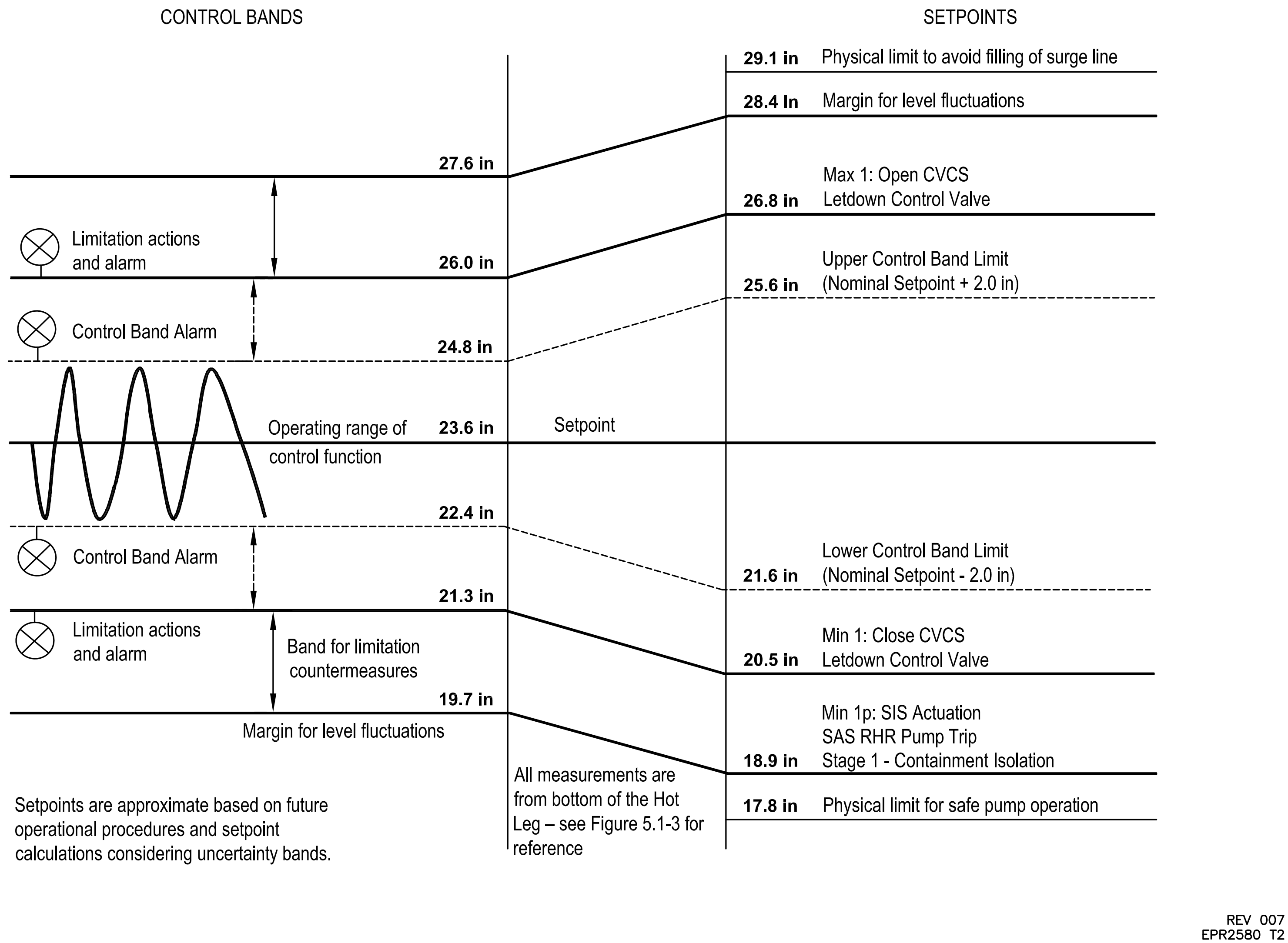
REV 007
EPR2570 T2

Figure 5.4-18—SG Levels



REV 007
EPR2575 T2

Figure 5.4-19—Level Control During Mid-loop Operation



Conforming Changes

3.7 Deleted.

3.8 Deleted.

3.9 Deleted.

3.10 Deleted.

3.11 Deleted.

3.12 Deleted.

3.13 Deleted.

3.14 Deleted.

3.15 ASME Code Class1 and 2 piping systems are designed in accordance with ASME Code Section III requirements.

3.16 As-built ASME Code Class 1 and 2 components listed in Table 2.2.3-1 are reconciled with the design requirements.

3.17 Pressure-boundary welds in ASME Code Class 1 and 2 components listed in Table 2.2.3-1 meet ASME Code Section III non-destructive examination requirements.

3.18 ASME Code Class 1 and 2 components listed in Table 2.2.3-1 retain their pressure-boundary integrity at their design pressure.

3.19 ASME Code Class 1 and 2 components listed in Table 2.2.3-1 are fabricated, installed, and inspected in accordance with ASME Code Section III requirements.

3.20 The LHSI system operating in RHR mode will safely operate during mid-loop conditions.

4.0 I&C Design Features, Displays, and Controls

4.1 Displays listed in Table 2.2.3-2 are indicated on the PICS operator workstations in the main control room (MCR) and the remote shutdown station (RSS).

4.2 Controls on the PICS operator workstations in the MCR and the RSS perform the function listed in Table 2.2.3-2.

4.3 Equipment listed as being controlled by a priority and actuator control system (PACS) module in Table 2.2.3-2 responds to the state requested and provides drive monitoring signals back to the PACS module. The PACS module will protect the equipment by terminating the output command upon the equipment reaching the requested state.

4.4 Interlocks for the SIS/RHRS initiate the following:

- Opening of the accumulator injection path.

Table 2.2.3-3—Safety Injection System and Residual Heat Removal System ITAAC
Sheet 4 of 9

	Commitment Wording	Inspections, Tests, Analyses	Acceptance Criteria
3.19	ASME Code Class 1 and 2 components listed in Table 2.2.3-1 are fabricated, installed, and inspected in accordance with ASME Code Section III requirements.	An inspection of the as-built construction activities and documentation for ASME Code Class 1 and 2 components listed in Table 2.2.3-1 will be conducted.	ASME Code Data Report(s) exist that conclude that ASME Code Class 1 and 2 components listed in Table 2.2.3-1 are fabricated, installed, and inspected in accordance with ASME Code Section III requirements.
3.20	The LHSI system operating in RHR mode will safely operate during mid-loop conditions.	Tests will be performed to demonstrate that the LHSI system operating in RHR mode can safely operate at the minimum mid-loop level and maximum design flow.	A report concludes that the LHSI system operating in RHR mode is capable of safe operation during mid-loop conditions.
4.1	Displays listed in Table 2.2.3-2 are indicated on the PICS operator workstations in the MCR and the RSS.	a. Tests will be performed to verify that the displays listed in Table 2.2.3-2 are indicated on the PICS operator workstations in the MCR. b. Tests will be performed to verify that the displays listed in Table 2.2.3-2 are indicated on the PICS operator workstations in the RSS.	a. Displays listed in Table 2.2.3-2 are indicated on the PICS operator workstations in the MCR. b. Displays listed in Table 2.2.3-2 are indicated on the PICS operator workstations in the RSS.
4.2	Controls on the PICS operator workstations in the MCR and the RSS perform the function listed in Table 2.2.3-2.	a. Tests will be performed using controls on the PICS operator workstations in the MCR. b. Tests will be performed using controls on the PICS operator workstations in the RSS.	a. Controls on the PICS operator workstations in the MCR perform the function listed in Table 2.2.3-2. b. Controls on the PICS operator workstations in the RSS perform the function listed in Table 2.2.3-2.

Table 2.4.4-2—Safety Automation System Automatic Functions and Input Variables
Sheet 9 of 9

System	Function Name	Input Variable
Safety Injection and Residual Heat Removal System (SIS/RHRS)	Automatic RHRS Flow Rate Control	RHRS Flow Rate Signal
		RHRS Temperature
		RHRS Pump Discharge Pressure
		LHSI Heat Exchanger Bypass Valve Position
	RHR Isolation Valves Interlock	LHSI Suction Isolation Valve Position
		RHR 1st RCPB Isolation Valve Position
		RHR 2nd RCPB Isolation Valve Position
	Automatic Trip of LHSI Pump (in RHR Mode) on Low ΔPsat Interlock	Hot Leg Temperature
		Hot Leg Pressure
	Automatic Trip of LHSI Pump (in RHR Mode) on Low RCS Loop Level Interlock	Hot Leg Loop Level
	Detection of RHRS Train Connected	RHR 1st RCPB Isolation Valve Position
		RHR 2nd RCPB Isolation Valve Position
		RHR Outside Containment Isolation Valve Position
		LHSI Suction Isolation Valve Position
		Hot Leg Injection Isolation Valve Position

Table 2.4.4-3—Safety Automation System Interlocks

CCWS Switchover Valves Interlock
CCWS RCP Thermal Barrier Containment Isolation Valve Interlock
CCWS RCP Thermal Barrier Containment Isolation Valves Opening Interlock
IRWST Boundary Isolation for Preserving IRWST Water Inventory Interlock
SCWS Train 1 to Train 2 Switchover on Train 1 Loss of Pump / Loss of Chiller / SCWS Chiller Evaporator Water Flow Control / LOOP Re-Start Failure Interlock
SCWS Train 2 to Train 1 Switchover on Train 2 Loss of Pump / Loss of Chiller / SCWS Chiller Evaporator Water Flow Control / LOOP Re-Start Failure Interlock
SCWS Train 3 to Train 4 Switchover on Train 3 / SCWS Chiller Evaporator Water Flow Control / LOOP Re-Start Failure Interlock
SCWS Train 4 to Train 3 Switchover on Train 4 Loss of Pump / Loss of Chiller / SCWS Chiller Evaporator Water Flow Control / LOOP Re-Start Failure Interlock
RHR Isolation Valves Interlock
Automatic Trip of LHSI Pump (in RHR Mode) on Low ΔPsat Interlock
Automatic Trip of LHSI Pump (in RHR Mode) on Low RCS Loop Level Interlock
Detection of RHRS Train Connected

Table 2.4.25-2—Signal Conditioning and Distribution System Input Signals
Sheet 6 of 7

Item #	Signal	Source	# Divisions
105	SBVSE Recirculation / Exhaust Fan Stopped Signal	Electrical Division of Safeguard Building Ventilation System	4
106	SBVSE Outside Air Damper Closed Position Signal	Electrical Division of Safeguard Building Ventilation System	4
107	SBVSE Recirculation Damper Closed Position Signal	Electrical Division of Safeguard Building Ventilation System	4
108	SBVSE Exhaust Fan Exhaust Damper Closed Position	Electrical Division of Safeguard Building Ventilation System	4
109	Filter Bank Differential Pressure	Electrical Division of Safeguard Building Ventilation System	4
110	Supply Air Downstream of Humidifier Temperature	Electrical Division of Safeguard Building Ventilation System	4
111	Battery Room Supply Air Downstream of Heaters Flow	Electrical Division of Safeguard Building Ventilation System	4
112	Battery Room Temperature	Electrical Division of Safeguard Building Ventilation System	4
113	Battery Room Supply Air Temperature	Electrical Division of Safeguard Building Ventilation System	4
114	EFWS Pump Room Temperature	Electrical Division of Safeguard Building Ventilation System	4
115	CCWS Pump Room Temperature	Electrical Division of Safeguard Building Ventilation System	4
116	SCWS Chiller Evaporator Outlet Temperature	Safety Chilled Water System	4
117	Deleted	Deleted	Deleted
118	SCWS Condenser Refrigerant Pressure	Safety Chilled Water System	4
119	SCWS Chiller Evaporator Flow Signal	Safety Chilled Water System	4
120	SCWS Cross-Tie Valves Position Signal	Safety Chilled Water System	4
121	SCWS Circulating Pump 1 Running Signal	Safety Chilled Water System	4
122	SCWS Circulating Pump 2 Running Signal	Safety Chilled Water System	4
123	SCWS Evaporator ΔP Signal	Safety Chilled Water System	4
124	SCWS Chiller Evaporator Flow Signal	Safety Chilled Water System	4
125	LHSI Heat Exchanger Bypass Control Valve Position	Safety Injection and Residual Heat Removal System	4

Table 2.4.25-2—Signal Conditioning and Distribution System Input Signals Sheet 7 of 7

Item #	Signal	Source	# Divisions
126	RHRS Flow Rate Signal	Safety Injection and Residual Heat Removal System	4
127	RHRS Temperature	Safety Injection and Residual Heat Removal System	4
128	RHRS Pump Discharge Pressure	Safety Injection and Residual Heat Removal System	4
129	Hot Leg Loop Level	Reactor Coolant System ~~Safety Injection and Residual Heat Removal System~~	4
130	Containment Isolation Signal	Fuel Building Ventilation System	4
131	LHSI Suction Isolation Valve Position	Safety Injection and Residual Heat Removal System	4
132	RHR 1st RCPB Isolation Valve Position	Safety Injection and Residual Heat Removal System	4
133	RHR 2nd RCPB Isolation Valve Position	Safety Injection and Residual Heat Removal System	4
134	RHR Outside Containment Isolation Valve Position	Safety Injection and Residual Heat Removal System	4
135	LHSI Hot Leg Injection Isolation Valve Position	Safety Injection and Residual Heat Removal System	4
136	CCWS Common 2a Supply Valve Position	Component Cooling Water System	2
137	CCWS Common 2a Return Valve Position	Component Cooling Water System	2
138	CCWS Common 2b Supply Valve Position	Component Cooling Water System	2
139	CCWS Common 2b Return Valve Position	Component Cooling Water System	2
140	Steam Generator Transfer Valve Position	Steam Generator Blowdown System	4

Table 2.4.25-3—Signal Conditioning and Distribution System Output Signals
Sheet 7 of 7

Item #	Signal	Recipient	# Divisions
121	SCWS Circulating Pump 1 Running Signal	Safety Automation System	4
122	SCWS Circulating Pump 2 Running Signal	Safety Automation System	4
123	SCWS Evaporator ΔP Signal	Safety Automation System	4
124	SCWS Chiller Evaporator Flow Signal	Safety Automation System	4
125	LHSI Heat Exchanger Bypass Control Valve Position	Safety Automation System	4
126	RHRS Flow Rate Signal	Safety Automation System	4
127	RHRS Temperature	Safety Automation System	4
128	RHRS Pump Discharge Pressure	Safety Automation System	4
129	Hot Leg Loop Level	Safety Automation System	4
130	Containment Isolation Signal	Safety Automation System	4
131	LHSI Suction Isolation Valve Position	Protection System Safety Automation System	4
132	RHR 1st RCPB Isolation Valve Position	Protection System Safety Automation System	4
133	RHR 2nd RCPB Isolation Valve Position	Protection System Safety Automation System	4
134	RHR Outside Containment Isolation Valve Position	Protection System	4
135	LHSI Hot Leg Injection Isolation Valve Position	Protection System	4
136	Steam Generator Transfer Valve Position	Protection System	4

Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 17 of 39

Item No.	Description	Section
3E-1	A COL applicant that references the U.S. EPR design certification will address critical sections relevant to site-specific Seismic Category I structures.	3E
5.2-1	Deleted.	
5.2-2	A COL applicant that references the U.S. EPR design certification will identify additional ASME code cases to be used.	5.2.1.2
5.2-3	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the site-specific ASME Section XI preservice and inservice inspection program for the reactor coolant pressure boundary, consistent with the requirements of 10 CFR 50.55a (g). The program will identify the applicable edition and addenda of the ASME Code Section XI, and will identify additional relief requests and alternatives to Code requirements.	5.2.4
5.2-4	A COL applicant that references the U.S. EPR design certification will develop procedures in accordance with RG 1.45, Revision 1.	5.2.5.5
5.3-1	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the material surveillance program.	5.3.1.6
5.3-2	A COL applicant that references the U.S. EPR design certification will provide a plant-specific pressure and temperature limits report (PTLR), consistent with an approved methodology.	5.3.2.1
5.3-3	A COL applicant that references the U.S. EPR design certification will provide plant-specific RT_{PTS} values in accordance with 10 CFR 50.61 for vessel beltline materials.	5.3.2.3
5.3-4	A COL applicant that references the U.S. EPR design certification will provide plant-specific surveillance data to benchmark BAW-2241P-A and demonstrate applicability to the specific plant.	5.3.1.6.2
5.4-1	A COL applicant that references the U.S. EPR design certification will identify the edition and addenda of ASME Section XI applicable to the site specific Steam Generator inspection program.	5.4.2.5.2.2
~~5.4-2~~	~~A COL applicant that references the U.S. EPR design certification will assess the risk (impact on the PRA and risk significant human actions) associated with RCS maintenance performed with fuel in the vessel.~~	~~5.4.7.2.1~~
6.1-1	A COL applicant that references the U.S. EPR design certification will review the fabrication and welding procedures and other QA methods of ESF component vendors to verify conformance with RGs 1.44 and 1.31.	6.1.1.1

Table 7.1-5—SAS Automatic Safety Function
Sheet 21 of 22

System[1]	Function Name[2]	Function Safety Basis[3]	Interdivisional Communications[4]	Type of Data[5]	Signal Selection Type[6]	Function Initiation[7]
Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Automatic Trip of LHSI Pump (in RHR Mode) on Low ΔPsat Interlock (Figure 7.6-9)	This function is described in Sections 5.4.7, 6.3, and 7.6.1.2.2.	Interdivisional communications is required because a low ΔPsat discrete signal is generated in each division, and 2/4 voting logic is used to trip the LHSI pump. Valve position measurements from multiple divisions are used to determine if an RHR train is connected.	Discrete	Vote	PS Signal
Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Automatic Trip of LHSI Pump (in RHR Mode) on Low RCS Loop Level Interlock (Figure 7.6-10)	This function is described in Sections 5.4.7, 6.3, and 7.6.1.2.3.	Interdivisional communications is required because a low RCS loop level discrete signal is generated in each division, and 2/4 voting logic is used to trip the LHSI pump. Valve position measurements from multiple divisions are used to determine if an RHR train is connected.	Discrete	Vote	PS Signal

Table 7.1-5—SAS Automatic Safety Function
Sheet 22 of 22

System[1]	Function Name[2]	Function Safety Basis[3]	Interdivisional Communications[4]	Type of Data[5]	Signal Selection Type[6]	Function Initiation[7]
Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Detection of RHRS Train Connected (Figure 7.6-13)	This function is described in Section 7.6.1.2.1.	The RHR 1st and 2nd RCPB Isolation Valves are powered from different divisions. (e.g., Train 1 RHR 1st RCPB Isolation Valve is powered by Division 1 and Train 1 2nd RCPB Isolation Valve is powered by Division 2.) Therefore, the state (open or closed) of these valves must be communicated across divisions.	Discrete	Vote	Continuous Operation

Notes:

1. System – Mechanical system described in the referenced FSAR section.
2. Function Name – The automatic safety-related function is controlled by SAS in each mechanical system.
3. Function Safety Basis – Safety-related functions that provide reasonable assurance of either:
   - The integrity of the reactor coolant pressure boundary.
   - The capability to shut down the reactor and maintain it in a safe shutdown condition.
   - The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.
4. Interdivisional Communication – Point-to-point data communications between different safety divisions of SAS.
5. Type of Data – Analog or Discrete Signal. This column is meant to indicate the type of information sent between divisions, not the transmission means by which the information is sent (hardwired, data message, etc.).

Table 7.1-7—SAS FMEA Results
Sheet 15 of 30

No	System	SAS Function	Name of Sensor, Functional Unit, or Equipment (2)	Failure Mode (1)	Method of Detection	Inherent Compensating Provision	Effect on the SAS Function	Comments
Systems With Functions Utilizing Voting Logic								
50	In-Containment Refueling Water Storage Tank System (IRWST)	IRWST Boundary Isolation for Preserving IRWST Water Inventory Interlock (Figure 7.6-4)	Master CU in 1 Division	a) Detected Failure	TXS inherent or engineered fault detection mechanism	Affected division switches to the standby CU	Master / Standby CU switchover occurs in faulted division. Voting logic remains 2/4 in faulted division. Voting logic in other divisions is modified to 2/3.	No effects on the system function
				b) Undetected - Spurious	None	Redundant divisions/ trains	Spurious trigger of one division / train. Voting in other divisions becomes 1/3.	
				c) Undetected - Blocking	None	Redundant divisions/ trains	Loss of one division / train. Voting in other divisions becomes 2/3.	
~~51~~	~~Deleted~~							
~~52~~	~~Deleted~~							
51	Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Automatic Trip of LHSI Pump (in RHR Mode) on Low ΔPsat (Figure 7.6-9)	Master CU in 1 Division	a) Detected Failure	TXS inherent or engineered fault-detection mechanism	Affected division switches to the standby CU	Master/Standby CU switchover occurs in faulted division. Voting logic remains 2/4 in faulted division. Voting logic in other divisions is modified to 2/3.	No effects on the system function
				b) Undetected -Spurious	None	Redundant divisions/trains	Spurious trigger of one division/train. Voting in other divisions becomes 1/3.	
				c) Undetected - Blocking	None	Redundant divisions/trains	Loss of one division/train. Voting in other divisions becomes 2/3.	
52	Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Automatic Trip of LHSI Pump (in RHR Mode) on Low RCS Loop Level (Figure 7.6-10)	Master CU in 1 Division	a) Detected Failure	TXS inherent or engineered fault-detection mechanism	Affected division switches to the standby CU	Master/Standby CU switchover occurs in faulted division. Voting logic remains 2/4 in faulted division. Voting logic in other divisions is modified to 2/3.	No effects on the system function
				b) Undetected -Spurious	None	Redundant divisions/trains	Spurious trigger of one division/train. Voting in other divisions becomes 1/3.	
				c) Undetected - Blocking	None	Redundant divisions/trains	Loss of one division/train. Voting in other divisions becomes 2/3.	
53	Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Detection of RHRS Train Connected (Figure 7.6-13)	Master CU in 1 Division	a) Detected Failure	TXS inherent or engineered fault-detection mechanism	Affected division switches to the standby CU	Master/Standby CU switchover occurs in faulted division. Voting logic remains 1/2 in faulted division. Voting logic in connected division is modified to 1/1.	No effects on the system function
				b) Undetected -Spurious	None	Redundant divisions/trains	Spurious trigger of one division/train. Spurious trigger of 1/2 voting logic in connected division.	
				c) Undetected - Blocking	None	Redundant divisions/trains	Loss of one division/train. Voting logic in connected division becomes 1/1.	

Table 7.1-7—SAS FMEA Results
Sheet 27 of 30

No	System	SAS Function	Name of Sensor, Functional Unit, or Equipment (2)	Failure Mode (1)	Method of Detection	Inherent Compensating Provision	Effect on the SAS Function	Comments
				Systems With Functions Utilizing Voting Logic				
96	In-Containment Refueling Water Storage Tank System (IRWST)	IRWST Boundary Isolation for Preserving IRWST Water Inventory Interlock (Figure 7.6-4)	Loss of 1 Division	a) Detected Failure	TXS inherent or engineered fault detection mechanism	Redundant divisions/ trains	Loss of Master CU and Standby CU in faulted division. Voting logic in other divisions is modified to 2/3.	No effects on the system function
				b) Undetected - Spurious	None	Redundant divisions/ trains	One division sends a spurious actuation. Voting logic in other divisions becomes 1/3.	
				c) Undetected - Blocking	None	Redundant divisions/ trains	Loss of Master CU and Standby CU in faulted division. Voting logic in other divisions becomes 2/3.	
~~96~~	~~Deleted~~							
~~97~~	~~Deleted~~							
97	Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Automatic Trip of LHSI Pump (in RHR Mode) on Low ΔPsat (Figure 7.6-9)	Loss of 1 Division	a) Detected Failure	TXS inherent or engineered fault-detection mechanism	Redundant divisions/trains	Loss of Master CU and Standby CU in faulted division. Voting logic in other divisions is modified to 2/3.	No effects on the system function
				b) Undetected -Spurious	None	Redundant divisions/trains	One division sends a spurious actuation. Voting logic in other divisions becomes 1/3.	
				c) Undetected - Blocking	None	Redundant divisions/trains	Loss of Master CU and Standby CU in faulted division. Voting logic in other divisions is modified to 2/3.	
98	Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Automatic Trip of LHSI Pump (in RHR Mode) on Low RCS Loop Level (Figure 7.6-10)	Loss of 1 Division	a) Detected Failure	TXS inherent or engineered fault-detection mechanism	Redundant divisions/trains	Loss of Master CU and Standby CU in faulted division. Voting logic in other divisions is modified to 2/3.	No effects on the system function
				b) Undetected -Spurious	None	Redundant divisions/trains	One division sends a spurious actuation. Voting logic in other divisions becomes 1/3.	
				c) Undetected - Blocking	None	Redundant divisions/trains	Loss of Master CU and Standby CU in faulted division. Voting logic in other divisions is modified to 2/3.	

Table 7.1-7—SAS FMEA Results
Sheet 28 of 30

No	System	SAS Function	Name of Sensor, Functional Unit, or Equipment (2)	Failure Mode (1)	Method of Detection	Inherent Compensating Provision	Effect on the SAS Function	Comments
99	Safety Injection and Residual Heat Removal System (SIS/ RHRS)	Detection of RHRS Train Connected (Figure 7.6-13)	Loss of 1 Division	a) Detected Failure	TXS inherent or engineered fault-detection mechanism	Redundant divisions/trains	Loss of Master CU and Standby CU in faulted division. Voting logic in connected division is modified to 1/1.	No effects on the system function
				b) Undetected -Spurious	None	Redundant divisions/trains	One division sends a spurious actuation. Spurious trigger of 1/2 voting logic in connected division.	
				c) Undetected - Blocking	None	Redundant divisions/trains	Loss of Master CU and Standby CU in faulted division. Voting logic in connected division becomes 1/1.	
				CCWS Switchover Functions				
100	Component Cooling Water System (CCWS)	CCWS Common 1.b Automatic Backup Switchover of Train 1 to Train 2 and Train 2 to Train 1 (Figure 7.3-33)	Loss of 1 Division	a) Detected Failure	TXS inherent or engineered fault detection mechanism	Failed sensor marked invalid; two redundant train pairs.	Unable to automatically perform switchover function in the faulted division.	A second pair serves its associated heat loads. Adequate cooling is provided by the second train pair.
				b) Undetected - Spurious	None	Two redundant trains pairs	Spurious trigger of one pilot valve. Remaining pilot valves provide safety function.	
				c) Undetected - Blocking	None	Two redundant trains pairs	Loss of one pilot valve. Remaining pilot valves provide safety function.	
101	Component Cooling Water System (CCWS)	CCWS Common 2.b Automatic Backup Switchover of Train 3 to Train 4 and Train 4 to Train 3 (Figure 7.3-33)	Loss of 1 Division	a) Detected Failure	TXS inherent or engineered fault detection mechanism	Failed sensor marked invalid; two redundant train pairs.	Unable to automatically perform switchover function in the faulted division.	A second pair serves its associated heat loads. Adequate cooling is provided by the second train pair
				b) Undetected - Spurious	None	Two redundant trains pairs	Spurious trigger of one pilot valve. Remaining pilot valves provide safety function.	
				c) Undetected - Blocking	None	Two redundant trains pairs	Loss of one pilot valve. Remaining pilot valves provide safety function.	
102	Component Cooling Water System (CCWS)	CCWS Emergency Leak Detection – Switchover Valves Leakage or Failure (Figure 7.3-36)	Loss of 1 Division	a) Detected Failure	TXS inherent or engineered fault detection mechanism	Failed sensor marked invalid; two redundant train pairs.	Unable to automatically perform switchover function in the faulted division.	A second pair serves its associated heat loads. Adequate cooling is provided by the second train pair
				b) Undetected - Spurious	None	Two redundant trains pairs	Spurious trigger of one pilot valve. Remaining pilot valves provide safety function.	
				c) Undetected - Blocking	None	Two redundant trains pairs	Loss of one pilot valve. Remaining pilot valves provide safety function.	

Next File

equipment during normal plant operation, outages and under AOOs and PAs. See Section 9.4.5 for more information about the SBVS.

SIS / RHRS Pump Rooms Heat Removal

The SBVS has a safety-related function that maintains ambient conditions below the maximum limits for the rooms of the SIS/RHRS safety-related system components (GDC 60, GDC 61). The functional logic is shown in Figure 7.3-46—SBVS SIS / RHRS Pump Rooms Heat Removal.

CCWS / EFWS Valve Rooms Heat Removal

The SBVS has a safety-related function that maintains ambient conditions below the maximum limits for the rooms of the CCWS/EFWS safety-related system components (GDC 60, GDC 61). The functional logic is shown in Figure 7.3-47—SBVS CCWS / EFWS Valve Rooms Heat Removal.

Isolation of Mechanical Areas of Safeguard Building on Containment Isolation

The SBVS has a safety-related function to automatically isolate the Safeguard Building hot mechanical areas and initiate filtration of exhaust from the areas in the event of a containment isolation signal. This functional logic is shown in Figure 7.3-65—SBVS Isolation of Mechanical Areas of Safeguard Building on Containment Isolation. See Section 9.4.5.3 for more information about this function.

Iodine Filtration Train Electric Heater Control

At the start of an accident, power to both stages of the two-stage electric heater is switched on when the fans start and filter bank isolation dampers open. When the temperature downstream of the heater increases to 158°F, one stage of heater power is switched off. As the temperature downstream of the heater reaches 176°F, the second stage of the heater is also switched off. The functional logic is shown in Figure 7.3-66—SBVS Iodine Filtration Train Electric Heater Control. See Section 9.4.5.2.2 for more information about this function.

7.3.1.3.6 Safety Injection System/Residual Heat Removal System

Automatic RHRS Flow Rate Control

The SIS/RHRS has a safety-related function to provide RCS decay heat removal to reach cold shutdown, refueling modes and to control primary temperature. The function to automatically control the flow rate of the RHRS supports the safety-related function of providing decay heat removal by modulating the bypass control valve ensuring a constant flow rate through the LHSI heat exchanger. Maximum flow protection is also provided by closing the LHSI Heat Exchanger Main Control Valve in the event that flow remains above the desired setpoint after the bypass control valve is closed. The functional logic is shown in Figure 7.3-60—SIS / RHRS Automatic RHRS Flow Rate Control.

7.3.1.4 Essential Auxiliary Support Controls Functional Descriptions

7.3.1.4.1 Component Cooling Water System

The component cooling water system (CCWS) is a closed loop cooling water system that, in conjunction with the essential water service system (ESWS) and the ultimate heat sink (UHS), removes heat generated from the plant's safety related components connected to the CCWS (GDC 44). See Section 9.2.2 for more information about the CCWS.

Common 1.b Automatic Backup Switchover of Train 1 to Train 2

The safety-related function to perform an automatic switchover from Train 1 to Train 2 verifies that the CCWS is capable of fulfilling its safety-related function to remove heat from safety-related components on the CCWS Common 1.a and 1.b headers. The functional logic is shown in Figure 7.3-33—CCWS Common 1.b and 2.b Automatic Backup Switchover.

Common 1.b Automatic Backup Switchover of Train 2 to 1

The safety-related function to perform an automatic switchover from Train 2 to Train 1 verifies that the CCWS is capable of fulfilling its safety-related function to remove heat from safety-related components on the CCWS Common 1.a and 1.b headers. The functional logic is shown in Figure 7.3-33—CCWS Common 1.b and 2.b Automatic Backup Switchover.

Common 2.b Automatic Backup Switchover of Train 3 to 4

The safety-related function to perform an automatic switchover from Train 3 to Train 4 verifies that the CCWS is capable of fulfilling its safety-related function to remove heat from safety-related components on the CCWS Common 2.a and 2.b headers. The functional logic is shown in Figure 7.3-33—CCWS Common 1.b and 2.b Automatic Backup Switchover.

Common 2.b Automatic Backup Switchover of Train 4 to 3

The safety-related function to perform an automatic switchover from Train 4 to Train 3 verifies that the CCWS is capable of fulfilling its safety-related function to remove heat from safety-related components on the CCWS Common 2.a and 2.b headers. The functional logic is shown in Figure 7.3-33—CCWS Common 1.b and 2.b Automatic Backup Switchover.

Figure 7.3-60—SIS / RHRS Automatic RHRS Flow Rate Control

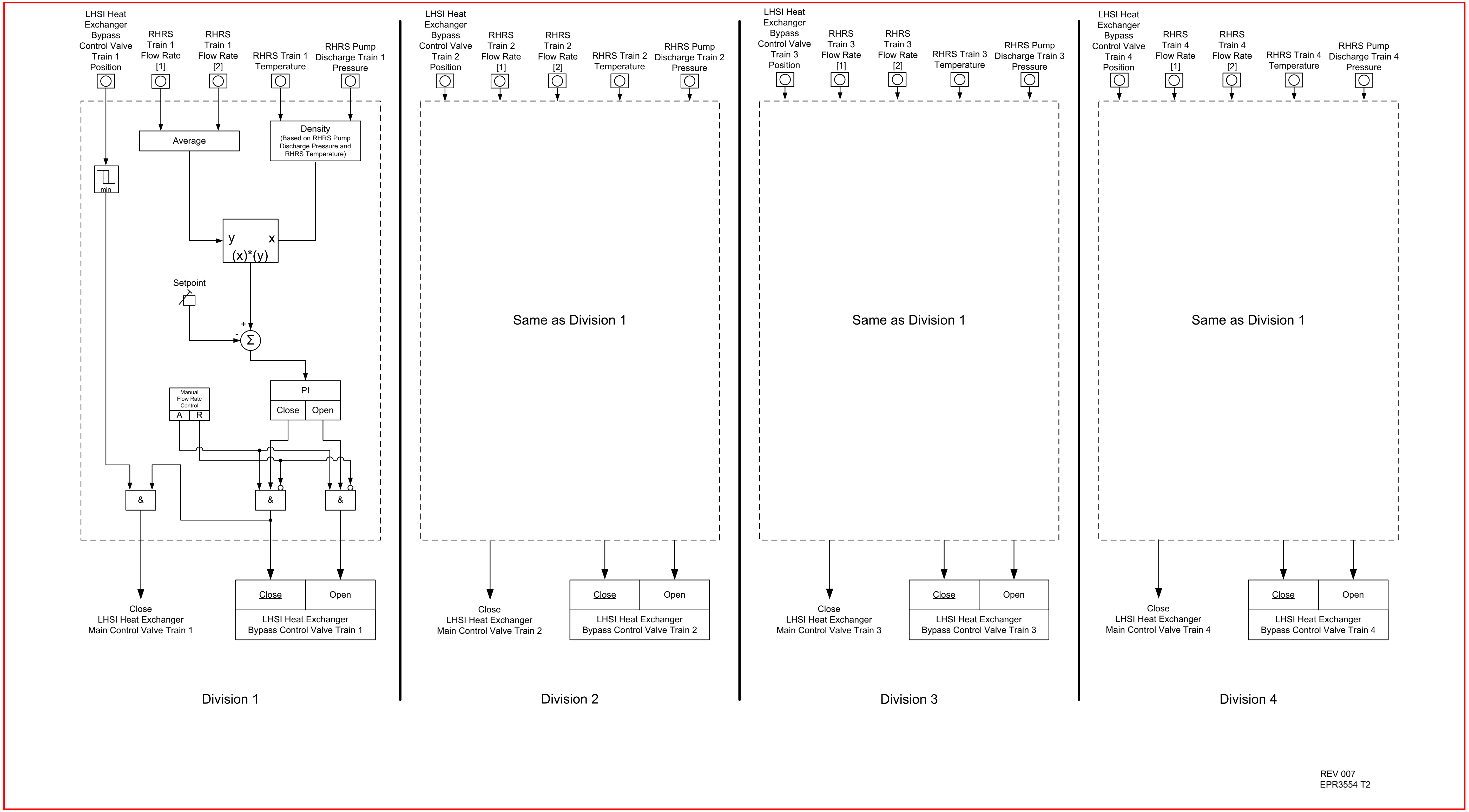
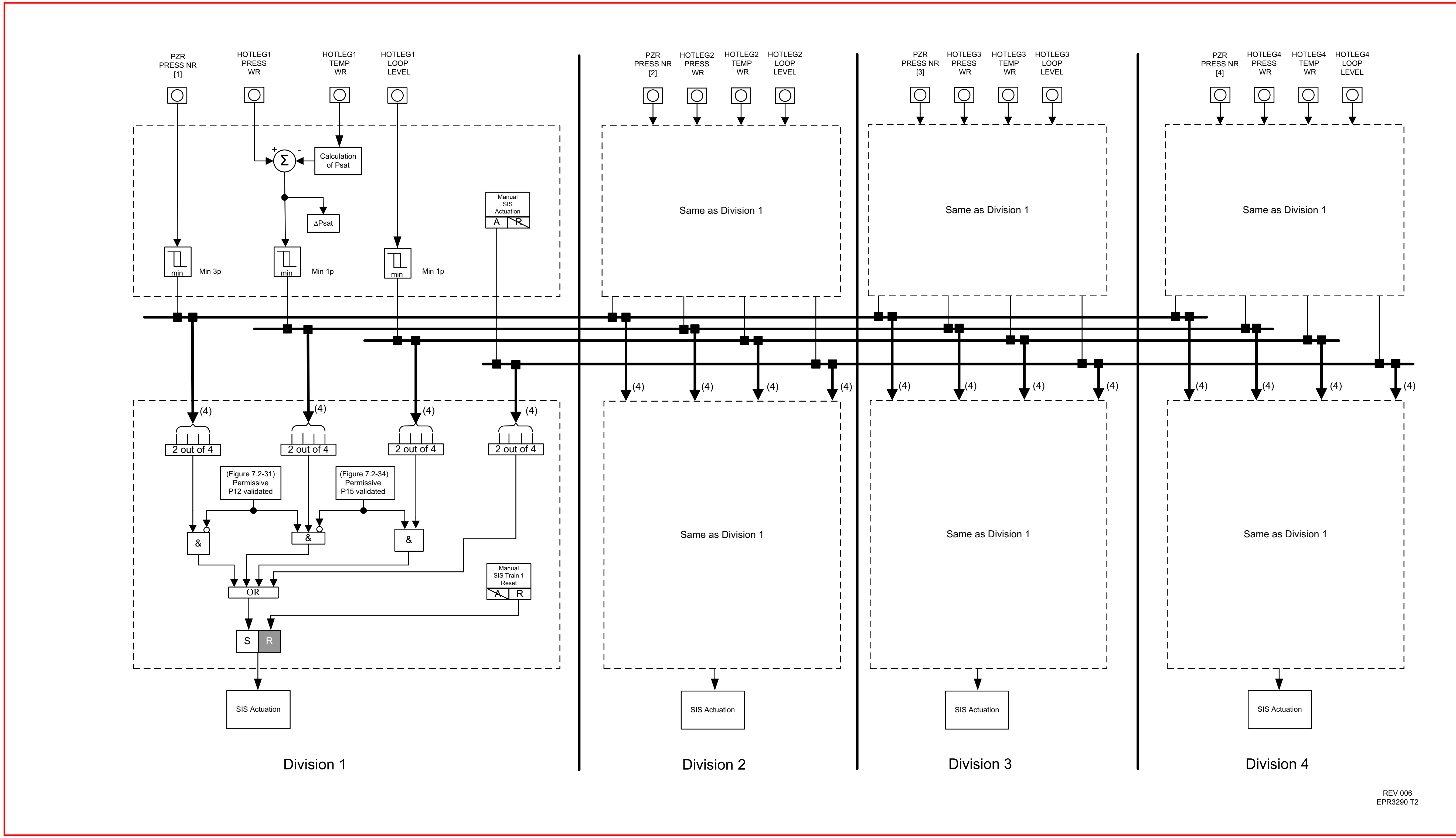


Figure 7.3-61—Deleted

Figure 7.3-2—SIS Actuation



the P14 permissive, providing a third diverse condition that must be satisfied to allow valve opening.

Another safety-related interlock prevents the opening of the RHR RCPB isolation valves, unless the LHSI suction isolation valve is closed. This prevents the LHSI suction from the IRWST from being exposed to the higher pressures of the RCS. The functional logic for this RHR isolation valve interlock is shown in Figure 7.6-11—RHR Isolation Valves Interlock.

When RHR is connected, an inadvertent increase in RCS pressure does not result in an automatic signal to close the RHR RCPB isolation valves. However, the following design features prevent an increasing pressure from exceeding the RHR system design pressure:

- Interlock holding the MHSI large miniflow lines open (see Section 7.6.1.2.8).
- Pressurizer safety relief valves operating in their LTOP mode (see Section 7.3.1.2.13).
- Spring loaded safety valves on the RHR suction lines.

During an intentional increase in pressure, when RCS temperature and pressure exceed the P14 permissive setpoint, the operator is prompted to manually inhibit the P14 permissive, and is then allowed to close the RHR RCPB isolation valves.

The operational status of the PS on a divisional basis is provided to the operator. Indications and alarms are provided to the operator regarding the state of the P14 permissive signal. Additionally, the following indications are provided to the operator to verify correct operation of the interlock:

- Open or closed position of first RHR RCPB isolation valve (each train).
- Open or closed position of second RHR RCPB isolation valve (each train).

The connection of a RHRS train can be detected based on the position of the following valves:

- First RHR RCPB isolation valve (each train).
- Second RHR RCPB isolation valve (each train).
- RHR outside containment isolation valve (each train).
- LHSI suction isolation valve (each train).
- Hot leg injection isolation valve (each train).

The functional logic for this detection of RHRS train connected is shown in Figure 7.6-13 - Detection of RHRS Train Connected.

7.6.1.2.2 SIS/RHRS Automatic Trip of LHSI Pump (in RHR Mode) on Low ΔPsat Interlock

The SIS/RHRS has a safety-related function to provide the RCS residual heat removal to reach cold shutdown and to control primary temperature. The function to automatically trip the LHSI pump upon a low ΔPsat signal supports the safety-related function of providing residual heat removal by maintaining LHSI pump operability by shutting down the pump to prevent pump damage due to inadequate net positive suction head (NPSH) or unavailability due to steam binding following a failure that results in RCS conditions approaching saturation. The functional logic is shown in Figure 7.6-9—SIS/RHRS Automatic Trip of LHSI Pump (in RHR Mode) on Low ΔPsat.

7.6.1.2.3 SIS/RHRS Automatic Trip of LHSI Pump (in RHR Mode) on Low RCS Loop Level Interlock

The SIS/RHRS has a safety-related function to provide the RCS residual heat removal to reach cold shutdown and to control primary temperature. The function to automatically trip the LHSI pump upon a low RCS loop level signal supports the safety-related function of providing residual heat removal by maintaining LHSI pump operability by shutting down the pump to prevent pump damage or unavailability due to air binding following a failure that results in low RCS loop level. The functional logic is shown in Figure 7.6-10—SIS/RHRS Automatic Trip of LHSI Pump (in RHR Mode) on Low RCS Loop Level.

7.6.1.2.4 Safety Injection Accumulator Interlocks

There are four accumulators, one associated with each of the four independent SIS trains. Borated water is injected into the RCS from the accumulators when RCS pressure falls below the internal pressure of the accumulators.

The operation of the SI accumulators is described in Section 6.3.

Each accumulator is connected to the cold leg injection line of its respective RCS loop through two check valves and a motor operated isolation valve in series. Each isolation valve is interlocked to remain fully open above a specified RCS pressure value in Modes 1, 2, 3, and 4. This pressure value is the P12 permissive threshold. The accumulators are used to provide safety injection into the RCS during higher pressures. At higher pressures (above P12 permissive value), the P12 inhibited signal is used to hold open the accumulator isolation valves. At lower pressures (below P12 permissive value), the open signal from the PS is removed and the operator can manually close the isolation valves. Isolation of the accumulator is necessary to prevent discharge of the accumulator at lower pressures.

Figure 7.6-9—SIS/RHRS Automatic Trip of LHSI Pump (in RHR Mode) on Low ΔPsat~~Deleted~~

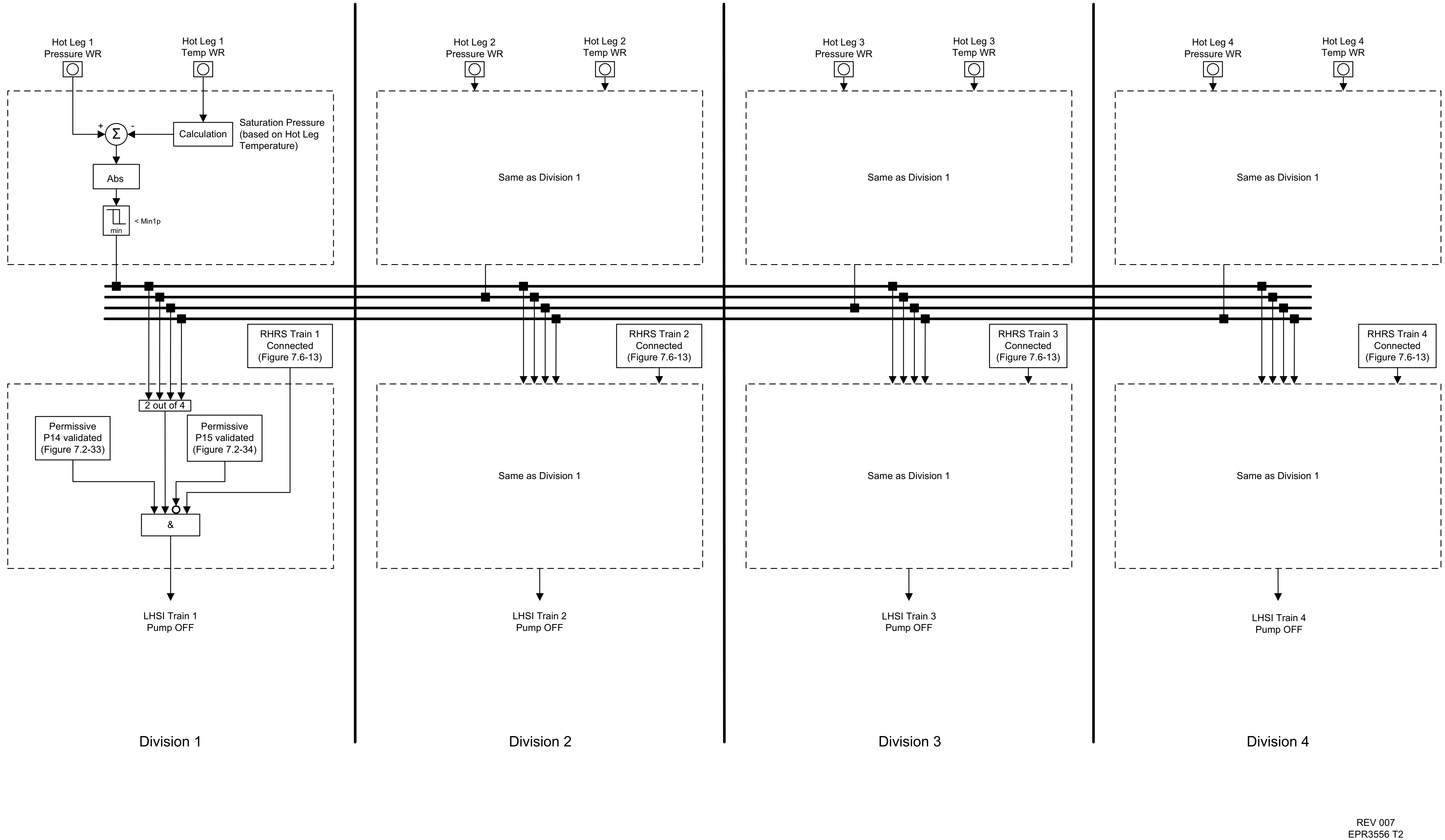


Figure 7.6-10—SIS/RHRS Automatic Trip of LHSI Pump (in RHR Mode) on Low RCS Loop Level~~Deleted~~

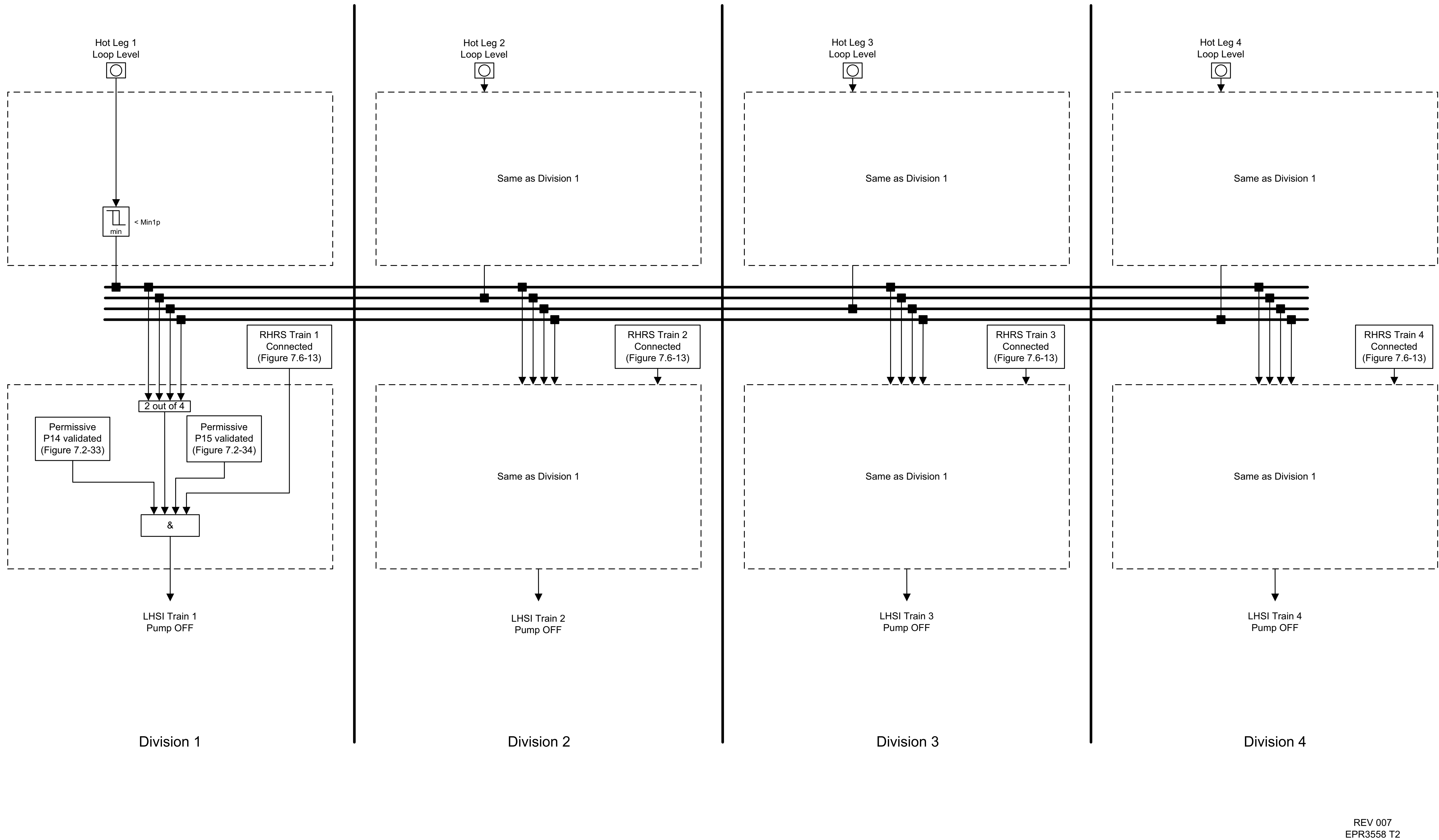
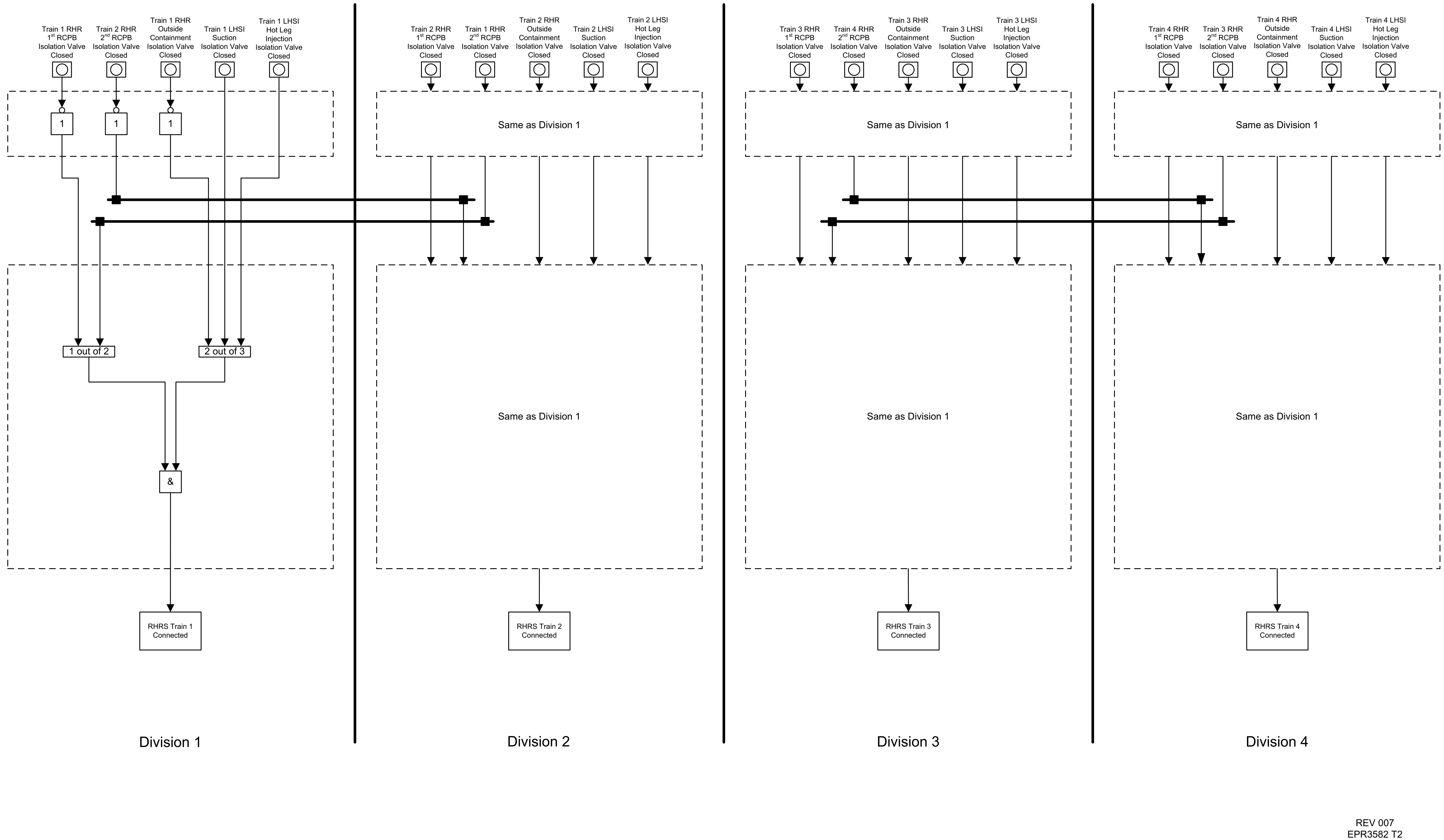


Figure 7.6-13—Detection of RHRS Train Connected



Next File

2.0 PREREQUISITES

2.1 Construction activities on the RCS mid-loop instrumentation system have been completed or exceptions have been recorded and the impact on system performance has been determined.

2.2 RCS mid-loop system instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.3 Support systems required for mid-loop operations are completed and functional.

2.4 Test instrumentation is available and calibrated. A record of calibrated test instrumentation used with individual tracking number and calibration due date shall be recorded in the official test record.

2.5 The RCS is at normal shutdown level, the pressurizer is drained and depressurized.

2.6 The RHR and mid-loop level instrumentation systems are functional.

2.7 Recommend installing an underwater camera in each RCS leg within sight of the RHR suction line but not close enough to introduce an unwanted effect.

3.0 TEST METHOD

3.1 Observe mid-loop operation of the RCS mid-loop level instrumentation, including indication, level controls, flow controls, and alarms.

3.1.1 Loop Level Control Function.

3.1.2 Max1 RCS Loop limitation Function.

3.1.3 Min1 RCS Loop Level Limitation Function.

3.1.4 Min1p RCS Loop Level Safety Functions.

3.2 Observe operation of the LHSI pump at minimum / maximum design flow conditions (e.g., motor current, pump vibration, system flow) while operating at various mid-loop levels.

3.3 Demonstrate that the LHSI pumps can operate without cavitation at the minimum mid-loop level and maximum design flow, for mid-loop conditions.

3.4 Demonstrate that the LHSI pumps can operate without excessive vibration at the minimum mid-loop level and minimum design flow, for mid-loop conditions.

3.5 Demonstrate that the RHR system can be throttled to the maximum allowable flow that prevents vortexing during mid-loop operation.

3.5.1 Take actions to prevent starting of the MHSI pumps.

3.5.2 Reduce RCS level to the Min1p RCS Loop Level.

3.5.3 Demonstrate RHR operation at the maximum allowable mid-loop flow without evidence of vortexing.

3.5.4 Remove power from the throttled valve(s) while operating at maximum flow and verify that the valve position fails as-is.

3.5.5 Gradually increase RHR flow until vortexing is observed.

3.5.6 Increase mid-loop level to normal level.

3.5.7 Restore MHSI ability to start on low mid-loop level and verify automatic start on Min1p RCS Loop Level.

3.6 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

4.0 DATA REQUIRED

4.1 Setpoints of alarms.

4.2 Mid-loop instrumentation data.

4.3 LHSI pump flow and vibration data.

4.4 LHSI pump performance data for limiting mid-loop design conditions.

5.0 ACCEPTANCE CRITERIA

5.1 The RHR and mid-loop instrumentation systems (e.g., controls, indication, alarms, interlocks) perform as designed (refer to Section 7.7.)

5.2 Verify that LHSI pump performance meets the following design parameters (refer to Section 7.7):

5.2.1 Maximum allowable pump vibration and temperature limits.

5.2.2 Minimum / maximum flow limiting features.

5.2.3 Acceptable indications of pump cavitation.

5.3 Verify that safety-related components meet electrical independence and redundancy requirements.

5.4 Verify mid-loop performance for each division.

14.2.12.2.6 Severe Accident Heat Removal System (Test #018)

1.0 OBJECTIVE

1.1 To perform the test described in this abstract on the severe accident heat removal system (SAHRS) simulating conditions that are representative of actual plant conditions or close enough to allow the data to be extrapolated to actual conditions.

1.2 To demonstrate that the SAHRS is properly installed and is functional prior to fuel loading.

1.3 To record data that is used to validate design basis assumptions or provide a baseline record of system performance for non-safety-related attributes.

Table 3.3.2-1
ESFAS Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED NUMBER	CONDITIONS	SURVEILLANCE REQUIREMENTS
1. SIS Actuation				
a. Low Pressurizer Pressure	1,2,3(a)	4 divisions	B,D,O	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.7
b. Low Δ~~Delta~~ P_{sat}	3	4 divisions	B,D,P	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.7
	4(b)(e)	3 divisions	B,D,P	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.7
c. Low Hot Leg Loop Level	4(f)	3 divisions	C,M	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.7
	5(f),6(f)	2 divisions	K	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.7
d. Manual	1,2,3	4 divisions	F,H,Q	SR 3.3.2.5
	4	3 divisions	E,M	SR 3.3.2.5
	5,6	2 divisions	X	SR 3.3.2.5
2. EFWS Actuation				
a. Low-Low SG Level (Affected SG)	1,2,3(c)	4 divisions	B,D,Q	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.7
b. Manual (Affected SG)	1,2,3	4 divisions	I,J	SR 3.3.2.5
	4(c)(k)	2 divisions	EE,FF	SR 3.3.2.5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required RHR loop is in operation~~ and circulating reactor coolant at a flow rate of ≥ 2200 gpm~~.	12 hours
SR 3.4.7.2	Verify SG secondary side water level is ≥ 20% in required SGs.	12 hours
SR 3.4.7.3	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required RHR loop is not in operation. --- Verify correct breaker alignment and indicated power are available to each required LHSI pump.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify required RHR loop is in operation~~ and circulating reactor coolant at a flow rate of ≥ 2200 gpm~~.	12 hours
SR 3.4.8.2	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after required RHR loop is not in operation. --- Verify correct breaker alignment and indicated power are available to each required LHSI pump.	 7 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.8 ECCS - Shutdown, MODES 5 and 6

LCO 3.5.8 Two Medium Head Safety Injection (MHSI) trains shall be OPERABLE.

~~-- NOTE --~~
~~The required OPERABLE MHSI trains may be removed from service for up to 24 hours with vessel level at or above mid-loop reactor vessel level provided no operations are permitted that would cause perturbation of RCS inventory.~~
~~---~~

APPLICABILITY: MODE 5,
MODE 6 with the refueling cavity not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MHSI train inoperable.	A.1 Restore required MHSI train to OPERABLE status.	72 hours
B. Two required MHSI trains inoperable.	B.1 Initiate action to restore at least one MHSI train to OPERABLE status.	Immediately
C. Required Action and associated Completion Time not met.	C.1.1 Initiate action to be in MODE 5 with the RCS pressure boundary intact and ≥ 25% pressurizer level.	Immediately
	<u>OR</u>	
	C.1.2 Initiate action to achieve refueling cavity water level ≥ 23 feet above the reactor vessel flange.	Immediately
	<u>AND</u>	
	C.2 Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation~~ and circulating reactor coolant at a flow rate of ≥ 2200 gpm~~.	12 hours
SR 3.9.5.2	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required RHR loop not in operation. --- Verify correct breaker alignment and indicated power are available to each required LHSI pump.	7 days

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific safety analysis and OPERABILITY requirements applicable to each protective function are identified below. Permissives that enable a credited function are included in the Technical Specifications.

LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," addresses the following Functions:

1. <u>Safety Injection System (SIS) Actuation</u>

a. SIS Actuation - Low Pressurizer Pressure

In the event of a decrease in RCS water inventory, the makeup is supplied by the Medium Head Safety Injection (MHSI) in the high pressure phase of the event and the Low Head Safety Injection (LHSI) in the low pressure phase. For a potential overcooling event, the reactivity insertion is limited by the boron injection via the MHSI. Even if the boron injection is not required, MHSI injection is needed to stabilize the RCS pressure. This Function mitigates the following postulated accidents or AOOs:

– Excessive increase in secondary steam flow,
– Steam Generator Tube Rupture (SGRT),
– Small break LOCA,
– Inadvertent opening of a pressurizer pilot operated safety valve,
– MSLB,
– Large break LOCA.

Four divisions of the SIS Actuation - Low Pressurizer Pressure Function are required to be OPERABLE in:

– MODES 1, 2, and
– MODE 3 with P12 permissive inhibited.

This Function utilizes the Pressurizer Pressure (Narrow Range) sensors.

The NTSP is sufficiently below the full load operating value for RCS pressure so as not to interfere with normal plant operation. However, the NTSP is high enough to provide an SIS actuation during an RCS depressurization.

BASES

LCO (continued)

During an event requiring ECCS MHSI actuation, a flow path is required to provide an abundant supply of water from the IRWST to the RCS via the ECCS pumps and to its associated four cold leg injection nozzles.

~~The LCO modified by a Note allows the required OPERABLE MHSI trains to be removed from service for up to 24 hours for personnel protection during RCS maintenance activities, such as installation of nozzle dams and replacement of reactor coolant pump seals, provided no operations are permitted that would cause perturbation of RCS inventory.~~

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2. MODE 4 OPERABILITY is covered by LCO 3.5.3.

In MODES 5 and 6, two OPERABLE ECCS MHSI trains are acceptable and provide for single failure consideration.

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "RHR Loops - High Water Level," and LCO 3.9.5, "RHR Loops - Low Water Level."

ACTIONS

A.1

With one required MHSI train inoperable, the inoperable MHSI train must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS MHSI train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

B.1

If two required ECCS MHSI trains are inoperable, immediate action must be taken to restore at least one MHSI train to OPERABLE status.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Containment Penetrations

BASES

BACKGROUND

Containment closure capability provides an added level of defense during reduced inventory conditions with fuel in the reactor vessel in the unlikely event of loss of core cooling. In MODES 5 and 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no requirement for containment leak tightness, the 10 CFR 50 Appendix J leakage criteria and tests are not required.

In MODES 5 and 6, the water inventory in the reactor vessel is controlled at varying levels. ~~For installation and removal of steam generator nozzle dams, the RCS is vented and the reactor vessel level is reduced to approximately mid-loop of the hot legs.~~ For reactor vessel head removal or installation, the reactor vessel level would be just below the reactor vessel flange. Prior to commencement of fuel handling operations, the level in the refueling cavity is maintained in accordance with LCO 3.9.6, "Refueling Cavity Water Level."

As discussed in NRC Generic Letter 88-17 (Reference 1) and in FSAR Section 19.1.6 (Reference 2), a loss of residual heat removal cooling can potentially lead to steam and radioactive material release from the RCS. As discussed in FSAR Section 5.4.7.2.1 (Reference 3), design features that support improved safety during shutdown and mid-loop operation include:

- During mid-loop operation, the RCS loop level is controlled by the CVCS low pressure reducing valve to ensure there is sufficient RCS water inventory for operation of the Low Head Safety Injection System (LHSI) pumps in residual heat removal (RHR) mode.
- Redundant hot leg level sensors that initiate RCS make-up when the RCS hot leg loop level has reached low level.
- Safety injection via Medium Head Safety Injection System (MHSI) with reduced discharge head during low loop level ensures availability of the LHSI pumps for RHR function.

BASES

ACTIONS (continued)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Ventilation System.

With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time and the features available to maintain RHR operation and vessel level (Ref. 1).

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one RHR loop is in operation. The flow rate is determined by the flow rate necessary to provide efficient decay heat removal capability and to prevent thermal and boron stratification in the core.

In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR loop flow rate determination must also consider the LHSI pump suction requirement. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the RHR System in the control room.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The minimum flow rate specified is to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.~~

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional LHSI pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience. This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. FSAR Section 5.4.7.

hours is required before any critical equipment reaches temperatures that would compromise equipment functionality.

- EBS Start for an ATWS event: This operator action is assumed to be required in less than 30 minutes.

- RHR Restart after a LOCA in POS C: The RHR pumps may be required to trip on a LOCA due to low coolant level, therefore requiring a manual restart. ~~Additionally if a significant LOCA occurs during shutdown, operator action may be required to manually trip the pumps (the LHSI pump trip on low RCS level is a non-safety function).~~

- LHSI Start after a loss of RHR in POS D: Provides inventory control after a loss of RHR during mid-loop.

- LHSI Start after a LOCA in POS D or E: Provides inventory control after a LOCA during POS with LHSI pumps not aligned for RHR.

- Open EFW suction crosstie valves to allow EFW inventory in EFW trains that are failed or unavailable to be utilized by EFW trains that are available (OPF-EFW-6H): There is ample time to perform the required action (greater than 6 hours when one train of EFW is available, and greater than 12 hours when 2 trains of EFW are available).

19.1.5.1.2.4 Key Assumptions and Insights

Assumptions and insights from the PRA-based seismic margin assessment are as follows:

- Plant level HCLPF – Based on the seismic margin assessment, it is concluded that the U.S. EPR HCLPF capacity will be equal to or greater than 1.67 times the CSDRS (0.5g pga). This conclusion is dependent on achieving the HCLPF commitments in Table 19.1-106 and additional activities after Design Certification as discussed in ISG-020.

- Seismic PRA model – The SMA analysis considers seismically induced LOOP, SLOCA, MLOCA, LLOCA, ATWS, and various shutdown initiating events. Equipment and structures that are not seismically qualified are not credited in the model. This treatment is judged conservative for a seismic margin assessment because of inherent seismic capacity and ruggedness that exists in non-seismic structures and equipment.

- The operator is important in protecting against a seismic event in shutdown conditions when the low pressure reducing station is in service (especially in reduced inventory conditions such as mid-loop). The automatic signal that closes the reducing valves on low RCS level is a non-safety signal, and the valves that are typically utilized to control letdown flow through the low pressure reducing station (the low pressure reducing station valves) are provided with a power supply that is not seismically qualified such that a significant seismic event will require operator action to isolate the LP reducing station, and will disable the

- RCS - Use of nozzle dams is not modeled in the LPSD PRA because the U.S. EPR design is based on a full fuel offload for RCS maintenance.~~Use of nozzle dams is not modeled in the LPSD PRA because they are not anticipated to be used in normal refueling outages. The U.S. EPR design is based on a full fuel offload, and use of nozzle dams is assumed only for cases of infrequent RCS maintenance activities, like mid-cycle outages for steam generator tube repair.~~

- CVCS – Charging pumps are not credited in shutdown.

- EFW – Auto reset of the P13 permissive is required for automatic EFW operation during POS C. Also, only the normal pressure control mode of MSRTs is required (MSSVs are not credited). A PCD function is disabled by P14 permissive, the MSRT pressure is set to 145 psia and is not automatically reset.

- RCP – Only two pumps are running during POS CAD and would be required to trip upon loss of motor cooling. Seal cooling is not required during shutdown.

The following summarizes LPSD systems with auto actuation signals modeled:

- RHR protective trip – Low loop level will trip the operating RHR pumps to protect the pumps and allow them to be restarted post trip either in RHR or the LHSI mode of operation. The level sensor taps are located on the top and bottom of each hot leg approximately ten feet from the steam generator center line and approximately six feet closer to the steam generator than the LHSI RHR suction line. This safety function is provided by SAS. ~~discharge nozzle. This non-safety function is performed by PAS.~~ Failure of this trip function is included as a failure mode of the RHR pumps. Success allows the pump to be manually recovered later.

- RHR isolation – High sump level in the SB automatically isolates the respective RHR train and trips the pump. This non-safety function is performed by PAS and is modeled in the RHR ISLOCA initiating event fault tree.

- Low pressure reducing station isolation – During an uncontrolled drain down event (ULD), low loop level automatic isolation of the low pressure reducing station is modeled. This non-safety function is performed by PAS; it initiates closure of the CVCS letdown in order to protect the RHR pumps and prevent challenge of the SIAS safety function. Failure is also assumed to result in diversion of IRWST water outside containment requiring containment isolation or operator response.

The probability of plugging the IRWST suction strainers is modeled the same as at-power operation (i.e., CCF). Maintenance work during shutdown could result in a higher probability of plugging. However, the IRWST design is somewhat unique in comparison to the PWR plants operating in the USA. The structure is very large with separation between suction lines to the four SB; three levels of filters are also provided: trash racks, retaining baskets, and six strainers with a back flush capability. This probability of plugging is also dependent on maintenance procedures that will be in place to control foreign material, but are not available in this phase. As a result, the present modeling of the IRWST suction strainers was not changed.

Table 19.1-94—U.S. EPR Risk-Significant Equipment based on RAW Importance – Level 1 Shutdown shows the top risk-significant SSC based on RAW importance. Most of the top SSC are from the HVAC and electrical system, including chillers crosstie MOVs, load centers, switchgears, MCCs, DC buses and safety batteries.

Table 19.1-95—U.S. EPR Risk-Significant Human Actions at Shutdown based on FV Importance – Level 1 Shutdown shows the top risk-significant human actions based on FV importance. The most important operator actions based on the FV are operator failure to isolate RHR flow diversion in states CA and CB, operator failure to isolate the CVCS low pressure reducing station, and operator failure to stop draindown at mid-loop. These actions are important because they are needed to prevent the occurrence or consequence of the important LPSD initiators.

Table 19.1-96—U.S. EPR Risk-Significant Human Actions based on RAW Importance – Level 1 Shutdown shows the risk-significant human actions based on RAW importance. The most important operator action based on RAW is the operator failure to isolate CVCS low pressure reducing station. This operator action is important because it prevents depletion of the IRWST in case of an uncontrolled level drop and failure of automatic isolation.~~This action is important because it is needed to prevent the occurrence of the important LPSD initiators: uncontrolled level drops.~~

Table 19.1-97—U.S. EPR Risk-Significant Common Cause Events based on RAW Importance – Level 1 Shutdown shows the risk-significant common-cause events based on RAW importance. The most important CCFs based on RAW importance are CCFs to open LHSI/MHSI common injection valves and CCF plugging of IRWST sump strainers. These events are important because both of these CCFs would disable all safety injection.

Table 19.1-98—U.S. EPR Risk-Significant Common Cause I&C Events based on RAW Importance – Level 1 Shutdown shows the significant common-cause I&C events based on RAW importance. As illustrated in this table, I&C common-cause events (e.g., I/O modules, software, sensors, or computer processors or SAS) have a high RAW. This is because a CCF of the signals could lead to an actuation failure of MHSI or EDGs (Protection system) or failure of ~~CVCS isolation on ULD~~ RHR and cooling chain control functions (SAS).

Table 19.1-99—U.S. EPR Risk-Significant PRA Parameters – Level 1 Shutdown shows the significant modeling parameters used in the analysis and the significant LOOP related basic events. The significance is determined based on either the FV or RAW importance measure, as defined above. This table illustrates the high significance (a high FV) of the parameters used to support cooling in the SBO conditions and in the modeling of shutdown initiating events (e.g., LOOP, induced LOCAs or ISLOCAs).

mid-loop and during plant startup after refuel. Given RCS temperatures and pressures, a loss of inventory in the form of steam was evaluated after a loss of RHR cooling. The pressurizer vent contains a flow restrictor, which significantly limits the flow well below the makeup capacity of the CVCS system. The RPV vent is a one-inch line, and it would take a large amount of time to uncover the core by venting steam through this line. The risk from this event is not considered significant because the operators have more than enough time to isolate the vent or to provide makeup to the RCS. Based on the above discussion, these events were not identified as transient LOCAs that need to be included in the analysis.

- Three of three PSVs are assumed to be required as in the power operation model for feed-and-bleed, which is conservative for shutdown (two of three is expected to be adequate and one of three is adequate post refueling).

- It is assumed that a transient-induced LOCA response requires feed-and-bleed cooling success, because LOCA size may not be large enough to provide sufficient bleed.

- The probability that the IRWST suction strainers are plugged was not increased relative to the power operation PRA model. The IRWST design (e.g., large, separation between suction lines, debris retaining capability) and plant procedures (e.g., foreign material control) are expected to ensure that this probability is low.

- Risk from the pressurizer solid state was not considered. Inadvertent start of a reactor coolant pump or a MHSI pump could cause an overpressure event when the pressurizer is solid. The PSVs and RHR relief valves would protect the system from overpressure and the exposure time is small. Thus, overfill events that could lead to a low temperature overpressure event have been considered not likely and have not been identified as initiating events that could significantly contribute to risk.

- ~~The EPR PRA does not specifically model nozzle dam installation since it is considered an infrequent evolution. The EPR PRA assumes that the core will be offloaded every fuel cycle. When nozzle dams are used, measures will be taken to reduce the CDF impact. To ensure that the risk of a sudden loss of RCS inventory during nozzle dam installation/removal and cold leg work does not have a large impact on the results of the EPR LPSD PRA, nozzle dams installation should be consistent with GL 88-17 and IN 88-36.~~

19.1.6.2.6 Sensitivity Analysis

A sensitivity analysis was performed to evaluate the impact of general modeling assumptions, most of them are also analyzed in Level 1 at power.

The sensitivity results are shown in Table 19.1-100—U.S. EPR LEVEL 1 Internal Events Sensitivity Studies – Level 1 Shutdown. Several insights can be drawn from the sensitivity cases analyzed.

Table 19.1-109—U.S. EPR PRA General Assumptions
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No.	Category[1]	PRA General Assumptions[2]
81	LPSD	~~Nozzle dams are not required during a plant shutdown, but may be used infrequently during mid-cycle maintenance, when full core off-load is not desirable. Appropriate RCS operating conditions will be considered in the specification of nozzle dams to provide reasonable assurance that nozzle dams will not fail.~~ ~~Plant procedures that cover reduced inventory operation will govern the installation of nozzle dams and the establishment of adequate venting to prevent pressurization of the RPV upper plenum due to a postulated loss of decay heat removal.~~ ~~Nozzle dams are the only U.S. EPR related~~The U.S. EPR design does not include any temporary reactor coolant system boundary as specified by NUREG-1449 and NUREG-1512. Freeze seals are not expected to be used; they will not be part of the maintenance procedures for the U.S. EPR. The U.S. EPR design does not include the use of nozzle dams.
82	LPSD	The efficiency of the Passive Autocatalytic Recombiners (PAR) during shutdown is assumed to be nominal. Maintenance unavailability, if any, is assumed to be limited to a small fraction of the PARs and would not affect the overall efficiency of the system.
83	LPSD	The RCS vents identified in state CB are not considered large enough to prevent RCS repressurization in the case of loss of cooling; therefore RCS repressurization is assumed in the time to boil calculation.
84	I&C	The principles and methods for defense against software CCF in the Protection System, including operating system features that reduce failure triggers and limit failure propagation, and lifecycle processes for application software development, (described in EMF-2110(NP)(A) and referenced in Section 7.1.1.2) are comparable to industry standards of good practice described in IEC-62340, IEC-60880, and IEC-61508 for safety integrity level four (SIL-4) applications.

Table 19.1-131—Key Sources of Uncertainties
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No.	PRA Category	Source of Uncertainty
10	I&C	**I&C impact on initiating event frequency:** Initiating event frequencies are based on historical operating experience with the conventional fleet. The digital control systems are expected to improve initiating event frequency over historic experience; however the specific impact of digital I&C on initiating event frequency, either positive or negative, is not quantified.
11	SYSTEMS-HVAC	**HVAC recovery:** HVAC recovery evaluation is based on the estimates of heat load of the electrical equipment and equipment responses to high temperature.
12	INTERNAL EVENTS - LOOP	**LOOP and DG Recoveries:** A simplified model is used to represent interaction between timings of LOOP and DGs recoveries.
13	FLOOD	**Flood Analysis:** A simplified and conservative flood analysis is performed because no detailed information is available on equipment and piping layouts, or on flood response procedures.
14	FLOOD	**Flood Frequency:** Flood Frequency is based on the number of piping segments. Given that a detailed piping layout is not available there was no detailed information on length of the piping, or number of welds. Also, non-piping elements were not considered in the frequency calculations.
15	FIRE	**Fire Analysis:** A simplified and conservative fire analysis is performed because no detailed information is available on equipment and cable layouts, combustible loads, transient combustible procedures, design of fire protection system, fire response procedures, etc.
16	FIRE	**Fire Frequencies:** Fire Frequencies are based on RES/OERAB/S02-01; NUREG/CR-6850 frequencies were not used because information is not available on total plant equipment (no design information on areas outside of the nuclear island), general equipment and cable layout and transient combustibles.
17	SD	**SD Schedule:** SD schedule and procedures are not available. Many assumptions are made based on the relevant plant experience, for example it was assumed that containment hatch will be closed in mid-loop operation, ~~that an use of the nozzle dams (and bypass on safety injection actuation) will be limited,~~ and that applicable GL 88-17 (and others) guidance will be followed.
18	L2	**Passive Equipment Performance:** No good data is available on estimating passive fuse valve failure likelihood.
19	L2 - HRA	**Interaction Between Operators and Emergency Response Centers:** Level 2 HRA is performed without evaluation in details a complex interaction between operators and emergency response centers.
20	L2	LRF Definition: Feedback from L3 to L2.